THE UNDERGROUND DISPOSAL OF
HIGH-ACTIVITY RADIOACTIVE WASTES

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SYNOPSIS

This thesis examines the practical engineering problems associated with high-activity radioactive waste disposal. It includes a critical review of all the available disposal options, but focuses on concepts involving deep underground burial in purpose-built repositories.

The construction and waste containment properties of crystalline, argillaceous and saliferous host rocks are examined and some inherent uncertainties in predicting their performance as natural barriers to radionuclide migration are described. It is shown that repository construction can introduce anomalous migration paths via peripheral zones of disturbance, rock/backfill separation planes and structural linings unless special preventative measures are taken. For jointed rock masses, it is shown that repository excavations may short-circuit natural flow paths.

Repository design proposals from the major nuclear power-producing countries are reviewed, with particular emphasis on crystalline rock repository systems. It is shown that although the international proposals demonstrate the broad feasibility of repository construction, they generally fail to achieve optimum solutions in terms of waste containment.
For jointed rocks, an optimisation study is described which demonstrates how adjustment of pre-disposal strategies for conditioning high-level wastes, altering the construction sequence, and manipulating the emplacement configuration can effect reductions in repository size to mitigate against the adverse effects of discontinuities. It is shown that applying the minimum repository size philosophy to other host rocks can also produce benefits in terms of waste containment and construction costs, and some radical design alternatives are proposed incorporating relevant principles.

The role of engineered barriers, in the form of high-integrity backfills and waste unit claddings, is examined. Potentially suitable backfills are identified by reference to geochemical stability and radionuclide containment properties, and cost implications are evaluated. Parametric sensitivity studies are performed to demonstrate the influence of waste unit and backfill properties on radionuclide containment, and some new practical design concepts and emplacement techniques are suggested.
TO

SUE, NINA AND LOIS
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AUTHOR'S NOTE

The work described in this thesis is related to two engineering research studies in the field of radioactive waste management and disposal undertaken on behalf of the following organisations:

- the United Kingdom Department of the Environment
- the Commission of the European Communities

These studies were completed by the author during the course of his employment with Mott, Hay & Anderson, consulting engineers (7, 9); and the thesis is a broad extension of relevant work, which was undertaken concurrently on the basis of a research studentship at the University of Surrey.

The author has drawn freely from the results of his professional engineering studies. However, it is emphasised that the opinions, conclusions and recommendations contained in this thesis are entirely his own, and do not necessarily reflect those of any of the organisations mentioned above.
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PART 1
BACKGROUND
1. GENERAL PERSPECTIVES

'Nuclear Power' is a phrase which gives rise to many emotive connotations and has become the subject of an intense and widespread international debate since its inception in the mid-1940s. During this period, the harnessing of nuclear power was researched by British, Canadian, American, French and other European scientists, under conditions of great secrecy, and was considerably hampered by a lack of readily available supplies of fissile uranium (78,79). Due to the exigencies of the period, however, the first manifestation of their remarkable scientific and engineering achievements was not the distribution of cheap and plentiful domestic and industrial energy; but the explosion of nuclear bombs on Hiroshima and Nagasaki, marking the close of the second world war.

It is an unhappy but inescapable historical consequence that military and political developments associated with nuclear power have since grown in parallel with the beneficial technological and economic achievements of the civilian nuclear industry. The enormous potential in both areas is due to the existence of vast world-wide reserves of natural uranium ore, the ability to extract the uranium metal and enrich the proportion of the fissile isotope uranium 235; and thence (through the use of thermal nuclear reactors), to produce significant quantities of plutonium.

Not unnaturally, the continued development of nuclear power for peaceful purposes has become inextricably linked, in the socio-political sense, with concern over the possible wide-spread proliferation of nuclear materials for military purposes. As a result, attempts
to address the technical issues associated with the safe
management and disposal of industrial nuclear wastes are
often obscured or hampered by complex non-technical
factors.

Until the 1970's, the scale of the problem was
relatively small and the management of high activity
radioactive wastes was not generally considered to be a
serious environmental problem. However, the last ten to
fifteen years has witnessed a marked acceleration in the
growth of world-wide nuclear energy capacity as shown in
figure 1. This reflects both an increase in world
energy demand and an attempt by the industrialised
nations to achieve a greater diversification and
self-sufficiency in methods of energy production. The
dramatic re-structuring of oil prices by OPEC since
1973-4, and international concern over the protection of
dwindling fossil fuel supplies has since reinforced the
commitment of many governments to the further expansion
of their nuclear power programmes (165).

At present, the world-wide nuclear power-producing
capacity amounts to approximately 160 GW(e)*, or 8% of
the 2000 GW(e) globally installed electrical capacity.
Of this total, nearly 98% is accounted for by the
industrialised OECD and centrally - planned (Comecon)
countries (65). Based upon the known number of
reactors, both planned and in current operation, it is
estimated that the world's nuclear generating capacity
will more than treble to 458 GW(e) by 1990, and will
then account for about 13% of the total world energy
supply. At that time, the industrialised nations as a

* 1 GW(e) is equivalent to an annual production of 1
Gigga Watt of electricity.
whole will probably account for about 94% of the world's nuclear power production, and the developing countries (notably those in Latin American and Asia) will have begun to acquire substantial nuclear power-producing capacity.

Figure 2 illustrates the pattern of this expansion on a geographical basis. The rapid acceleration predicted over the next decade is self-evident and, due to the long lead-times involved in the commissioning of nuclear reactors (generally between 8 and 15 years), there is no reason to doubt the validity of the forecasts. It is also clear that the industrialised nations will continue to dominate the trend towards a nuclear-based society for a considerable period.

Predictions beyond the year 2000 are more speculative and depend, inter alia, upon future development trends in third-world countries and the industrial success of alternative methods of energy production. However, current indications suggest an almost exponential increase in total energy demand over the next 20 to 30 years. Thereafter, significant increases will probably continue for an appreciable period, and it is likely that energy derived from nuclear fission will continue to play an important role for several decades.

Against this background, increasing attention has been focussed upon the environmental problems associated with the management and disposal of radioactive wastes. These comprise a very wide spectrum of gaseous, liquid and solid radioactive waste materials produced at all stages of the nuclear fuel cycle, from fuel fabrication to the decommissioning of reactors.
At present, industrial-scale disposal procedures exist only for radioactive wastes of low activity. These include some gaseous and liquid effluents which are routinely dispersed into the environment at low concentration levels; together with short-lived solid wastes of low activity which, after conditioning and packaging, are either dumped at sea or buried in shallow land-fill sites. The disposal procedures adopted for these 'low-level' wastes are subject to strict regulations in each of the countries concerned and, by common agreement, are based on guidelines formulated by the International Commission for Radiological Protection (ICRP), (117).

Corresponding radiation doses to the public are claimed to be minimal. In the United Kingdom, for example, the National Radiological Protection Board (NRPB) has estimated that the radiation dose received by the population due to current routine industrial discharges is only 0.1% of the total due to background radiation (147).

Considerable public controversy has occurred in relation to these gaseous and liquid radioactive waste emissions and current procedures for the disposal of solidified low-level wastes. Nevertheless, there exists a status quo in which the monitoring of radioactivity in the environment can and does provide a basis for rational argument. However, high activity radioactive wastes, for which no established disposal routes are available, are accumulating in increasing volumes. These comprise the more hazardous 'intermediate-level' and 'high-level' wastes which are currently stored at reactor sites and special nuclear establishments throughout the world. They are being produced at an increasing rate.
and a substantial back-log exists on an international scale.

Despite isolated incidents, involving the leakage of large amounts of radioactivity, current methods of high activity radioactive waste storage are claimed to be satisfactory by those responsible for their safe-keeping (180). However, two important factors demand that a more permanent and effective solution should be adopted within the near future. The first of these is the rapid reduction in the reserve storage capacity of many established nuclear sites. The second is the increasing level of technical awareness and public concern over the radiological safety issues and nuclear proliferation risks involved.

The former problem may be resolved at a national level by constructing large-scale, secure, centralised storage facilities capable of receiving predicted waste arisings from present and planned nuclear reactors within each of the countries concerned. However, the radioactive lives of several of the radionuclides involved are measurable in many thousands of years. Long-term storage is therefore seen to be satisfactory only to the extent that its safety can be assured by an almost indefinite succession of well-informed and technically competent custodians.

A significant moral issue is seen to be at stake, in which it is argued that those responsible for the production of long lived, high-activity radioactive wastes must also devise and implement safe and effective procedures for their disposal. The objective is to protect the environment and prevent the imposition of an unwarranted burden upon future generations. The term 'disposal' in this context therefore means the isolation
of the wastes from man's environment in a manner which requires no long-term surveillance or monitoring, and imposes no significant risk of adverse health effects in the future. The argument has far-reaching implications for present-day industrialised societies; since, if the principles involved are broadly accepted, failure to achieve an effective solution should logically lead to an abandonment of further nuclear power production.

In certain countries, concern over these issues has already begun to impose restrictions upon further industrial nuclear development. In Sweden, for example, legislation was enacted in 1977 to prevent the granting of licences to fuel new reactors unless corresponding proposals for the management, conditioning and disposal of radioactive wastes were submitted to Parliament by the utilities concerned and were accepted by the government as being 'absolutely safe'. The definition of 'absolutely safe' has also posed a dilemma for the Swedes, and led to a crisis of confidence and a national referendum in 1980 (165). As a result, Sweden has decided not to construct any new nuclear reactors beyond those planned or in current operation; see Chapter 13.

In the Netherlands, the planned construction of six nuclear reactors during the course of the 1990's has been a source of similar national controversy. Legislation has been passed which links the introduction of a nuclear power industry to the development of an acceptable system of high-activity radioactive waste disposal within the next few years.

Similar trends may be observed in other industrialised countries; although it is notable that in the United States, Canada, France and Britain, the public-at-large has not yet brought the issue of radioactive waste
management and disposal to the fore-front of domestic politics. These countries were the pioneers of nuclear power, and their governments and nuclear establishments are able to refer to a record of significant industrial achievements over some thirty years; including a safety record whose statistics are demonstrably better than those of rival energy industries over the same period (89, 206).

In these countries, a tradition has developed in which the arguments of the anti-nuclear lobby, sometimes well-reasoned and sometimes desperate, are painstakingly countered in scientific and statistical terms by the nuclear establishment. For operational aspects of nuclear production, this involves the assessment of relatively short-term risks; based on medical science and the extrapolation of past industrial performance (139, 205). The statistical evidence which has been produced is generally re-assuring and convincing. Public relations exercises have flourished, increased safeguards have been implemented, and the majority of investigators have been satisfied.

Statistics of this kind, however, cannot conceal the growing stockpiles of high-activity radioactive wastes; and assurances that safe disposal solutions are at hand are not readily accepted at face value by the lay community. Much of this scepticism may be attributed to the fact that radiation in the environment cannot be seen, heard or felt, unlike the more familiar and tangible health hazards associated with fossil-fuelled power stations. Radiation is also more readily associated with a risk of insidious long-term health effects, such as cancer and genetic defects in off-spring. This has led to considerable fears, although there is little evidence to suggest that the
incidence of genetic defects in humans has been increased significantly as a result of civil or military nuclear radiations to date (206); whereas the wastes discharged from traditional fossil-fuelled power stations are known to have caused a significant increase in the occurrence of several debilitating or fatal diseases, including cancers (89).

An equally important cause of concern is the time-scale of the problem. In view of the longevity of the wastes under consideration, experimental verification of the effectiveness of any isolation procedure must be regarded as impossible. Furthermore, safety assessments based on the projection of hypothetical risks over many thousands of years not only stretch the minds of scientists, but are a mystery and an anathema to the general public.

In combination, these various factors are regarded by many as the Achilles Heel of the civilian nuclear power industry. Significantly, they have also begun to enter the more long-standing international debate concerning the continued development of nuclear weapons.

The first high-activity radioactive wastes were produced in the United States during the Second World War, as part of the allied nuclear weapons programme. Thus far, U.S. strategic nuclear weapons development has generated (in equivalent solidified volumes) some 0.2 M m$^3$ of high-level waste, which is about 700 times more than the 300 m$^3$ which has accumulated from the operation of American commercial nuclear power plants to date. The backlog of civilian reactor wastes in the United States is not expected to reach even 10% of the volume of military waste until the end of the century (118).
However, despite the relatively recent upsurge of public interest in these matters, the problems of high-activity radioactive waste management and disposal have long been recognised by nuclear scientists, and serious research programmes have been in progress for many years. These have been undertaken largely under the auspices of the civilian nuclear establishment and have intensified considerably over the past ten years or so, in parallel with the world-wide growth in nuclear energy capacity.

Some varied and interesting proposals have been put forward, with varying degrees of success. However, the solution most widely advocated involves the burial of the wastes in deep-level underground repositories. It is claimed that by careful conditioning and packaging of the wastes, and with judicious repository site selection, deep underground burial will effectively isolate the wastes from the biosphere until their radioactivity has decayed to negligible levels. The difficulty of this task will be greatest for high-level wastes; which not only contain large concentrations of extremely long-lived radionuclides, but also contain highly radioactive fission products which emit significant quantities of heat for several hundred years after their production.

The radioactive waste disposal problem clearly has many complex ramifications. This thesis examines the technical aspects of the problem from basic principles; commencing with the nature and origin of high-activity radioactive wastes and the various alternative disposal procedures which have been studied. Emphasis is placed on the disposal of high-level wastes, although the special problems associated with intermediate-level wastes are also examined. The thesis also concentrates primarily on crystalline host rocks of the granitic or
gneissic type; since they pose particularly interesting problems in terms of waste containment. References to other host rock environments are made largely for comparative purposes.

Unfortunately, to a large section of the engineering community, research and development in the field of radioactive waste management and disposal has appeared somewhat esoteric, if not abstract. However, the problem is certainly a real one, since the wastes are with us now and increasing volumes will be produced for a substantial period. Additional radioactive waste management problems, associated with the de-commissioning of nuclear reactors and military nuclear facilities, are also waiting to be solved. The public demands that an effective high-activity waste disposal solution must be found within the forseeable future and claims its right to be the ultimate arbiter in the assessment of any repository scheme.

In the author's view, it is essential that the final disposal solution must not only be capable of surviving the closest possible technical scrutiny; but must also convey an intuitive sense of correctness to the uninitiated. Furthermore, it must be economically viable if the nuclear industry and nuclear society as a whole are to thrive in the future. Hence, above all, the solution must be well-engineered. This thesis therefore deliberately adopts a broadly-based engineering approach to the subject.
2. SCIENTIFIC BACKGROUND

2.1 Introduction

A full discourse on nuclear physics is beyond the scope of this thesis. However, a basic account of some relevant scientific principles is a necessary starting point from which to describe the origin and properties of high-activity radioactive wastes. The outline presented in this chapter is therefore included for completeness. It is largely based on a number of authoritative technical and historical reviews (3, 39, 79, 80, 107, 163); together with a few more detailed scientific references which are separately identified.

Historically, it was the discovery of radioactivity which led to modern ideas concerning the structure of the atom, and not vice versa as might be supposed. However, it is more convenient here to commence with a simplified description of atomic structure; and thence to outline the relevance of radiation phenomena and the principles of nuclear fission.

2.2 Atomic Structure

Although a large number of sub-atomic particles have been identified during the last forty years, it is sufficient for present purposes to consider electrons, protons and neutrons as being the fundamental components of atoms. The familiar solar system analogy is a useful guide; but quantum mechanics theory precludes the visualisation of the electrons as occupying definable positions in pre-determined 'planetary orbits'.

11
Each atom is nearly all empty space, the electron 'cloud' having a radius some ten thousand times that of the central nucleus. The nucleus is an extremely dense structure, comprising a number of protons and neutrons in close combination. Protons and neutrons have almost identical mass and may be regarded as relatively heavy; the mass of a proton being approximately $1.84 \times 10^3$ times that of an electron.

Whereas protons are positively charged, neutrons are electrically neutral. Thus, the number of protons is equal to the number of units of net positive charge on the nucleus. This is exactly balanced by the number of negatively-charged moving electrons held in the outer structure, and hence overall electrical neutrality is maintained.

Chemical reactions between elements are controlled by the transfer or sharing of electrons; and the chemical properties of materials are therefore determined by the number and arrangement of electrons in a neutral atom. Hence the number of protons (which determines the number of electrons) is a fundamental property which identifies each chemical element. This number is the atomic number, $Z$, which is a constant for all the atoms of a particular element.

Although the 19th century chemist, Mendeleev, formulated his periodic table on the basis of atomic weights, the modern periodic classification of the elements is based on atomic numbers and represents a more fundamental scientific interpretation of the results of earlier observations.
A fully extended periodic classification of the elements is shown in figure 3, and will be referred to frequently in later sections of this thesis. The elements are listed from left to right in successive rows, in order of increasing atomic number. Thus, the lightest element is hydrogen, having an atomic number $Z = 1$ (i.e. the nucleus contains a single proton). The heaviest known element, hafnium has an atomic number $Z = 105$ (i.e. the nucleus contains 105 protons).

As previously noted, the atomic number of each element identifies it uniquely in chemical terms (the use of chemical symbols is therefore strictly unnecessary although it is found to be more convenient and meaningful in practice). However, the number of neutrons within the nucleus of a given element is not necessarily constant. For a given number of protons, the number of neutrons may vary to produce nuclei of identical charge but unequal mass. These variants are termed isotopes, and they differ from one another in terms of physical rather than chemical properties. The element iodine, for example (atomic number 53) has 21 different isotopes; their atomic numbers and hence their chemical properties being identical.

Since the isotopes of all the elements are distinguished in terms of nuclear structure, they are termed nuclides. Different nuclides are identified by quoting the total number of protons and neutrons. This figure gives the total number of units of mass of the nuclide and is called the mass number, defined by the relationship:

$$A = Z + N \quad (1)$$

where $A$ is the mass number
$Z$ is the atomic number
$N$ is the number of neutrons
Based on the above relationship, specific nuclides can be fully identified by the general formula $^{A}_{Z}X$, where $Z$ represents the chemical symbol for the element. For example, uranium, the heaviest of the naturally-occurring elements has atomic number 92 and is found as a mixture of two isotopes having mass numbers 235 and 238; i.e. with 143 and 146 neutrons respectively.

The full scientific notation which identifies these two uranium nuclides would thus be $^{235}_{92}U$ and $^{238}_{92}U$. However, since the atomic number is always repeated for the same element and is also defined by association with the chemical symbol, it is common practice to identify nuclides in terms of their chemical symbols and mass numbers, e.g. $^{235}U$ and $^{238}U$. This abbreviated form of nuclide structure representation will be generally used hereafter.

Clearly, the variations in composition of all nuclides are not fully represented in the periodic classification shown in figure 3. A complete listing of all the known nuclides (isotopic species) would provide a total in excess of 1200 (compared with the 103 known elements).

These various nuclides all differ in terms of their physical properties, due to variations in nuclear composition. Differences in mass have already been described. However, as will be described in subsequent sections, the number and type of particles (nucleons) contained in each nuclide also determines its stability in terms of the internal forces which bind the particles into a coherent, dense nuclear structure.

Two different kinds of nuclear force may be identified.
The first is an electrostatic (Coulomb) force of repulsion between protons, which occurs as a result of their positive charge. This force tends to disrupt the nucleus and is a relatively weak, long-range force which is similar in nature to the electrostatic force of attraction which maintains the outer electrons in a structured series of energy bands.

The second nuclear force is less well-understood. It occurs only within the nucleus itself, acts over an extremely short-range, and is always attractive; irrespective of the charges held by the various particles within the nuclide. For very small separations, this 'binding force' is extremely intense, giving rise to the close-packed configuration of all atomic nuclei.

However, the existence of these two opposing nuclear forces means that only certain combinations of protons and neutrons can form stable nuclei. For relatively low mass numbers, an approximately equal number of protons and neutrons gives a high degree of nuclear stability. However, as the mass number increases, the effect of proton-proton repulsion becomes more prominent and the proportion of neutrons must increase to maintain nuclide stability. Since the naturally-occurring isotopes of the heavy element uranium contain approximately 1.5 times as many neutrons as protons, they may be regarded as relatively unstable.

2.3 *Radioactivity and Radioactive Decay*

The emission of radioactivity by the nuclei of heavy elements demonstrates that they are physically unstable. The phenomenon was first discovered in natural
uranium-bearing minerals by Henri Bequerel in 1896.

In 1898, Marie Curie commenced a systematic investigation of several elements to see if they showed similar effects. She detected a slight radiation only in thorium; but by chemical treatment and separation of uranium-bearing minerals she discovered two new elements, polonium and radium; both of which she found to be radioactive. A number of other radioactive substances were subsequently discovered in uranium and thorium-bearing minerals; all of which were found to be elements of high atomic weight.

In 1899, Rutherford and others carried out studies which demonstrated that the radiation emitted by uranium was non-homogeneous. He identified two types which he termed alpha and beta rays. Subsequently, Villard discovered a third kind which he termed gamma rays.

The nature and properties of these three types of radiation are of fundamental importance in the context of this thesis and are therefore described below:-

- **Alpha rays** (\(\alpha\)-rays) are streams of heavy, positively-charged particles (\(\alpha\)-particles) which are identical to the atomic nuclei of the gas helium. Thus, they are nuclides with mass number 4 and atomic number 2 (\(^4\)He\(_2\)) and comprise two protons and 2 neutrons in close combination. They emerge from radioactive parent nuclides with great energy. However, because of their double charge and appreciable mass they are not deeply penetrating and can be stopped by a sheet of thin perspex or thick paper.
Beta rays ($\beta$-rays) are streams of particles ($\beta$-particles) each of which has a mass and charge of equal magnitude to that of an electron. The term beta particle is used to distinguish electrons emitted by unstable nuclides from those occurring in the outer electron structure of atoms; and because the charge on a beta particle may sometimes be positive; in which case the beta particle is called a positron. Beta rays are moderately penetrating, but can be stopped by a thin sheet of light metal such as aluminium.

Gamma rays ($\gamma$-rays) are electromagnetic waves without either mass or charge (of the same nature as X-rays or light rays). Because of their short wave-length, they are deeply penetrating and can be stopped only by a considerable thickness of lead or other high-density material.

The penetrating power of these radiations also depends upon their energies. The energy of all types of radiation is measured in terms of the electron volt; equivalent to the energy gained by an electron on passing through a potential difference of 1 volt ($1eV = 1.6 \times 10^{-19}$ joules approximately).

It is apparent that the emission of radiation must be accompanied by internal energy changes in the unstable parent nuclide. Unstable nuclides are termed radionuclides, and most types undergo gradual change to a more stable form by emitting alpha or beta radiation.

Alpha particles bear the largest charge and mass, and are therefore associated with relatively large energy changes. Beta particles arise through the
disintegration of neutrons into protons and electrons; with emission of the latter in the form of beta radiation. The emission of a beta particle in the form of a positron implies that a proton has disintegrated into a neutron and a positron. In either case, additional energy is normally emitted simultaneously in the form of gamma radiation.

It is clear that the emission of particulate forms of radiation involves changes in mass, and thus the parent nuclides must be transferred into nuclides of other elements. This is best illustrated by reference to figure 3. Emission of an α-particle reduces the atomic number by 2 and thus leaves behind an element two places lower in the periodic classification. Emission of a beta particle in the form of an electron implies a gain in net positive charge on the nucleus; increasing the atomic number by 1 and leaving an element one place higher up. Similarly, emission of a beta particle in the form of a positron reduces the atomic number by 1 and creates an element one place lower down. This law, known as the 'radioactive displacement law' was first enunciated by Russel and Soddy in 1913.

At around the same time, the law of radioactive decay was formulated. This law states that the rate at which a quantity of a given radioactive element is transformed (decays), at any instant is proportional to the number of radionuclides present. In its mathematical form the law may be expressed:

\[-dN = \lambda N \]
\[dt\]

where \(N\) is the Number of atoms present in a given sample, \(\lambda\) is a proportionality factor, known as the
On integration, the above equation becomes

\[ N = N_0 e^{-\lambda t} \] ......................... (3)

or

\[ N_0 = N e^{\lambda t} \] ......................... (4)

where \( N_0 \) indicates the number of radionuclides originally present.

\[ N \] is the number present after the elapse of time \( t \).

Hence, the number of radiogenic daughter nuclear atoms, \( N_d \), formed during a period of time \( t \) is given by

\[ N_d = N_0 - N = N(e^{\lambda t} - 1) \] ......................... (5)

or

\[ \frac{N_d}{N} = (e^{\lambda t} - 1) \] ......................... (6)

and, solving for \( t \) in equation (4) above gives

\[ t = \frac{1}{\lambda} \ln \left(1 + \frac{N_d}{N}\right) \] ......................... (7)

The above equations form the basis for all calculations concerning radioactive decay or time-dependent energy emission by radionuclides. The more readily envisaged notion of radioactive half-life follows from equation 6, since if \( T_{\frac{1}{2}} \) is the time required for a given amount of radioactive material to decay to one half of its initial value, then

\[ T_{\frac{1}{2}} = 0.693 \lambda \] ......................... (8)

The half-life of a particular radionuclide is thus the
period at the end of which one half of the atoms initially present will have decayed. Of those remaining, one half will then decay during an identical half-life period, and so on. Each radionuclide species has a half-life which is uniquely characteristic and provides, in conjunction with the energy and nature of the radiation emitted, a means of identification.

Figure 4 shows the relationship between the proportion of the original quantity of radioactive material and elapsed time, expressed in multiples of the characteristic half-life period. As shown, the amount of radioactive material present decays to $10^{-6}$ times its initial value after a period equivalent to about 20 half-lives, or $10^{-9}$ times its initial value after 30 half-lives. If released to the environment, the amount of radioactive matter present at these times might or might not be considered negligible, according to the original quantity of material and its radiological toxicity.

When the daughter nuclide being formed by the decay of a parent nuclide is itself radioactive, the situation becomes more complicated. Suppose there are three radioactive substances A, B and C which form part of a larger radioactive decay chain; A decomposing into B and B into C. Then from equation 2, it is apparent that substance B is being formed at the rate $\lambda_A N_A$ and is decomposing into C at the rate $\lambda_B N_B$. Hence the net rate of growth of substance B is $(\lambda_A N_A - \lambda_B N_B)$, or

$$\frac{dN_B}{dt} = \lambda_A N_A - \lambda_B N_B \quad \cdots \quad (9)$$

If substance A is very long-lived compared with B, then the rate of formation of B is effectively constant.
However, its rate of decay increases as more of it is formed. This process continues until sufficient quantity of B has accumulated to make its rate of decay equal to its rate of formation; i.e. $\frac{dN_B}{dt} = 0$.

The amount of substance B present will then decay at the same rate as substance A, and $\lambda_A N_A = \lambda_B N_B$. In this condition, the parent nuclide A and daughter product B are said to be in radioactive equilibrium, and the relative quantities of the two substances at any time are inversely proportional to their decay constants and hence directly proportional to their half-lives.

A different situation arises where the daughter product has a long lifetime compared with the parent; since almost all of the parent radionuclide will have decayed before any appreciable decay of the product has occurred. The product will then decay with its own characteristic half-life. However, if the short-lived product B is in equilibrium with a parent A whose half-life is long compared with the second product C, then all three substances will reach equilibrium. The equilibrium between B and C will then have the same form as that between A and B described above.

Hence a whole series of radionuclides undergoing successive transformations will reach equilibrium if the half-life of the ultimate parent is long compared with any of the daughter products. When in equilibrium, the quantity of each radionuclide in the decay chain present at any time is proportional to its lifetime.

Radioactive substances exhibit an enormous range of half-lives, varying from millions of years to fractions of a second. Significantly, the actinide series of
elements, which are all present in spent nuclear fuel (see figure 3), have long half-lives and form the parent members of long and complex decay chains, in which the various daughter products gradually attain a state of radioactive equilibrium with their precursors.

A description of the complete scheme of these decay chains is beyond the scope of this thesis. However, it is important to note that, with the passage of time, a complex 'cocktail' of radionuclides will emerge, including α and β emitters; the successive radioactive displacements generally culminating in stable isotopes of lead. Thus the various radiogenic daughter products present in the equilibrium condition can exhibit a very wide range of radioactive characteristics and chemical properties. In the context of radioactive waste disposal, it is necessary to identify the most hazardous of these in radiological terms, in order to devise appropriate physical or chemical means of isolation or containment.

2.4 Nuclear Energy and Nuclear Fission

From about 1920 onwards, considerable experimental and theoretical progress was made in the investigation of energies associated with radiation phenomena. Using a mass spectrograph, F.W. Asten demonstrated that the masses of individual atoms were somewhat smaller than the sum of the masses of their constituent particles. He concluded, on the basis of Einstein's hypothesis, that the 'mass defect' was equivalent to a substantial amount of energy which binds the particles within the atomic nucleus.

The wave mechanics and quantum mechanics theories
evolved by Schroedinger, Heisenbury and Bohr showed that the energy levels of alpha particles emitted by heavy radionuclides could be sufficient to overcome the electrostatic repulsion exerted by the nuclei of lighter elements, enabling them to enter the nucleus itself. However, the rates of emission from ordinary radioactive sources were insufficient to obtain a high probability of the projectiles hitting their nuclear targets. It was concluded that a machine capable of accelerating particles through a few hundred thousand volts could enable a reasonable fraction to penetrate the electric potential barrier.

This led Cockcroft and Walton, in 1932, to develop an experimental 500,000 volt proton accelerator. The apparatus was used to bombard lithium foil with high-energy protons; disrupting the lithium nuclei, and so splitting the atom for the first time. The result of the reaction was the production of two helium nuclei (alpha particles) per collision, according to the equation:

\[ ^7\text{Li}_4 + ^1\text{H}_0 \rightarrow ^4\text{He}_2 + ^4\text{He}_2 \]  

They were able to measure the energies and masses of all the particles involved, and thus (by reference to the mass defect), to verify the Einstein mass-energy equation, \( e=mc^2 \).

Of equal significance, however, was Chadwick's discovery of the neutron in the same year. It had previously been shown that alpha bombardment of certain light elements produced a radiation more penetrating than any other
then known. Chadwick established that this radiation comprised a stream of particles of mass slightly greater than that of a proton, but with no net electrical charge. He named the particle the neutron, and suggested that it comprised a proton and an electron in close combination, with a binding energy of from 1 to 2 MeV. Furthermore, he proposed that the neutron was a basic constituent of all atomic nuclei.

Chadwick's discovery of the neutron more or less completed the modern scientific picture of atomic structure, as previously described. It also promoted further scientific advances which were to lead to the present large-scale exploitation of nuclear energy.

In 1936, Niels Bohr put forward a theoretical model in which a nucleus is considered as a combination of nucleons (protons and neutrons) held together by a finite amount of binding energy. He showed that disintegration of a nucleus would be accompanied by a release of a proportion of this binding energy $\Delta E$ according to the relationship:

$$\Delta E = (M_0 - \Sigma M_i) c^2 \quad \text{.......................... (10)}$$

where $(M_0 - \Sigma M_i)$ corresponds to the mass defect when the compound nucleus of total mass $M_0$ is disintegrated into constituent particles of mass $\Sigma M_i$.

The average binding energy per nucleon is obtained by dividing the total binding energy of the nuclide by its mass number;

i.e. average binding energy per nucleon $= \frac{\Delta E}{A} \quad \text{...... (11)}$
Hence the amount of energy available due to the disintegration of a nucleus varies with the mass number (the total number of protons and neutrons).

The form of the relationship between average binding energy per nucleon and mass number is shown in figure 5. The curve rises to a maximum for mass numbers of about 60, corresponding to elements such as iron, cobalt, nickel and copper; indicating that the nuclei of these elements are more stable than those of either the very light elements, such as hydrogen, or the very heavy elements, such as uranium. Since all physical systems release energy as they move to a more stable state, the curve indicates that there are two ways in which binding energy might be released from a nucleus.

Bohr’s theory indicated that the fusion of light elements to form heavier nuclei would produce a significant net energy gain provided that sufficient energy could be put into the system. At the other end of the scale, the phenomenon of radioactivity had already indicated that certain heavy radionuclides were sufficiently unstable to make a spontaneous (though gradual) release of energy, and it became evident that the addition of a relatively small amount of energy by nuclear bombardment could bring about a more dramatic disruption, accompanied by a significant net energy release. This latter route was seen to be more readily attainable than by the fusion process; since it was recognised that the fusion of light element plasmas would require temperatures of several millions of degrees.

Bohr’s theory envisaged that a particle impinging upon the nucleus of an atom would cause a temporary state of excitation and that disruption would occur if, as a
result of induced random motions, sufficient energy became concentrated upon a single nucleon to enable it to overcome its binding energy within the nucleus.

Such a model is analogous to the behaviour of molecules held in a background state of Brownian motion within a drop of liquid; whereby a definite quantity of additional energy is required to break down the surface tension forces which bind the drop together and bring about a division into two or more droplets. Accordingly, Bohr's description of such nuclear reactions was termed the 'liquid drop' model.

With Chadwick's discovery of the neutron it was realised that neutron bombardment could achieve better penetration of the electrical energy barrier around atomic nuclei (because of its electrical neutrality) than had hitherto been possible.

Enrico Fermi therefore systematically bombarded over sixty elements with neutrons, and induced radioactivity in all but eighteen. He also discovered that the effects of neutron bombardment were improved if the neutron source was surrounded by paraffin wax or some other material containing hydrogen. Although he was unable to explain this phenomenon, Fermi realised that in successive collisions with hydrogen nuclei (which have virtually the same mass as neutrons), neutrons lose their excess kinetic energy and are thereby slowed down to ambient or 'thermal' velocities.

In 1934 he showed that uranium was activated by slow neutron bombardment, and that the isotope $^{238}\text{U}$ absorbed a neutron to form a new isotope $^{239}\text{U}$. Subsequently that later emitted a $\beta$-particle, so gaining a unit of positive charge and forming an isotope of an element.
with atomic number 93 (Np). This in turn emitted a $\beta$ particle to form an isotope of an element of atomic number 94 (Pu), thus:

$$^{239}\text{U} \xrightarrow{\beta} ^{239}\text{Np} \xrightarrow{\beta} ^{239}\text{Pu}$$

The two new elements so formed were Neptunium ($^{239}\text{Np}$) and Plutonium ($^{239}\text{Pu}$), see figure 3.

Between 1937 and 1939, Hahn and Straussman undertook similar experiments and showed that in addition to the production of new 'transuranic' elements, the neutron bombardment of uranium produced radioactive isotopes of barium and other elements in the middle of the periodic table.

The results of experiment and theory were then successfully brought together by Meitner and Frisch, who evolved an explanation of these phenomena in terms of Bohr's liquid-drop model (140). The theory suggested that the increase in kinetic energy brought about by the arrival of a neutron was sufficient to overcome the internal binding energy of a uranium nucleus. This caused the atom to split into two radionuclides; each having approximately half the mass and half the charge (atomic number) of the uranium atom.

The phenomenon was new, since previously observed nuclear disintegrations had only involved the emission of $\alpha$-particles. The particles emitted by the disintegration of the uranium nucleus were relatively massive and, to distinguish the discovery, Frisch termed it fission. He measured the energies and masses involved, and calculated that the two fast-moving fission products had a combined kinetic energy of about 200 MeV.
The significance of the latter was immediately apparent to the international scientific community. The energy produced during the fission process was seen to be about thirty times that associated with the release of an alpha particle and more than a million times the chemical combustion energy of a carbon atom. Within a few weeks of the publication of Meitner and Frisch's paper in, January 1939, their experiments were being repeated all over Europe and America (79).

In France, von Halban, Joliot and Kawarski soon showed that the fission of uranium produced some fast-moving neutrons in addition to the two large fission fragments (an average number of 2.5 free neutrons per fission is accepted today). From a practical viewpoint this discovery was of immense significance because it implied the possibility of a chain fission reaction. Moreover, further investigation by Fermi and others showed that although a fraction of the neutrons were emitted spontaneously (prompt neutrons), some were delayed (delayed neutrons).

Meanwhile, Bohr made further theoretical advances. He showed that the fission process is governed by two factors; namely, the density and energy spectra of the neutron radiation and the binding energies of the target nuclides. He deduced that the probability of neutron absorption to form a nuclide of higher mass number or, alternatively, fission into two nuclides of approximately equal proportions, depends on the kinetic energy (hence wave form) of the neutron radiation and the average binding energy per nucleon within the target nuclide.

He showed, in accordance with his 'liquid drop' model,
that when a neutron impinges upon a nucleus it initially produces an 'excited' compound nucleus, in which the excitation energy is equal to the neutron binding energy in the compound nucleus (expressed as a mass defect) plus the kinetic energy of the neutron prior to capture. Whether the compound nucleus will then fission depends upon whether the energy required to break it apart is greater or less than the binding energy of the neutron in the excited compound nucleus. If it is lower than the neutron binding energy, the nucleus can be fissioned by a 'slow' neutron, i.e. one with negligible kinetic energy. Such a nucleus is said to be fissile, and fission will occur at all neutron energy levels. However, other nuclei may only fission when the impinging neutron is fast-moving; i.e. has the ability to impart sufficient additional kinetic energy to overcome the nuclear binding energy.

The probability of either type of reaction (fission or neutron capture) may be expressed in terms of 'nuclear cross-sections', representing the target area of the nucleus in relation to the impinging neutron. This parameter varies according to the energy of the neutron and, due to the wave nature of matter, is expected to vary with \( \frac{1}{v} \); \( v \) being the neutron velocity. Neutron cross-sections are therefore much greater for slow-moving neutrons.

Fast neutrons, although more likely to cause fission in the event of a collision, cause a reduction in nuclear cross-section; and hence the probability of occurrence is relatively low. To increase the likelihood of reactions with fast neutrons it is necessary to increase the amount of fissile material present so as to increase the overall neutron density.
These scientific principles were expounded in Bohr's classical paper entitled 'The Mechanism of Nuclear Fission' in September 1939 (26). His reasoning showed that the fission of uranium by slow neutrons was attributable only to the isotope $^{235}\text{U}$, whilst $^{238}\text{U}$ could only absorb slow neutrons to produce $^{239}\text{Pu}$. Furthermore, he showed that the fission products were responsible for the delayed neutron emissions which had been observed by Fermi.

Thus, it was perceived that a self-sustaining fission chain reaction would be governed by probabilities which in turn would depend upon various ambient operating conditions. It became clear that when a neutron collides with a uranium nucleus, it may collide with $^{238}\text{U}$ to form $^{239}\text{Pu}$ or with $^{235}\text{U}$ to cause a fission. A proportion of free neutrons may also escape without undergoing any collisions, and unless the rate of loss of neutrons is less than the rate of collisions which produce fission, the reaction cannot be self-sustaining.

Bohr's theories had also supported the notion that a 'critical mass' of uranium was required to sustain a nuclear reaction and that this mass would depend upon the purity of the uranium and the concentration of the fissile isotope $^{235}\text{U}$. Natural uranium contains only some 0.7% of $^{235}\text{U}$, the remaining 99.3% comprising $^{238}\text{U}$.

Rudolf Pierls extended these theories to show that the neutron emission intensity produced by natural uranium was insufficient to sustain a fission chain reaction, however large the mass, based on fast neutrons. However, a critical mass could possibly be achieved with slow neutrons, due to the associated increase in nuclear
cross-section. Pierls suggested that a chain reaction might therefore be achieved by using a hydrogen-based 'moderator' to slow down the neutrons, as in Fermi's original experiments.

Thus in 1939, after a rapid series of remarkable discoveries and shortly before the outbreak of war, the Paris group of scientists von Halban, Joliot, Kowarski and Perrin experimented with a system which used ordinary water as a moderator. The reaction was not self-sustaining, but continued as long as some neutrons were supplied from an external source. They concluded that heavy water (D₂O), having a lower neutron capture cross-section, could be more successful and also worked out the idea of using neutron-absorbing elements such as cadmium or boron to alter the rate of multiplication of neutrons so as to produce a controllable nuclear fission reactor.
3. NUCLEAR POWER PRODUCTION

3.1 First Applications

The outbreak of war in late 1939 did not lessen the pace of atomic research based on nuclear fission, but actually increased it. As described in Chapter 2.4 above, it had been established that the exploitation of nuclear energy was theoretically possible; either in a controlled nuclear reactor or in a nuclear bomb of potentially enormous destructive power. However, it was clear that the application of the pre-war scientific discoveries would require more sophisticated experiments and the commitment of considerable funds in order to achieve practical results on a large scale.

The spirit of open international scientific co-operation up to and including 1939 had been remarkable; and it would be interesting to speculate upon possible developments in a more peaceful era. However, with the declaration of war, the freedom of scientific exchange was suddenly brought to an end. In the war-time atmosphere, there was no doubt that efforts to develop a nuclear bomb should receive priority; and the development of controlled nuclear reactors would be either secondary or incidental to that objective.

Fortunately for the western allies, many of the eminent scientists of mainland Europe took refuge in Britain. A significant bonus was a package containing some 185 Kg of heavy water (D$_2$O); brought across the Channel from France by von Halban and Kowarski. This represented the total world stock at that time, which
had been previously purchased from Norway under a secret agreement with the French government. (78)

It is inappropriate here to recount a detailed history of the war-time 'Tube Alloys' and 'Manhattan' projects which led to the production of the first self-sustaining thermal nuclear reactor and to the testing and use of the nuclear bomb. An account of these developments and the subsequent establishment of a fully-fledged civilian nuclear industry has been produced by Professor Margaret Gowing, in her official histories, researched from the archives of the United Kingdom Atomic Energy Authority (79, 80).

However, it was the war-time research efforts of Britain, America, France and Canada which established the present international pattern of industrial developments in reactor design and fuel processing. It is therefore instructive to outline the historical background to show how this pattern was brought about.

The war-time dilemma consisted of deciding upon the quickest route to the manufacture of a nuclear bomb. It was known that $^{235}$U produced a significant number of spontaneous neutron emissions, and would form a critical mass if obtained in sufficient quantity and at a sufficiently high-level of purity. Thus, two sub-critical hemispheres of high-grade $^{235}$U, slammed together with sufficient rapidity, would create a nuclear explosion, based on the rapid multiplication of fast neutron fission reactions. However, the low concentration of $^{235}$U in natural uranium meant that some form of artificial treatment was required, involving an increase in the proportion of fissile $^{235}$U to form 'weapons-grade' uranium (approx. 90% $^{235}$U).
The possibility of an alternative route became apparent with the discovery that the new isotope $^{239}\text{Pu}$ was also fissile. Thus, if a controllable thermal nuclear reactor could be built and operated, using natural uranium fuel, it could create fissile $^{239}\text{Pu}$ through slow neutron capture by the fertile uranium isotope $^{238}\text{U}$. Isolation of the $^{239}\text{Pu}$ could then be achieved by chemical separation much more easily than the physical separation of $^{235}\text{U}$ from an intimate mixture of natural uranium isotopes.

It was perceived that both possibilities had to be pursued simultaneously to mitigate against the chances of total failure, and rapid advances were made in Britain by European Scientists under the direction of Chadwick. However, Britain's vulnerability to air-attack, combined with a shortage of uncommitted funds and increased difficulties in securing sufficient supplies of uranium, meant that there was little chance of short-term success. Thus, with America's entry into the war, an Anglo-American agreement was reached whereby the majority of the development work was transferred to America, based at the Los Alamos and Hanford research centres and at Fermi's laboratories at the University of Chicago.

The Americans were given full access to all the new scientific knowledge and expertise gained in Britain, and their vastly superior financial and engineering resources allowed rapid advances thereafter. Research was directed towards the development of experimental uranium enrichment plants for the separation of $^{235}\text{U}$ by gaseous diffusion, electro-magnetic separation and high-speed centrifuge methods; using refined uranium in the form of uranium hexafluoride ($\text{UF}_6$) gas. In
addition, natural uranium graphite-moderated nuclear reactors were developed for the production and subsequent chemical isolation of \(^{239}\)Pu; the first graphite reactor (Fermi's 'pile') achieving 'criticality' in Chicago in 1942.

Stringent conditions were imposed on the British-based European scientists, who were given restricted access to these new developments. This was due to 'compartmentalisation'; a security policy which prevented the free transfer of information between the various separate scientific groups involved in the overall war-time project. However, due to pressure by the British group, a separate Anglo-Canadian research centre was eventually established in Montreal and was subsequently transferred to newly established facilities at Chalk River, in Ontario, Canada.

Canada's involvement was largely fortuitous, in that she was a neutral country with readily available supplies of uranium (the Eldorado Mine). Canada's principal contribution was the supply of sufficient uranium to meet the research and development needs of the American effort. However, small amounts of uranium were eventually made available for the independent Anglo-Canadian venture at Chalk River, specifically for the development of a heavy water-moderated thermal reactor. Cockcroft assumed directorship of the project, with Von Halban and Kowarski as senior members of the team.

By the end of the war, success had been achieved in all areas; although the knowledge and expertise which had been gained was fragmented as a result of the compartmentalisation policy; the Americans having the clear industrial advantage. The nuclear bombs
exploded on Hiroshima and Nagasaki were actually based on enriched $^{235}$U and $^{239}$Pu respectively. The fact that the Americans were able to use both materials demonstrated that they had successfully established the following:

- an industrial scale process for the enrichment of $^{235}$U
- graphite-moderated, thermo-nuclear reactors operating with natural uranium fuel for the production of $^{239}$Pu
- techniques and facilities for the chemical treatment of irradiated fuel and the extraction of plutonium.

The Anglo-Canadian team at Chalk River had also successfully developed a heavy-water production plant and had operated two heavy water reactors. The first was 'Zeep'; a zero-energy experimental reactor, and the second was a relatively high-powered reactor producing approximately 4MW of heat per tonne of natural uranium fuel. The Canadians were therefore left with the beginnings of their own nuclear power-producing capability; based on heavy water-moderated natural uranium reactors (80).

In 1946, the British half of the Anglo-Canadian team returned to Britain to establish the Harwell Research Establishment. Britain had no industrial nuclear 'hardware' of her own, but her scientists had accumulated valuable technical knowledge, based on their participation in the combined war-time research efforts (49). In addition, Britain had become a member of the Anglo-American Development Trust which
gave Britain and America a near monopoly of the western world's uranium production for several years (79).

The French scientists (von Halban, Kowarski and others) were in a less favourable position, despite the valuable war-time contributions they had made. The British membership of the Anglo-American Trust precluded the sharing of classified atomic information with scientists from other countries, beyond those particular aspects with which the individual scientists had been directly involved (78).

This overall state of affairs in the immediate post-war period set the scene for the subsequent development of nuclear power industries in America, Canada, Britain and France; see Section 3.2 below. The American preference for light-water reactors, using enriched uranium fuel; Canada's reliance on heavy water reactors based on natural uranium; Britain's independent development of Magnox and Advanced Gas-cooled graphite-moderated natural uranium reactors; and France's efforts initially along similar lines to the British, can all be traced to the history of nuclear research and development during the Second World War, and the status of knowledge, expertise and engineering resources which each country had attained.

3.2 Reactor Types and Operating Features

The design and operating details of a fission reactor depend fundamentally upon the choice of materials for the following components:
The choice of materials for any one of the above imposes limitations on the remainder. As a result, reactor designs tend to fall into relatively few categories (3, 163).

The types of moderator which can be used include ordinary water (light-water), heavy water, (D\textsubscript{2}O), graphite and beryllium. As described in Chapter 2.4, the function of the moderator is to slow down the fast neutrons which occur as a result of the fission process, so as to increase the fission cross-section of 235 U. Hydrogen-rich materials are the best moderators (since the mass of a hydrogen nucleus is virtually identical to that of a neutron), and ordinary water forms the most readily available source.

However, ordinary water has a significant neutron capture cross-section, which prevents a sustained slow neutron fission chain reaction in natural uranium fuel. Therefore, light-water moderated reactors require 235 U enrichment to about 3\% in order to increase the neutron density to a level which compensates for their relatively high neutron absorption properties.

By comparison, heavy water is a more efficient moderator, since it has a relatively low neutron capture cross-section. Heavy water reactors are therefore able to sustain fission reactions in ordinary uranium fuel. However, heavy water is
difficult and expensive to manufacture. Ordinary water contains hydrogen as a mixture of hydrogen ($^1\text{H}_2\text{O}$) and deuterium ($^2\text{H}_2\text{O}$) isotopes. The manufacture of heavy water requires an enrichment of the proportion of deuterium by an industrial-scale process similar to that used for the production of enriched uranium. The use of heavy water moderated reactors therefore implies relatively low fuel processing costs, but high moderator costs. It should be noted, however, that the moderator is not consumed during reactor operation.

Graphite is a comparatively inefficient moderator, since its mass number ($A=12$) is relatively high. However, it has the significant advantages of being readily available and a solid. The latter allows it to fulfil a secondary function as part of the structure of the reactor core. In addition, due to the low neutron absorption of graphite, graphite-moderated reactors can operate with natural (un-enriched) uranium fuel. Beryllium has similar properties, but is less readily available than graphite. Beryllium-moderated reactors therefore tend to be used only for research purposes.

These rudimentary aspects of reactor design, together with the post-war circumstances outlined in 3.1 above, more or less determined the types of thermal reactors which were initially developed by America, Britain, Canada and France; and subsequently exported to the rest of the western world.

America had the greatest range of choices; having successfully operated heavy water and
graphite-moderated, plutonium-producing reactors at the Hanford reservation and Chicago University respectively. In addition, large-scale gaseous diffusion facilities had been established at Oak Ridge Laboratories in Tennessee. However, America's immediate post-war pre-occupation was with the further development of its defences. It was realised that a nuclear-powered submarine could remain submerged for very long periods by comparison with fossil-fueled submarines, since frequent re-surfacing for oxygen intake would be unnecessary. Accordingly, America concentrated its post-war research effort on the development of compact, light water-moderated reactors using enriched uranium fuel for use in nuclear-powered submarines. The compactness was achieved by using pressurised water as a combined moderator/coolant and by using relatively small volumes of enriched fuel to achieve sufficient increase in power density. These early submarine reactor designs formed the basis for all subsequent pressurised light water reactor systems (P.W.R.'s and B.W.R.'s)* developed by America and sold to many other countries (3).

Canada's reliance on heavy water reactors is directly attributable to the Anglo-Canadian war-time research programme at Chalk River. Heavy water production facilities were available, whereas uranium enrichment facilities were not. Abundant supplies of natural uranium ore were also at hand. A family of natural uranium-fuelled heavy water reactors has since been developed successfully and marketed abroad. These are referred to as 'CANDU' reactors (Canada-Deuterium-Uranium).

* PWR = Pressurised Water Reactor
BWR = Boiling Water Reactor
During the immediate post-war period, Britain had no facilities either for uranium enrichment or for heavy water production. From this position in 1946, the only possible choice for an industrial reactor design was a graphite-moderated reactor, using natural uranium fuel. France found herself in a similar position; although, as previously noted, the French atomic scientists who were to become founder members of the Commissariat a l'Energie Atomique (C.E.A.) had very restricted access both to classified research information and to the then available supplies of uranium (78).

Thus, graphite-moderated, gas-cooled natural uranium reactor designs formed the basis for the first generation of industrial thermal reactors developed in Britain and France (49; 78). Britain's series of Magnox and Advanced Gas-cooled reactors, commencing with Calder Hall in 1956, have continued to form the basis of its nuclear power industry. However, France changed to American light-water reactor systems at a relatively early stage. Britain now appears to be contemplating the same course, with the planned introduction of the Sizewell-B PWR during the 1990's.

The essential features of the various types of reactor designs are shown in table 1, and are also illustrated in figures 6, 7 and 8. The factors which influence the choice of moderator and level of fuel enrichment have already been described. These in turn influence the choice of coolant. For water-moderated reactors, water also forms the obvious choice for the coolant. However, this requires special corrosion-resistant fuel claddings; and an alloy of zirconium and aluminium ('zircaloy') is normally used. For
graphite-moderated reactors, it is normal to use carbon dioxide as a coolant, but for the High-Temperature Gas Cooled Reactor (HTGR) helium must be used to avoid problems associated with the corrosion of magnesium alloy (Magnox) by carbon dioxide at high-temperatures.

Further description of the design details of these various types of thermal reactor is beyond the scope of this thesis, although detailed accounts are readily available elsewhere (3, 107, 163). However, it is appropriate to outline the various reactor operation characteristics which influence the quantities and composition of the radioactive wastes which they produce.

The principal elements of a typical thermal reactor are shown in figure 6. The reactor is fuelled with a carefully designed geometrical pattern of fuel rods which, in combination, represent much more than a 'critical' mass of fissile material. The neutron-absorbing control rods (usually made of cadmium, hafnium or boron) are interspersed with the fuel rods and, before start-up, are fully inserted within the core. Gradual withdrawal in a symmetrical arrangement then results in an increase in neutron density until 'criticality' is reached.

Loss of free neutrons produced by the fission process can often be minimised by providing moderator materials beyond the region of the fuel elements to reflect neutrons back towards the core. However, the whole of the core must also be surrounded by concrete or other dense material to prevent neutron or \( \gamma \) -radiation from causing biological damage. The \( \alpha \) and \( \beta \) -radiation is normally contained within the fuel.
rods, since it is unable to penetrate the fuel cladding materials.

The reactions which take place within the reactor core are highly complex. As the fission chain reaction proceeds, a build-up of new radionuclides occurs within the fuel. A great variety of fission products is created, some of which have high neutron capture cross-sections, e.g. $^{135}$Xe. In addition, some neutrons are lost to newly-created transuranic elements. For example, a proportion of the $^{238}$U is converted to $^{239}$Pu and, although a proportion of the latter contributes to the fission process, some of the $^{239}$Pu captures additional neutrons to form $^{240}$Pu, $^{241}$Pu, and other heavier actinide elements; see figure 3.

Thus, during reactor operation, the multiplication of fission reactions varies and the overall fuel composition becomes increasingly complex. Constant regulation is therefore required and, in general, the reactivity must be gradually increased by the withdrawal of control rods. It is for this reason that thermal reactors must be designed to contain much more fuel than is required for their initial operation.

With prolonged irradiation of the fuel, the production of gaseous fission products such as krypton, xenon and iodine tends to exert a pressure which distorts the fuel claddings. The intense neutron radiation can also effect the crystal structure of the cladding materials; inducing high stresses which could cause leakage of fission products into the moderator and/or coolant. At some stage, therefore, it is necessary to withdraw the fuel rods and to re-fuel the reactor; the
maximum period for safe and economic fuel irradiation being about 3 to 5 years for most thermal reactor designs.

Due to the differences in reactor fuel composition and operating characteristics, as outlined in table 1, the composition of 'spent' fuel varies from one reactor to another. However the typical composition of spent fuel from a PWR after 3 years continual irradiation is as shown in table 2 and serves to illustrate the changes brought about by the fission process (50).

It is apparent that the proportion of the original uranium actually consumed is very small. In fact, less than 1% of the original $^{238}$U actually contributes to the fission process (after conversion to $^{239}$Pu) and only 4.5% of the total amount of uranium is transmuted by fission or neutron capture processes. As will be shown in 3.3 below, this has given rise to the development of other industrial processes for the extraction and further utilisation of the 'un-burned' fissile and fertile isotopes contained in spent thermal reactor fuels.

In thermal reactors, the low-level of utilisation of $^{238}$U is an obvious draw-back. Fast reactors form a separate group of reactors which offer greater potential for the extraction of fission energy, by increasing the 'burn-up' of $^{238}$U. In these reactors, no moderator is present and energy is released by 'fast' neutron fission reactions.

As previously described, fission cross-sections are greater for slow neutron bombardment. The reduction in nuclear cross-sections for fast neutrons means that about 400 times as many neutron emissions are required
to promote a sustained fission reaction (142). Accordingly, a much higher neutron density must be created in a fast reactor.

The latter can only be achieved by using a very high concentration of fissile material. Therefore, fast reactor fuel normally comprises a mixture of plutonium and uranium having an equivalent enrichment of some 25-30%. The core contains no moderator and the minimum of structural materials in order to minimise neutron losses, as shown in figure 8. In consequence, the core of a fast reactor is very compact and typically has a power density some 10-100 times greater than thermal-type reactors; thereby achieving a much greater 'burn-up' per tonne of fuel. In addition, since plutonium fuel releases a relatively large number of fast neutrons per fission, fast reactors can be designed to 'breed' more fissile material than they consume during reactor operation.

Because of the high-power density of fast reactors, water or gas coolants are inadequate, and a liquid metal coolant is required to provide a sufficiently high thermal conductivity. The coolant generally used is liquid sodium. However, the intense neutron radiation in the core induces radioactivity in the sodium to produce $^{24}$Na. For this reason, fast reactors require both a primary and secondary coolant circuit; the primary circuit and the heat exchanger being contained within the biological shield.

The first industrial-scale fast reactor was the British Dounreay Fast Reactor which commenced operation in 1959. Other experimental fast reactors were built at around the same time in America and France. However, a new generation of prototype fast
reactors has been developed since 1972; notably the Dounreay Prototype Fast Reactor and the French Phénix and Super Phénix reactors. These new generation fast reactors are likely to play an increasingly significant role in terms of international fuel cycle strategies, as described in 3.3 below.

3.3 Nuclear Fuel Cycles

'The nuclear fuel cycle' is a phrase which is open to mis-interpretation. In fact, in the absence of effective high-activity radioactive waste disposal procedures, there are few aspects of present day nuclear power production which can be described as cyclic. Figure 9 summarises the range of possible nuclear fuel 'cycles' and indicates the various stages at which intermediate-level and high-level wastes are produced.

Nuclear reactor fuel originates in the ground as natural uranium ore. High grade ores can contain up to 4% uranium; although lower grade ores, with less than 1% uranium content, are now becoming economically attractive (143). Processing the ore by crushing and chemical solvent processes results in the production of a refined uranium oxide ($\text{U}_3\text{O}_8$), known as 'Yellow Cake' containing some 85% of uranium by weight. It should be noted that these processes produce considerable quantities of radioactive waste in the form of mine tailings and liquid effluents. However, the problems associated with these types of wastes are not considered in this thesis.

The next stage of processing involves uranium enrichment; although, as shown in 3.2 above,
enrichment is not necessary for heavy water or graphite moderated thermal reactors. For many years, America had a monopoly on the uranium enrichment process; inherited from the war-time situation. However, since the ALEM treaty of 1968, two major European groups, URENCO, and EURODIF have established large-scale uranium enrichment industries in Europe, based on centrifuge and gaseous diffusion processes respectively (58, 136A). New processes are also being researched in Germany and America, based on jet deflector principles and ionisation by laser beams (18).

The enrichment process results in the production of fissile material of the required $^{235}$U content, but also creates a large residue of depleted uranium; i.e. uranium composed almost entirely of $^{238}$U. All countries which operate thermal reactors which utilise enriched uranium fuel are therefore creating stocks of $^{238}$U which cannot be consumed in conventional fission reactors.

The fuel assemblies required for reactor operation are, in the case of Magnox reactors, manufactured directly from natural uranium billets which are melted, alloyed and cast into fuel rods. Most other reactor types (as shown in table 1) use uranium oxide fuel in the form of ceramic pellets. These are compressed from a dry powder, loaded into helium-filled cladding tubes, and sealed to form reactor fuel assemblies. The processes involved in this type of fuel fabrication also create some waste, mainly in the form of failed equipment, contaminated protective clothing etc.

The next stage involves the irradiation of the fuel in a thermal reactor. As described in 3.2 above, the
'spent' fuel which is eventually discharged is actually very rich in uranium. Like the residue from the enrichment process, it is depleted in $^{235}\text{U}$. However, it contains a significant quantity of fissile $^{239}\text{Pu}$. In addition to these 'resource' materials, it contains high-activity wastes in the form of actinide elements and fission products.

In this form, the spent fuel represents an intimate mixture of materials which are of no further industrial value and, in the absence of further treatment, would have to be regarded as waste. However, 'reprocessing' provides a means of recovering and re-using the uranium and plutonium. This becomes feasible after the spent fuel rods have been stored in cooling ponds for a period of 2-3 years to allow the most active fission products (e.g. $^{133}\text{Xe}$ and $^{131}\text{I}$) to decay to very low concentration levels.

Reprocessing is achieved by dissolving the spent fuel in concentrated nitric acid and subsequently separating the uranium and plutonium by chemical solvent extraction methods. Uranium recovered from enriched uranium reactors contains a significant amount of $^{235}\text{U}$, which makes re-cycling to the enrichment and fuel fabrication plant worthwhile. However, the uranium recovered from spent fuel discharged from natural uranium reactors is more severely depleted in $^{235}\text{U}$ and enrichment is generally regarded as uneconomic. The depleted uranium is therefore stockpiled and only the $^{239}\text{Pu}$ is re-utilised. The highly radioactive liquid waste stream which remains after the extraction process has no resource potential and can therefore only be regarded as waste (3).
Reprocessing, however, involves expensive and sophisticated technology and, although it has become recognised as a commercially viable international industry, industrial-scale plants are necessarily large and few in number. At present, the principal centres are BNFL’s reprocessing facilities in Sellafield (United Kingdom) and the French facilities at Marcoule, operated by COGEMA. Another centre at Mol, in Belgium, formerly operated by the European consortium EUROCHEMIE, may also be re-opened under the auspices of the Belgian authorities in the near future (208). In these centres, reprocessing is carried out on a commercial basis, involving the shipment of spent fuel from other countries and the export of advanced reactor fuel assemblies.

Thus far, the various industrial processes described have involved only thermal reactors. As noted in 3.2 above, fast reactors require a considerably higher proportion of fissile material to achieve the high power density necessary for their operation. This could be achieved by using highly enriched uranium from an enrichment plant. However, this fuel manufacturing route is not economically viable and fast reactor fuel is normally made by mixing $^{239}$Pu obtained from fuel reprocessing together with depleted uranium (comprising mostly $^{239}$U), to obtain an 'equivalent fuel enrichment.'

During fast reactor operation, the high neutron density results in a high conversion ratio* for the

* The conversion ratio is the ratio of the quantity of fissile material created by neutron capture in fertile isotopes to the quantity of fissile material consumed by fission reactions. For thermal reactors the conversion ratio is always less than one.
production of $^{239}$Pu. By surrounding the core of the reactor with a 'blanket' of depleted uranium, it is possible to produce a conversion ratio greater than one; i.e. to create more $^{239}$Pu by neutron capture than is consumed by fission within the reactor core. In this mode of operation, the reactor is termed a 'fast breeder'.

The various reactor types and industrial processes described above give rise to a variety of possible 'nuclear fuel cycles' with a wide range of overall fuel economies. It appears that the greatest industrial attraction of the near future may consists of a mixed contingent of thermal reactors and a relatively small number of fast breeder reactors. After initial start up, the latter could then breed sufficient fissile material in the form of $^{239}$Pu to supply mixed uranium/plutonium fuel of the required grades for both reactor types. Thus, at a subsequent stage, conventional thermal reactors could theoretically be phased out altogether.

Clearly a reprocessing industry is essential for such a scenario, although the need for uranium enrichment plants would gradually disappear. Thus, the 'fuel cycle' would effectively be limited to the sequence illustrated on the right-hand side of figure 9. The uranium efficiency could then probably increase to a value in excess of 60% as opposed to the 2-3% consumption levels based on thermal reactor operation (3); and even higher efficiency levels are possible in theory (34).

However, questions of nuclear proliferation enter into the arguments. The reprocessing industry, which is
essential to advanced developments of this type, clearly provides opportunity for access to high-grade plutonium. In 1977, this led President Carter to announce a ban on reprocessing and fast reactor development in America. He also established an International Fuel Cycle Evaluation Study to undertake a technical evaluation of different fuel cycles, with a view to minimising any associated risks of proliferation of nuclear materials (58).

The Carter ban in America meant that only a 'once-through' thermal reactor fuel cycle was possible, and therefore spent reactor fuel, including the unused uranium and plutonium contained in it, were to be regarded as waste (16). Proposals for the disposal of spent fuel were also developed at around the same time by Canada and Sweden (95, 123).

Against the apparent moral stance of the Carter initiative of 1977, it is significant that America and Canada together possess over half the world's known reserves of uranium. The dependence of most other nuclear power-producing countries on imported uranium suggests that they would be most unwilling to adopt a 'throw-away' option. It is therefore not surprising that almost all European countries have made a firm commitment to reprocessing and the development of fast reactor fuel cycles (165). Within America itself, the 'Carter' reprocessing ban caused considerable controversy amongst industrialists and economists; particularly in relation to the loss of revenue and concern over American industrial reliability with respect to the international reprocessing trade. Significantly, soon after President Reagan's administration began, the American reprocessing ban was lifted (216).
In the author's view, a number of important factors mitigate against the once-through fuel cycle and the disposal of spent fuel. Firstly, spent fuel always constitutes a potential energy resource wherever it is; and therefore future generations may seek to retrieve it. Secondly, the use of fast reactors (which depend on reprocessing for fuel supply) represent a very satisfactory method of utilising accumulated stocks of depleted uranium and fissile plutonium 'waste' in a constructive way (136A). Thirdly, $^{235}$U enrichment technology (the need for which could be largely eliminated by advanced reactor fuel cycles) represents a much easier and increasingly inexpensive route to obtaining supplies of weapons-grade fissile material (107).

The above factors indicate a fundamental illogicality in current arguments concerning the proliferation risks associated with reprocessing. Accordingly, both on pragmatic and logical grounds, it will be assumed in this thesis that nuclear power production will include fuel reprocessing and fast reactor development, and the disposal of unreprocessed spent fuel will not be considered in any significant detail.
4. THE COMPOSITION AND CHARACTERISTICS OF HIGH-ACTIVITY RADIOACTIVE WASTES

4.1 Introduction

In Part 4 of this thesis it will be shown that it is essential to obtain an understanding of the characteristics of high-activity radioactive wastes in order to assess the most satisfactory methods for their disposal (8, 34). More specifically, it is important to explore the opportunities which exist for exercising engineering control over the form and condition of the wastes before they are permanently isolated from the human environment.

In Chapter 3, it has been shown that radioactive wastes occur as direct end-products of each of the main processes involved in nuclear power production; namely, enrichment, fuel fabrication, irradiation and fuel re-processing. Furthermore, according to the overall nuclear power production strategy adopted, 'primary' waste from one cycle of energy production can be utilised to manufacture fuel for another. However, at each stage 'secondary' (intermediate-level and low-level) waste streams are also created, due to the contamination of processing materials, handling equipment, protective clothing, etc. For this reason, the whole spectrum of radioactive wastes is extremely heterogeneous and varies in overall composition according to the reactor types and fuel cycle adopted, and the efficiency of operating methods; see figure 9.

Before describing the characteristics of high-activity radioactive wastes, it is important to review the radiological characteristics of three categories of
radionuclides (147). These comprise:

- transuranic elements
- fission products
- neutron-activation products

The transuranic elements, such as neptunium, plutonium and americium form part of the actinide series of elements identified in figure 3. They are formed as a result of neutron capture during the irradiation of nuclear fuel, and they decay primarily through $\alpha$-emission. Their characteristic half-lives are very long; ranging from 433 years for $^{241}$Am to $2.4 \times 10^6$ years for $^{237}$Np; and long sequences of daughter radionuclides are produced as a result of their decay chains, the latter gradually reaching a state of equilibrium with their precursors, as described in Chapter 2.3 (99, 100, 39).

The fission products include a wide variety of intensely active radionuclides, based on elements of medium atomic weight. The vast majority of these decay by emission of $\beta$ and $\gamma$-radiation and have relatively short half-lives; typically less than one year. However, a significant number have half-lives in the range 30 to 105 years and a few notable exceptions are extremely long-lived; e.g. $^{99}$Te with a half-life of $2.1 \times 10^5$ years and $^{129}$I with a half-life of $1.7 \times 10^7$ years.

The neutron-activation products are unavoidable by-products of nuclear fission. They are generated in the structural materials of the reactor, the coolant, the moderator and the fuel claddings, by neutron capture in stable isotopes. They consist of 'excited' compound nuclides which decay through $\beta$ and
γ-radiation; their half-lives being generally shorter than those of the fission products. A typical example is $^{60}$Co, produced by the neutron activation of steel components. Storage of such wastes for 10 years generally reduces their activity by about a half.

The fission products and the neutron-activation products share a common feature. In both cases, their radioactivity levels are initially very high; due to their intense excitation energy. In consequence, they emit significant quantities of heat as their radiation energy dissipates to their surroundings.

All radioactive wastes comprise a combination of the three types of radionuclides described above, although one or other type may dominate. However, due to wide variations in overall composition, there can be no rigorous system for classifying the wastes. Although the terms 'high-level', 'intermediate-level' and 'low-level' are widely used, definitions vary from one country to another. The terms usually denote different levels of radiological hazard, but confusion often arises in the absence of unified definitions of threshold radioactivity concentration levels, etc.

The problem of classification is made more complicated by the fact that the wastes can occur in gaseous, liquid or solid form. For a given radioactivity concentration, gaseous and liquid wastes clearly pose greater containment hazards, due to their mobility.

The International Atomic Energy Agency (I.A.E.A.) has attempted to standardise the categories of wastes which arise during the operation of nuclear power plants (116), and has defined a number of categories according to their activity concentration, expressed in
Mci/cm\(^3\); (see Chapter 4.4 below for definition of units). In the United Kingdom, Duncan and Brown have published a detailed description of the waste inventory predicted to the year 2000, in which a similar method of classification has been adopted (62).

However, for the purpose of this thesis, it is more convenient to adopt a general classification, based on the familiar terms high-level, intermediate-level and low-level. Furthermore, to avoid undue complexities, descriptions will generally be related to solidified wastes; i.e. wastes which have been conditioned and packaged into a solid form prior to disposal. As a rough and simple guide, Cope and Briscoe give the following upper limits of activity concentration for each of these categories (53):

<table>
<thead>
<tr>
<th>Waste Category</th>
<th>Gross Activity Concentration</th>
</tr>
</thead>
<tbody>
<tr>
<td>Low-Level</td>
<td>(10^2)</td>
</tr>
<tr>
<td>Intermediate-level</td>
<td>(10^4)</td>
</tr>
<tr>
<td>High-level</td>
<td>(10^7)</td>
</tr>
</tbody>
</table>

Low-level wastes will not be considered in detail, since they are generally short-lived, and routine disposal procedures already exist, as described in Chapter 1. They include contaminated materials from various secondary processes. Solid varieties include protective clothing and heterogeneous rubbish with low or negligible activity-concentrations. Where possible, they are reduced in volume, immobilised and packaged (generally in steel drums) for disposal by shallow land-burial or sea-dumping.
Other gaseous and liquid low-level wastes are discharged directly into the atmosphere or to the sea at nuclear sites. A notable example is $^{41}$Ar, which is a neutron-activation product produced in the air on the inside of reactor pressure vessels. By discharging the contaminated air into the atmosphere from a high stack, the activity induced at or near ground level is maintained at a low level, since $^{41}$Ar has a half-life of only 1.8 hours (163).

High-level and intermediate-level wastes represent a far greater radiological hazard. They are the high-activity wastes which arise from the primary and secondary processes of nuclear power production; and as noted in Chapter 1, there are as yet no industrial-scale facilities available for their disposal.

4.2 High-Level Wastes

High-level radioactive wastes comprise the unwanted materials contained within the fuel rods when they are removed from the core of a reactor. The composition of the spent fuel varies according to the period of irradiation, the power density and hence the initial level of fuel enrichment.

Since enrichment levels for most thermal reactors only vary in range 0-3%, the typical composition given by Cohen (50) for spent PWR fuel may be regarded as broadly representative for all types; see Table 2. However, spent fuel from fast reactors would be expected to contain a much higher proportion of plutonium and fission products; the actual percentages depending upon the conversion ratio for which the
reactor was designed.

During reprocessing, the plutonium and uranium are removed by a succession of solvent extraction cycles, using organic solvents, such as tributyl phosphate. The water-based acid solution remains as a liquid raffinate containing nearly all the non-volatile fission products together with the various actinide elements, including traces of the original uranium and plutonium (typically 0.05% to 0.5%) which cannot be economically removed. In addition, small amounts of iron, chromium, nickel, aluminium, zirconium, magnesium and their neutron activation products may be present. These extraneous materials are introduced into the waste stream by the corrosion of processing equipment and small remnant pieces of fuel cladding during the acid dissolution of the fuel (111).

The liquid high-level waste is both thermally hot and intensely radioactive; and therefore requires very careful handling. It is normally led from the solvent extraction process into special stainless steel tanks, equipped with cooling circuits and internal agitators. The permissible concentration of fission products contained in the liquid is generally governed by the heat removal capacity of the cooling system; which is normally designed to maintain temperatures at or below about 50°C (163).

Liquid high-level waste storage tanks of this type exist at all the reprocessing sites throughout the world. It is estimated that those at Sellafield in the U.K. now contain some $1000 \text{m}^3$ of liquid wastes (192A, 163), whilst the Hanford reprocessing establishment in the United States is reported to have some $250,000 \text{m}^3$ of liquid wastes in storage from its
This is the form in which a vast majority of the international back-log of high-level radioactive wastes is stored today. Without continuous cooling, the rate of evaporation would rapidly lead to high fission product concentrations, boiling and ultimately melting of the containers. For this reason, and because the carbon steel and stainless steel tanks in current use have limited design lives, a considerable amount of research has been undertaken to devise satisfactory methods for solidifying and storing the wastes (84, 192A, 112).

The process which has received the greatest attention is vitrification. This involves evaporating, calcining and fusing the wastes together with a glass 'frit' to form an intimate molten borosilicate glass/waste mixture. This molten waste composite may then be cast into thin-walled stainless steel canisters and allowed to cool to form cylindrical-shaped units of solidified high-level waste (84, 192A).

Vitrification is obviously advantageous in reducing the risks associated with the storage of liquid high-level reprocessing wastes. In most countries, it is also considered to be a highly effective form of conditioning for geological disposal; since borosilicate glass has extremely low solubility in groundwater.

However, following vitrification, the fission products within the immobilised wastes continue to emit heat at a significant rate. A typical heat decay curve for a waste-bearing borosilicate glass containing
approximately 15% by weight of fission products is shown in figure 10; based on a mean of PWR and Magnox wastes (7). For a given waste unit, the heat output is greatest at the time of vitrification and decays progressively thereafter, according to a composite decay curve which depends upon the properties and half-lives of the various radionuclides present, and their initial concentration levels. At any instant, the actual heat output depends upon:

- the quantity of glass in each waste unit (unit size);
- the concentration of fission products within the glass matrix (15% by weight being an approximate practical upper limit);
- the period of liquid waste storage prior to vitrification;
- the period of temporary storage/cooling of the solidified waste unit after vitrification.

Clearly, a wide variation is possible in the size, shape, composition and thermal characteristics of the waste units. In addition, high-integrity overpacks may be used to encase the waste units and provide increased shielding and corrosion protection. Materials which have been considered include lead, copper, titanium, cast steel alloys and others (64, 8, 137).

It is appropriate at this stage to note the quantities of high-level waste involved, in volumetric terms. A 0.5GW(e) PWR reactor in continuous operation would produce an equivalent of 25 tonnes of spent fuel per year, which in turn would produce about 100m$^3$ of liquid high-level reprocessing wastes (163). The vitrification of this waste at 15% by weight fission product concentration would produce approximately 2m$^3$ of
solidified reprocessing waste (163). However, it is clear that reducing the fission product concentration in the glass matrix would increase this volume on a pro-rata basis. Thus, adopting a very low fission product concentration of say 1% would result in a total immobilised waste volume of about 30m$^3$.

Most countries which have developed outline conceptual proposals for the geological disposal of high-level wastes have already made (or are about to make) decisions concerning the form and concentration of vitrified high-level waste units; see Chapter 13. Waste vitrification and storage facilities are already operating on an industrial scale at La Hague and Marcoule in France, and a new vitrification plant is due to commence operation at Sellafield in the U.K. during the late 1980's (192A).

It will be shown later that the choices made concerning waste vitrification methods, packaging and storage, can have a profound influence in terms of the heat generation and containment properties of solidified waste units. In the author's view, the design of the waste units should therefore form a carefully integrated part of the design of any geological disposal system, and it is possible that the somewhat arbitrary industrial decisions being made at present may impose undesirable limitations (8, 34). Further consideration of these factors will be presented in Parts 3 and 4.

4.3 Intermediate-Level Wastes

Intermediate-level wastes are indirect or 'secondary' waste products of the nuclear fuel cycle which occur in relatively large volumes (typically in excess of 100
times the volume of high-level wastes). They are clearly distinguishable from high-level wastes because of their secondary occurrence. However, they are less easily defined and more variable in composition.

Their constituent radionuclides include fission products, neutron-activation products and transuranic elements. The principal sources from nuclear power plants may be identified as follows (62, 111):

(a) used air filters;
(b) ion-exchange resins used in the decontamination of reactor coolants;
(c) certain types of incinerator residues;
(d) failed equipment;
(e) certain categories of contaminated rubbish;
(f) irradiated fuel claddings;
(g) ion-exchange resins used in the decontamination of pond water for the temporary storage of spent fuel;
(h) chemical sludges arising from the 'floc' treatment of liquid effluents;
(i) evaporator concentrates;
(j) liquid wastes arising from second and third stage reprocessing of spent fuel;
(k) contaminated or neutron-activated structural materials from the de-commissioning of nuclear reactors, enrichment plants, etc.

The above conveniently fall into three sub-groups. The first sub-group (a to e) may contain a variety of neutron-activation products and significant quantities of transuranic elements, but relatively small quantities of fission products. Thus, although the wastes may require relatively short-term cooling to allow the decay of most of the neutron-activated $\beta$ and $\gamma$-radiation, their heat output is insignificant in the long-term. An
important component is tritium ($^{3}\text{H}$) which has a half-life of 12.3 years and is produced as a result of neutron absorption in water coolants (see b above).

The second sub-group (f to j) generally contains a significant proportion of fission products. Although the activity concentration is generally low, the presence of the latter results in a small but significant but short-lived heat output.

The third category (k) must be distinguished from the remainder, as no major commercial or industrial-scale reactors or enrichment plants have been successfully decommissioned and dismantled to date. The dismantling of a reactor core and its surrounding pressure vessel (especially where this is of pre-stressed concrete construction) represents a formidable task, and will no doubt become a matter of future controversy. Clearly, decommissioning wastes are likely to include some large and awkwardly-shaped lumps of contaminated construction material, and correspondingly large waste units are likely to be required. Cope and Briscoe report that a single 250 MW(e) Magnox station would produce some 3100 tonnes of steel, 2500 tonnes of graphite and 15,000 tonnes of concrete as decommissioning waste (53).

The various stages at which decommissioning wastes may be produced are not illustrated in figure 9. However, it may be noted that all the secondary processing industries, including fuel enrichment and fuel reprocessing, require large-scale plant and equipment which will ultimately add to the inventory of decommissioning wastes. The scope of this problem has yet to be clearly defined within the international programmes on radioactive waste management and disposal.
Various methods for the treatment of intermediate-level wastes have been used or considered (19, 52, 68, 148, 214). Methods proposed for volume reduction include de-watering, shredding, laser cutting, compaction, incineration and acid ingestion. However the extent of the volume reduction must be limited, to prevent undesirable increase in radioactivity concentration levels, or generation of significant quantities of heat.

Conversion of intermediate-level wastes into forms more suitable for storage and disposal may involve immobilisation in one of the following matrix materials:

- cement;
- bitumen;
- polymer-impregnated cement;
- thermo-setting plastic resins (e.g. polyester, epoxy).

Other immobilisation proposals include incorporation into a ceramic matrix material by hot isostatic pressing or incorporation into metals by melting and casting (19, 59, 62).

In general, conditioned intermediate-level wastes are currently packaged in steel drums or concrete blocks of a standard size and shape (cylindrical or prismatic). Depending on the type of wastes and the design of the packages, the units may require some shielding during handling and some interim cooling. However, for all practical purposes associated with disposal, it may be assumed that heat output will be negligible (7).

As in the case of high-level wastes, the characteristics of the units, in terms of size, shape, volume and waste concentration may have a strong influence on the design
of an underground disposal facility.

4.4 Radionuclide Inventories

Successful radioactive waste management and containment clearly requires a knowledge of the composition of the wastes, both in radiological and chemical terms. The complete inventory of radionuclides present in intermediate and high-level wastes would comprise a formidable list, including actinides, fission products and neutron-activation products. Furthermore, with the passage of time, the relative concentrations of the various parent radionuclides and daughter products varies as numerous decay chains progress towards an equilibrium condition.

However, for present purposes, it is considered reasonable to simplify the problem by considering two distinct time periods comprising:

1. approximately the first thousand years after waste production;
2. the remaining long-term period, measurable in several thousands of years, during which only the longer-lived radionuclides are present in any significant quantities.

During the first thousand years, radionuclides having a half-life of less than 30 years will have decayed to less than $10^{-10}$ times the original quantity. None of the neutron-activation products would survive this period. The great majority of the fission products would also disappear.
The remaining, relatively long-lived radionuclides pose a more fundamental long-term problem; since their longevity means they have a greater chance of breaching man-made containment systems. The principal radionuclides within this group are identified in table 3.

The choice of the thousand year interval implies a confidence in the ability of future high-activity waste storage and disposal systems to ensure complete containment of all radionuclides during this period. Identifying the changing chemical composition during the first thousand years therefore becomes relatively unimportant. However, it remains necessary to understand the bulk radiological and heat production characteristics of the wastes.
5. RADIATION HAZARDS AND THE PERCEPTION OF RISK

5.1 Introduction

A confusing array of units are used to provide a quantitative description of the biological effects of radiation. This has arisen because radiology is a relatively new and complex developing science.

The intensity of radiation may be measured in terms of the number of radioactive emissions (or the number of radionuclides which decay) per unit of time. The process of radioactive decay means that the precise number of radionuclides present in a complex mixture of radioactive wastes is an elusive quantity. Activity (or rate of radioactive emissions) is therefore conveniently used as a measure of the total amount of radioactive material present (see Chapter 2.3).

Two units are in common use. The first is the Curie (ci); defined as the disintegration rate of radon in equilibrium with 1gm of radium. This is actually equivalent to $3.7 \times 10^{10}$ disintegrations per second (147). However, the bequerel (Bq) is a simpler unit which is now in more common use. One Bq corresponds to a decay rate of one radionuclide per second. Since the numbers involved tend to be very large, mega bequerels (Mbq) are more common.

As an illustrative example, the total activity within the core of a Magnox reactor, on load, has been estimated as about $1 \times 10^{14}$ MBq = $2.7 \times 10^{10}$ Ci (147). Extrapolation of other relevant data, from a variety of sources (50, 53, 112, 147, 163), suggests that the radioactivity contained in $1000\text{m}^3$ of liquid high-level
reprocessing wastes, such as those now stored at Sellafield, could be about $2.5 \times 10^{19}$ MBq.

Clearly, the concentration of the radioactive source is significant. Radioactivity concentration (or specific activity) may be measured and expressed in Ci/gm or MBq/gm. The radioactivity concentration of pure $^{239}$Pu, for example, is about $2 \times 10^3$ MBq/gm (147). Based on information provided in Chapter 4, and assuming a density of 2.67 gms/cc for borosilicate glass, the author estimates that the initial specific activity of vitrified reprocessing wastes from a thermal reactor at 15% fission product concentration (assuming immediate vitrification) would typically be about $4.5 \times 10^{11}$ MBq/gm.

However, descriptions of the radioactivity, or radioactivity concentration of the source are not sufficient to indicate the corresponding adverse radiological health effects. This depends upon the penetrating power and energy of the radiations which are emitted, and the manner in which the energy is dissipated to the matter through which it passes.

Radiation can cause biological damage in two ways (139, 193, 205, 206, 215). The first is by ionization, involving the liberation of electrons from molecules within the tissue, possibly causing these in turn to ionize neighbouring molecules. Ionization tends to promote chemical change and, since human cells are composed of 80% water, the creation of the chemically active free radicals $O^{2+}$ and $(OH)^{-}$ is a relatively likely occurrence. This may lead to various types of temporary or permanent cell damage or, alternatively, to cell death. Adverse reactions could also occur within cell nuclei; disrupting the normal process of cell
reproduction to cause cancers. The chance ionization of DNA molecules or DNA reactions with free (ionised) radicals in reproductive organs is also possible, and could lead to genetic defects in off-spring. A wide range of other potentially adverse health effects (temporary and permanent) are well documented (139, 205). However, the two extremes; namely potentially fatal cancers and genetic defects in off-spring clearly give rise to the greatest concern to individuals.

A second form of biological damage can be attributable to molecular excitation, where radiation energy is dissipated by heat rather than as chemical energy. Molecular excitation is the cause of radiation 'burns' which occur when relatively large radiation doses are received (139). This form of biological damage is readily associated with nuclear weapon bursts and large-scale industrial nuclear accidents. Where radiation doses are large enough to cause moderate or severe burns, the combination of direct and indirect biological damage may cause almost instantaneous death. However, consideration of high dose levels of this type is not relevant in the context of this thesis.

Penetrating power is an important factor in distinguishing the health hazards associated with different types of radiation (147, 193, 215). Alpha-radiation, for example, cannot normally penetrate the outer layer of dead skin. Hence, α-emitting radionuclides do not generally pose a major hazard in terms of direct radiation. Indeed the actinide elements, including uranium and plutonium, can be safely handled with a gloved-hand. However, inhalation or ingestion of α-emitting substances or contact with broken skin would lead to their assimilation within the body, and is likely to lead to severe and prolonged
internal tissue damage. Alpha-emitting radionuclides therefore pose a highly significant indirect radiological hazard.

Beta-radiation also has limited direct radiological hazard potential, since $\beta$-particles can normally only penetrate about 10-20 mm below the skin surface. Thus, relatively superficial tissue damage would occur as a result of direct exposure. Like $\alpha$-radiation, however, the indirect radiological hazard potential (due to assimilation) is high.

Unlike the particulate forms of radiation, $\gamma$-radiation is extremely penetrating and therefore poses significant direct and indirect radiological hazards. As with other forms of radiation, however, the indirect radiological hazard is greater, since the recipient will inevitably sustain a more prolonged and widespread body dose.

As described in Chapter 2.3, the energy of all types of radiation may be expressed in terms of the electron volt. However, for a given radiation energy, the rate of energy transfer differs according to the penetrating power, and hence the form of the radiation. Thus, $\alpha$-radiation dissipates its energy over very short distances due to its low penetration power. Beta-radiation dissipates its energy more widely; whilst $\gamma$-radiation only releases its energy over great distances, due to its exceptionally high penetration power.

5.2 Radiation Dose Assessments

Recognition of the different energy dissipation characteristics of different forms of radiation has led
to the definition of the 'absorbed dose'; defined as the amount of energy dissipated per unit mass of the material through which the radiation travels (205, 215). The original unit adopted was the 'rad'; defined as the quantity of radiation required to impart 100 ergs of energy per gramme of material (215). The unit is still widely used, but the preferred SI equivalent is the Gray (Gy), equivalent to 1 Joule/kg; hence 1 Gy=100 rads (147).

Due to the multifarious effects of high radiation doses, corresponding fatality rates are high. For example, it is reported that there is only a 50% chance of surviving a single whole body dose of about 4 Gy; and a 10 Gy dose would inevitably be fatal, due to excessive cell destruction in the digestive system and in red bone marrow cells (215). Such large single doses, however, could only be sustained under extreme conditions of sudden exposure which are not relevant here.

Due to differences in rates of energy dissipation as outlined above, the heavier particulate forms of radiation cause more biological damage per unit of absorbed dose. Therefore, a 'quality factor', Q, has been introduced which varies according to the type of radiation received, as follows (117):

\[
Q = \begin{cases} 
1 & \text{for electrons, } \beta \text{-particles and electro-magnetic radiations (e.g. X-rays and } \gamma \text{-rays)} \\
2.3 & \text{for neutrons} \\
10 & \text{for protons} \\
20 & \text{for } \alpha \text{-particles, fission fragments and other large particles.}
\end{cases}
\]

The 'dose equivalent' is then defined as:
dose equivalent = absorbed dose x Q ........ (1)

and is used as a measure of the biological hazard potential of all types of radiation. The c.g.s. and SI units of dose equivalent are the rem and sievert (Sv) respectively, where:

\[
\text{rem} = \text{rad} \times Q \quad \text{(2)}
\]

\[
\text{Sv} = \text{Gy} \times Q \quad \text{(3)}
\]

Neutron and proton radiations need not be considered further, since they are not produced in significant quantities by radioactive wastes. Therefore, based on the quality factors defined above, together with equation 1, it is apparent that \(\alpha\)-emitting transuranic radionuclides are 20-times more damaging in biological terms than the \(\beta\) and \(\gamma\)-emitting fission products and neutron-activation products, per unit of absorbed dose.

The significance of dose equivalents is that they may be numerically correlated with the frequency of occurrence, or risk, associated with one or more debilitating or potentially fatal biological effects. Extensive studies by the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) have indicated that different risk weighting factors should be applied, according to the different parts of the body likely to be affected (205).

Table 4, based on risk factors recommended by the International Commission on Radiological Protection (ICRP), indicates that the risk of contracting a fatal cancer or the occurrence of serious hereditary defects is 1 in 60 per Sv of effective dose of whole body radiation (117). The fractional contributions towards
this risk due to selective radiation of various parts or organs of the body are also indicated. Thus, a 1 Sv dose applied only to the thyroid would imply a 1 in 2000 risk of fatal cancer and is said to be 'equivalent' to a whole body dose of 0.03 Sv.

This gives rise to the important concept of 'effective dose equivalent', which forms the basis for radiological assessments performed by the ICRP, NRPB and other organisations (147). The effective dose equivalent provides a measure of the equivalent whole body radiation dose when one or more parts of the body are irradiated, and is calculated by use of risk weighting factors according to the expression:

$$H_{wb} = T \cdot W_T \cdot H_T$$ .................................. (4)

where $H_{wb}$ is the effective dose equivalent in terms of whole body radiation

$H_T$ is the dose equivalent for tissue $T$

$W_T$ is the appropriate risk weighting factor for tissue $T$ (based on the values shown in table 4)

The assessment of risk factors by UNSCEAR and ICRP is based on past observations, a significant proportion of which are associated with radiation victims at Hiroshima, Nagasaki, the Marshall Islands, early uranium mine workers, etc. It is partly the immotile aspects of the use of such data, and a lack of understanding of the statistical methods of evaluation, which gives rise to controversy over radiological protection assessments, as outlined in Chapter 1.

The National Radiological Protection Board (NRPB) has estimated that the average effective dose equivalent attributable to all radiations received in the U.K. is
currently about 2.4 m Sv per year per head of population (147). Of this total, some 78% is attributed to natural sources (e.g. cosmic radiation) and some 20.7% to medical radiation treatments, including X-rays. The remaining 1.3% of the total is said to be distributed approximately as follows:

- 0.4% nuclear fall-out from weapons testing
- 0.4% miscellaneous (luminous watches, T.V. receivers, etc.)
- 0.4% occupational exposures (radiation workers)
- 0.1% industrial discharges (assumed to mean unconditioned, low-level radioactive effluents)

However, the average annual effective dose equivalent of 2.4mSv per head of national population, when correlated with the risk factors shown in table 4, would give misleading results. In practice, the risks associated with radiation vary widely around the country, according to the natural surroundings and life style of various population groups.

The risk of occurrence of a given biological effect actually correlates with the 'collective dose equivalent' to which a given population is subjected. The effective dose equivalent per head is then obtained by dividing the collective dose equivalent by the population concerned. Thus, for a given collective dose equivalent received by a particular sector of the population, the risk to individuals becomes greater as the population size is decreased (139).

Based on the current NRPB estimates described above, the current collective dose equivalent from all radioactive sources within the U.K. is about 2.4m Sv x 60 x 10^6 =
1.44 x 10^5 man-Sv/year (assuming a U.K. population of 60M people). Now the 0.4% of this collective dose equivalent received by radiation workers alone is therefore 576 man-Sv/year, and since this is distributed amongst some 110,000 workers (147), the average occupational dose equivalent received is 5.2m Sv per year per head. However, the effect of these operational exposures averaged over the whole population is negligible in terms of the average dose equivalent per person. Hence, on average, radiation workers appear to be at greater risk of adverse radiological effects compared with other members of the population by a factor of (5.2 + 2.4)/2.4 = 3.2.

A similar, but more hypothetical example of the effect of population size could be illustrated for the industrial discharges which are said to contribute only 0.1% of the total background radiation. Spread over the entire U.K. population, the average effective dose equivalent per individual would be of the order of 2.4 μSv per person per year. However, let us suppose that these discharges were to be focussed in the vicinity of one town whose population was of the order of 50,000 inhabitants. The average effective dose to that section of the population would then be 2.99m Sv per person per year, and accordingly the inhabitants would be (2.99 + 2.4)/2.4 = 2.2 more times at risk than the average for the remainder of the national population. By similar reasoning, radiation workers who might live in the same town would typically be (5.2 + 2.99 + 2.4)/2.4 = 4.4 times more at risk.

The above simple examples illustrate the way in which the effects of radiation doses vary in relation to the magnitude of the effective dose and the size of the population concerned. The effects of an accidental
concentrated release of activity on a small population may thus be more readily envisaged. Clearly, it is important to identify those populations which are at greatest risk, in order to make appropriate radiological hazard assessments.

However, it is apparent that the dose equivalents attributable to current industrially controlled radiations are low and sustained whereas, as previously noted, the available data concerning biological effects in humans have generally been obtained from incidents involving relatively high doses over short periods of time. Controlled radiological experiments with mice and other animals, involving a wide range of doses and exposure periods, indicate a relatively rapid reduction in health risk with reduction in dose and dose rate (139, 205). The problem (perhaps fortunately) in the case of humans is that insufficient data exists to predict a similar trend (147).

Thus, the guidelines formulated by the International Commission for Radiological Protection (ICRP) are based on a simple linear extrapolation between health risks and effective dose; irrespective of the period of exposure (117). Such a basis for radiological protection standards is seen to be a necessarily cautious approach, in the absence of sufficient data concerning the effects of prolonged low radiation doses on human beings. Almost all countries have followed the ICRP recommendations in formulating their industrial regulations concerning the release of radioactivity to the environment (147).

The ICRP recommendations provide considerable technical detail. However three fundamental requirements are stipulated, namely:
1. No practice shall be adopted unless its introduction produces a net positive benefit.

2. All exposures shall be kept as low as reasonably achievable (ALARA); economic and social factors being taken into account.

3. The dose equivalent to individuals shall not exceed the limits recommended for the appropriate circumstances by the ICRP.

For individual members of the public (i.e. non-radiation workers) the maximum effective dose equivalent recommended by the ICRP is 5mSv per year. However, in cases where prolonged exposures could occur, ICRP recommends that the maximum effective dose equivalent should be limited to 1mSv per year. Significantly, the ICRP has not yet been able to decide whether these recommended dose limits should also apply in relation to high-activity radioactive waste disposal.

As shown in 5.3 below, the application of the three basic ICRP requirements to the disposal of high-activity radioactive wastes is far from straightforward.

5.3 Long-Term Hazards Associated with High-Activity Wastes

Based on the outline review of standard radiological hazard assessments in 5.1 and 5.2 above, it is possible to illustrate the actual health risks associated with high-activity wastes; albeit in a very general and necessarily rather unrepresentative way. The problem is to predict the radiological consequences of the release of the wastes to the environment. However, as shown in
5.2, the distribution of the release is significant. Assuming that attempts are made to isolate the wastes for extremely long periods (e.g. by geological disposal) the timing of the release is also important.

Figure 11 is reproduced from a well-publicised paper by Cohen (50), and is intended to illustrate the potential direct and indirect radiological hazards associated with high-level reprocessing wastes, as a function of time. Figure 11a indicates the predicted number of radiation fatalities per year in the United States, in the unlikely event that all high-level reprocessing wastes arising from 400 GW(e) year PWR nuclear power production were to be simultaneously spread, in a random fashion, on the ground.

As shown, the occurrence of direct-radiation fatalities reduces as a function of time, and becomes relatively insignificant after approximately 1000 years. This corresponds to the period required for the decay of most of the $\beta$ and $\gamma$-activity associated with the fission products (c.f. the heat-decay curve shown in figure 10). As described in 5.1 above, $\alpha$-radiation poses a comparatively insignificant direct radiological hazard; hence the continued presence of the long-lived transuranic elements would pose a comparatively low direct radiological hazard.

Figure 11b shows the predicted indirect radiological hazard potential in terms of fatal cancers per year if;

(a) the wastes were all to be directly consumed by the American population (left-hand scale)

(b) the wastes were to be dumped randomly in rivers; thereby contaminating the nation's drinking water
It is apparent that the predicted rate of fatalities is also decreased by many orders of magnitude during the first 1000 years, due to fission product ($\beta$ and $\gamma$-radiation) decay. However, a significant radiological hazard persists for several millennia, due to the longevity of the relatively radiotoxic $\alpha$-emitting transuranic elements.

The diagrams show that shielding of the wastes for 1000 years would reduce the direct radiation hazards considerably. However, it is clear that the indirect radiological hazard poses a considerably longer-term environmental problem.

The author is not qualified to comment on the validity of the hypothetical predictions made in Cohen's diagrams. There is certainly no reason to doubt them. However, their primary value is in indicating the comparative hazards, in terms of the radionuclide composition of the wastes as a function of time. Since the distribution scenarios on which they are based are totally unrealistic, they cannot reflect the real health risks to representative sections of the American population due to high-level reprocessing wastes, either now or in the future.

For example, figure 11b predicts that release into rivers at 10,000 years after reprocessing would result in a fatal cancer rate of about 20 per year. Spread over the entire population of the United States, (say about 200M people) this represents a risk of $1 \times 10^{-7}$ per person per year. However, in the more likely event that the release were relatively concentrated, so as to contaminate the drinking supplies of one city of 0.25M
people, for example, the risk to the city-dwellers would be about 1 in 12,000 per year. At 1000 years after reprocessing this risk would be about 1 in 200 per year. A similar risk of genetic defects would also be incurred together with a much higher risk of a whole series of debilitating but not necessarily fatal illnesses. In addition, a proportion of the wastes would inevitably be ingested in local food supplies (through animilation by crops, plants and animals, etc) to produce superimposed health risks.

Generalised hypothetical assessments can therefore be highly mis-leading when examined out of context. In many cases, there can be no doubt that mis-leading statements have been made by representatives of the nuclear establishment. One of the statements frequently encountered, in relation to geological disposal, refers to the fact that the amount of radioactivity introduced into the ground by construction of a deep underground repository would represent a comparatively small addition to the radioactivity contained in the overlying crustal rocks themselves. Such statements clearly fail to identify the extreme concentration of radioactivity in the wastes by comparison with the rocks, and the consequences of release and migration of radionuclides along specific pathways at some future time, to hypothetical, near-by populations.

For geological disposal concepts, in particular, there are immense problems in assessing the possible range of collective dose commitments and associated risks of adverse health effects to future generations. This is due to the difficulties in modelling the mechanisms of radionuclide release and migration through natural geological materials; and subsequent assimilation through drinking water supplies or food chains. In
addition, there are problems in predicting future demographic trends, since it cannot be assumed that centres of population will remain static. At best, therefore, the adequacy of predictive radiological modelling is only as good as the assumptions applied and it must be recognised that they are subject to considerable uncertainty. (57, 98A, 99, 100).

Hence, serious problems arise in deciding what long-term radiological protection criteria should be applied for radioactive waste disposal. The three central principles put forward by the ICRP in relation to current practices may be used as a reference for discussion; see 5.2 above.

The first ICRP principle is self-evident. The second principle, known as the ALARA principle (As Low As Reasonably Achievable) can be interpreted in a number of ways. The most likely industrial and economic interpretation would be that waste disposal procedures should use all possible means of preventing long-term releases of radioactivity; provided that the associated costs do not become disproportionate in relation to the cost of the electricity produced. This interpretation is clearly open to criticism by those who would question whether the effectiveness of the waste disposal system should be influenced by cost restraints and the need to maintain the economic viability of the nuclear industry.

In the author's view, the ALARA principal does not help significantly in defining the standards to be applied for high-activity radioactive waste disposal. However, it does perhaps help to indicate the basis on which alternative disposal practices should be compared; namely the likelihood of radiation exposure. Thus, for any given hypothetical release and environmental
contamination scenario, the preferred disposal system should be the one resulting in the lowest risk level associated with a given level of exposure. Only when similar results are obtained, or when risk levels are insignificant, should cost comparisons be taken into account.

The above leads to difficulties in comparing one system which may be subject to relatively low probability/high-consequence events, with another which is subject to relatively high probability/low consequence events. The common element here is the risk to the health of individuals or to a hypothetical population, and this inevitably leads back to the question of the radiological protection standards to be applied now in designing disposal facilities which are intended to provide extremely long-term future environmental protection.

Thus, the central issue is currently embodied in the third IRCP requirement; i.e. the dilemma consists of establishing what dose limitations should apply and how to assess the extent to which these limitations can be assured by a given disposal system.

Clearly, the most ethical answer to the first of these questions is to assume that the radiological protection standards to be applied to future generations should be at least as stringent as those applied to our own. At present, there is no international consensus concerning the resolution of this issue (98A). However, the National Radiological Protection Board (NRPB) has recently published its own guideline recommendations which are based on an extension of the current ICRP principles (146).
The NRPB recommendations recognise that the only effective rational basis for establishing performance criteria for the efficacy of high-activity disposal systems is on the basis of risk; since the sequences of events and processes which could lead to environmental contamination from a waste disposal system have either constant or time-varying probabilities of occurrence. In this context, NRPB define risk as:

\[
\left[ \text{probability that a given dose will be received} \right] \times \left[ \text{probability that the dose will cause deleterious health effects} \right]
\]

The NRPB recommendations suggest that the single effective dose equivalent limit of 5 mSv per year and the prolonged effective dose equivalent limit of 1 mSv per year, as recommended by the ICRP, should be replaced by appropriate risk limit objectives for limiting the occurrence of fatal illness or serious genetic effects. Their recommendations are:

(i) the risk to an individual from the waste disposal system due to a dose of short duration should not exceed \(10^{-4}\) per year (1 in 10,000);

(ii) the risk to an individual where doses could be received over periods exceeding 10 years should not exceed \(2 \times 10^{-5}\) per year (1 in 50,000).

(iii) events having extremely low probabilities of occurrence of about \(10^{-10}\) to \(10^{-12}\) per year should be excluded from total risk calculations, on the basis that low probabilities of this order are not considered when taken decisions on the acceptability of any present day practices.
These risks may be compared with the risk factors given in Table 4 for a 1 Sv annual dose; and it may be noted, for perspective, that the risk associated with the current average annual effective dose per head of population in the U.K. of 2.44 mSv is equivalent to a risk of $4 \times 10^{-5}$ per year (approximately 1 in 25,000). For further comparison, the ICRP dose limits of 5 mSv and 1 mSv per year are equivalent to risks of $2.25 \times 10^{-5}$ (1 in 12,000) and $1.65 \times 10^{-5}$ per year (1 in 60,000) respectively. It may be seen, therefore, that the NRPB recommended risk limit objectives for high-activity waste disposal practices simply correspond to a 20% improvement on the current ICRP recommendations concerning dose limits for operational industrial radioactive waste discharges.

Risk limit objectives appear to provide a more appropriate basis for the establishment of performance criteria than hitherto. However, as will be shown in later sections of this thesis, the problems inherent in long-term risk assessment, especially when based on containment by natural geological materials, are so great that margins of uncertainty will generally exceed desirable threshold limits. In the author's view, therefore, the over-riding objective in the design of a disposal system must be to mitigate against these areas of uncertainty.
6. DISPOSAL CONCEPTS

6.1 Introduction

A very wide range of concepts for the disposal of high-activity radioactive wastes has been considered over the past 25 years or so. These vary considerably in terms of technological difficulty, risk and cost.

Much of the original thinking may be attributed to American scientists and engineers who initiated serious research programmes under the auspices of the Atomic Energy Commission in the late 1950's. Elsewhere, little development took place until the mid-1970's, when research programmes were first undertaken by Canada, Sweden and various member countries of the European Economic Community (9).

However, by this stage a consensus had emerged, (as a result of American studies), in favour of deep underground disposal, and subsequent studies have generally been concentrated on the further development of this concept; see Part 3. In the light of more recent developments, some of the early American proposals now appear somewhat bizarre; both in the environmental sense and from an economic and engineering viewpoint. However, in the author's view, the initial development of a wide range of disposal concepts by earlier American workers has greatly benefited the progress of more recent international studies in this field (194).

The remaining sections of this Chapter review the essential features of disposal concepts which have been advocated to date; including those which do not involve underground burial. In section 6.3, the geological disposal concept is described in rather more detail.
6.2 Non-Geological Disposal

A review of the available literature shows that four alternatives to geological disposal have been seriously considered at a conceptual level. These are:

- transmutation
- extra-terrestrial disposal
- ice sheet disposal
- sub-sea bed disposal

The first two of these concepts involve the complete destruction or removal of radionuclides from the wastes. The remaining two involve isolation and containment over extremely long periods, so that the radioactivity concentration of the wastes is reduced to negligible levels before there is any likelihood of release to the environment.

6.2.1. Transmutation

As shown in Chapter 2.3, transmutation occurs during the natural process of radioactive decay, which ultimately leads to the conversion of radionuclides into stable forms. However, it can also occur as a result of artificially induced changes in nuclear structure due to bombardment with sub-atomic particles, as occurs during the fission process in a nuclear reactor. There is therefore an intrinsic appeal in the concept of an industrial-scale process for the transmutation of the radionuclides contained in radioactive wastes into non-radioactive isotopes.
Transmutation into stable nuclides or shorter-lived radionuclides could theoretically be achieved in some of the wastes by further irradiation, as proposed by Liikala, Burnham et al (132). However, from a purely economic view-point, it would be necessary to ensure an energy balance such that the transmutation process consumes significantly less energy than is originally produced by the fission reactor in generating the wastes. Furthermore, the rate of transmutation achieved would have to be significantly greater than the rate of decay of the radionuclides concerned.

In reviewing the possibilities, Liikala, Burnham et al concluded that it would be possible to transmute fission products into stable nuclides by neutron irradiation. However, their relatively low neutron-capture cross sections would require extremely high neutron energies and intensities, obtainable only in:

- spallation accelerators
- thermo-nuclear explosions
- fusion reactors

The first of these options would be technologically difficult and probably economically unviable. The use of thermo-nuclear explosive devices would clearly be unacceptable to the public, and would in any event create further fission products. The use of neutron radiation from fusion reactors is theoretically possible, but it would diminish the effective heat generation capacity (and hence efficiency) of the fusion reactor. In any event, industrial scale fusion reactor technology is not likely to be available for several decades (164).

Furthermore, transmutation of the fission products would require the introduction of a chemical partitioning
process in order to separate them from the actinides. For high-level wastes, this would impose substantial additional fuel reprocessing costs, and in the case of intermediate-level wastes would probably be impracticable, both from a technological and economic viewpoint.

Even if the transmutation of the fission products could be achieved, transmutation of the actinides would only create other long-lived radionuclides of the same series. In general, the latter would be of no value. However, as described in Chapter 3, the large-scale transmutation of $^{238}U$ into $^{239}Pu$ through the use of fast breeder reactors is already contemplated; see figure 9. This prospect is advantageous in waste management terms, since the breeding of plutonium fuel from stocks of depleted uranium would serve to reduce the overall volume of intermediate-level wastes.

Taken as a whole, however, transmutation cannot be regarded as an effective waste disposal solution. The technology for transmutation of the fission products is not yet available, would require substantial additional costs in terms of chemical partitioning, and could be applied only to high-level reprocessing wastes. Transmutation of the actinides would provide no benefit, except in relation to the limited application of fast breeder reactors as outlined above. Thus, based on current knowledge, transmutation can only offer the prospect of a partial waste disposal solution. It appears to be an ideological proposition rather than a practical one, and cannot rank highly among the priorities for future research.
6.2.2 Extra-Terrestrial Disposal

Rather more detailed consideration has been given to proposals for the extra-terrestrial disposal of high-activity radioactive wastes (60, 191). Drumbeller, Brown et al have evaluated a number of possible space trajectories, based on the deployment of special 'waste tugs' from the American space shuttle (60). However, in each case, the cost of carrying the payload is found to be prohibitive if both high-level and intermediate-level wastes are included. Therefore, only high-level wastes are considered, and chemical partitioning is proposed as a means of isolating the extremely long-lived radionuclides, comprising the transuranics together with the long-lived fission products (e.g. $^{129}$I and $^{99}$Te), for transportation into space. This means that a partial disposal solution would be achieved which would be more or less the reverse of that associated with transmutation. However, the remaining disposal problem presented by the fission products would arguably be a less onerous one, in view of their comparatively short lives.

Irrespective of the choice of space trajectory, a specialised form of waste packaging would be required, capable of withstanding a major launch-pad accident, or atmospheric re-entry and earth impact (at an estimated velocity of 11Km/sec). One of the specialised forms of container proposed by Drumbeller, Brown et al would comprise an outer skin of protective high-impact steel; the overall package comprising a 1.5m diameter cylinder, 1.5m in length.

The space trajectories considered by both expert groups (60, 191) include:

- high earth orbit (at approx. 150,000km)
- impact with the sun
The above order is reported to correspond, in broad terms, with increasing launch energy requirements, levels of risk and technological difficulty. The trajectories for solar impact and solar system escape, for example, would both require additional waste capsule accelerations to be achieved by planetary 'swing-by'; thereby increasing technical problems and also limiting the timing of launches to coincide with optimum planetary positions. The remaining orbital trajectories, although technologically less difficult, are said to pose other significant risks in terms of orbital instability and possible return of the wastes to the earth.

The range of costs, estimated by Drumheller, Brown et al increase in a similar rank order from about $2000 per Kg to $8000 per Kg of wastes (at 1972 prices); excluding the costs of land transport and waste packaging (60). Ruppe, Hayn et al predict similar order of magnitude costs (191). However, none of the estimates appear to include the cost of chemical partitioning, the long-term management of the fission products, the construction of special transport containers and waste handling facilities, or the cost of constructing special launch-pad facilities. These additional costs are likely to cause significant increases in the overall estimates.

Furthermore, it is clear that few countries would be in a position to opt for industrial-scale space disposal within the foreseeable future, and wide-scale international co-operation would therefore be necessary for success. Perhaps most significantly, it appears
unlikely that the international public-at-large would be prepared to accept either the possibility of high-consequence events associated with launch failure or re-entry, or the general political implications of the extra-terrestrial disposal concept. International treaties already exist which prevent industrial pollution of the moon. It is quite easy to imagine a hard-fought international lobby against any proposals involving the industrial-scale pollution of space in general.

6.2.3 Ice Sheet Disposal

Ice sheet disposal offers considerable appeal, due to the remoteness the continental ice sheets; which actually occupy some 11% of the earth's land area. The most detailed and authoritative technical assessment to date appears to be that provided by Tillson, Wallace, et al (202), which considers three basic alternatives.

The first of these, termed the 'melt down free-flow concept', applies only to high-level waste units. Holes would be drilled into the ice sheet at approximately 1km intervals, to depths of the order of 50 to 100m, and one or more waste units would be placed in each hole. The heat produced by the fission products would melt the ice to form an envelope of water around each unit, bringing about a descent at an estimated rate of 1 to 1.5m per day. Thus, the estimated time for penetration to the base of a 3000m thick continental ice sheet would be about 5 to 8 years.

As a refinement to this concept, Tillson, Wallace et al have suggested that the waste units could be attached by cables to large anchor blocks resting on the surface. In theory, the mass and geometry of the anchor-block
assembly could then be adjusted to obtain any desired rate of descent through the ice. An arrangement is suggested which it is claimed would increase the descent period to some 30,000 years. This implies that the majority of the descent would be due to weight only; since significant levels of heat output would only occur during the first 1000 years.

The author is skeptical of this proposal, since it requires cable attachments to withstand the high rate of corrosion attack associated with melt water around the waste units during the prolonged period of thermal output. It must be envisaged that the brine produced by the melting ice would have significant corrosion potential.

The third proposal appears to have a sounder practical basis. This involves placing a number of units on a slab, supported by jack-up piles. The units could be stored for a prolonged period by jacking the platform as the piles sink into the ice. At any desired stage, the whole assembly could then be allowed to sink into the ice at a suitably slow rate.

However, each of these proposals poses a number of technical problems, since the behaviour of large ice masses is not fully understood. Continental ice masses are formed where the mean rate of precipitation of snow and subsequent ice formation exceeds the rate of surface evaporation, melting and run-off. However, ice masses are not static. The build-up of hydrostatic stress leads to visco-plastic flow towards the margins, where ice wastage occurs (formation of ice-bergs, etc). Based on surface measurements, rates of movement of ice may vary from as much as 2.7m per day for glaciers to 0.05m per day for ice sheets in areas of relatively low relief.
However, if water lubrication occurs at the base of the ice, rates of movement may be accelerated considerably (219). Clearly, such a situation could arise where heat-generating high-level waste units reach the base of the ice sheet. Moreover, a pre-existing basal water layer could provide a direct pathway for radionuclide migration to the sea.

For concepts involving more gradual descent, further problems are encountered in predicting the long-term rate of ice accumulation. Wallace, et al (202) recognise that ice sheet stability cannot be predicted over periods in excess of a few thousand years. Future climatic changes could alter the mass balance between precipitation and lateral removal of ice, making predictions of very long-term containment highly speculative.

In any event, logistic and socio-political factors appear to over-ride the purely technical considerations outlined above. Firstly, the hostile environment which renders the continental ice sheets uninhabitable also imposes severe logistic problems in terms of waste transport and repository operation. The over-land transport of waste units from specially designed port facilities would present severe practical difficulties; as would the waste handling and emplacement operations themselves; note that temperatures as low as -88.3°C, with wind speeds up to 200Km/hr, have been recorded at weather stations in Antarctica (219).

Secondly, the only two potentially suitable continental ice sheets are Greenland and Antarctica. The former is owned by Denmark, who is unlikely to accept the role of
host to the world's 'nuclear dumpers'. Antarctica is a more massive ice wilderness, but waste disposal has been specifically banned under the terms of the 1959 Antarctic Treaty. Thus, technical considerations apart, it seems that ice sheet disposal cannot be considered as a practicable option within the foreseeable future.

6.2.4 Sub-Sea Bed Disposal

Sub-sea bed disposal has also been considered in detail and remains a serious technical contender, albeit generally at a lower level of priority than land-burial proposals. As mentioned previously, simple sea-bed dumping of low-level wastes is already carried out. However, this procedure amounts to a 'dilution and dispersal' system which would not be appropriate for high-activity wastes (62, 131).

For high-activity wastes, deep penetration of sea bed sediments is required. This may be achieved by:

- free-fall/dynamic penetration
- static penetration (constant force)
- insertion into pre-drilled holes

The free-fall system would require the design of specially profiled waste units, designed to embed themselves deep into the ocean sediments. Dynamic or static penetration would require a specially adapted drilling platform vessel, utilising a drill string and waste unit driving apparatus. The use of pre-drilled holes would provide a more positive assurance of penetration than any of the other methods, but is necessarily limited in terms of operating depth and borehole diameter.
It has been suggested by Valent, Lee, et al (212) that the deep oceanic trenches could be ideal sites for sub-sea bed disposal. In these trenches, the sediment thickness may allow deep penetration; and further downward movement may occur due to liquification associated with frequent earthquakes. In addition, subduction theories suggest that the wastes could ultimately be carried far below the earth's crust.

Unfortunately, however, the great depth of the oceanic trenches makes them very difficult to explore. They are thought to be V-shaped, with relatively narrow, flat bottoms some 500-3000m wide, filled with turbidite sediments or graded sand and silt beds. However, free-fall methods of emplacement can probably only achieve a maximum target accuracy of ± 1m per 275m of depth, which would be insufficient for subduction trenches 20-30Km below sea level (212). Also, drill-strings cannot be operated to depths of such magnitude.

In addition, if subduction theories are accepted, sediments (and the waste units) may actually be scraped upwards by the landward wall, as the opposing plates move in relation to each other (106). It therefore seems that the abysal planes of the sea bed are to be preferred as potential disposal areas.

However, it should be noted that the sub-sea bed disposal concept is unlikely to be viable for intermediate-level wastes. The system requires that waste units should be long and thin and preferably cylindrical in shape. For static penetration or drill-hole methods of emplacement it is unlikely that waste unit diameters in excess of 0.5m would be
acceptable. Whilst these constraints do not impose undue difficulties in the manufacture of high-level waste units, some intermediate wastes (particularly decommissioning wastes) would require excessive and impracticable processing in order to produce packages of the required shape and size.

Moreover, it may be readily seen that any proposal involving sub sea-bed disposal in international waters is likely to meet considerable political opposition. Furthermore, current understanding of the geotechnical and containment properties of deep sea sediments is incomplete. In the author's view, sub sea-bed disposal is therefore unlikely to be adopted as a viable practical solution to the high-activity radioactive waste disposal problem. A review of the literature also indicates that the concept has received relatively little detailed technical consideration at an international level by comparison with underground disposal concepts.

6.3 Geological Disposal

The geological disposal concept involves the deep underground burial of high-activity wastes within a stable host rock environment. The geological record provides ample evidence that many types of rock formation will remain stable over periods exceeding the radioactive life of the wastes; whereas no comparable proof can be made available for any man-made containment material.

Furthermore, deep burial provides a means of imposing a substantial material barrier thickness between the wastes and the biosphere without any immediate surface
environmental impact. It is also perceived that underground burial could be achieved at low cost using relatively simple adaptations of existing civil and mining engineering technology, and that repositories may be created within any waste-producing country. Thus, each nuclear power producing country can develop and implement its own waste management and disposal system.

Based on figure 11a, it is clear that provided the wastes remain buried (shielded) for a thousand years or so, no appreciable direct radiological hazard can occur thereafter. Such a period of confinement is within the limits of engineering design experience as evidenced by archeological remains. However, figure 11b shows that a significant indirect radiological hazard will persist over periods which are significant even on the geological time-scale. The design of containment facilities with an operating life of such magnitude are clearly beyond the realms of human experience.

Because of the time-scale of containment envisaged, catastrophic events which are normally discounted due to their extremely low levels of probability, may need to be re-considered in detail. Likewise, phenomena which are normally discounted because they produce extremely slow rates of change become highly significant and require close examination.

Various scenarios have been developed to describe catastrophic mechanisms which could lead to a significant release of radionuclides from a deep-level underground repository (181). These include:

a) direct exhumation of the wastes by inadvertant human intrusion;
b) natural cataclysms such as earthquakes, vulcanism or meteor (or weapon) impact;

c) accelerated natural erosion due to climatic changes, eustatic variations in sea level, orogenesis, isostatic uplift of land masses, glaciation, etc.

The inclusion of (a) above recognises that an underground repository must continue to achieve isolation of the wastes despite the fact that records of its existence may be lost or forgotten or that institutional controls may be suspended at some time in the future. Clearly the likelihood of a random human intrusion for construction purposes at great-depth is remote, whilst the likelihood of intrusion due to mining activities could be reduced to very low levels by selecting a host rock having a suitably low mineral resource value. The risks associated with the other events ((b) and (c) above) may also be reduced to suitably low levels by siting the repository in an area of proven tectonic stability and by achieving a burial depth well below the influence of sub-aerial geological processes. Based on the NRPB risk criteria described in Chapter 5.3 above, the siting of a repository should ensure that the risks associated with catastrophic events of this type should be less than $10^{-10}$ per year (146).

However, the gradual interaction of the buried wastes with surrounding rock and the effects produced by circulating groundwater (where present) cannot be prevented. Thus, most geoscientific studies have recognised that the dominating mechanism by which contamination of the biosphere could occur involves the gradual degradation of waste packaging materials.
followed by the slow leaching of the wastes and subsequent migration of radionuclides by groundwater flow and molecular diffusion processes as illustrated in figure 12.

It is generally assumed for repository design purposes that the probability of such a process is 1. However, assuming migration occurs through the rock mass as envisaged, migration would be accompanied by considerable dilution and dispersion of the wastes and, for large burial depths in low permeability rocks, extremely low radionuclide migration rates are implied. Nevertheless, it is recognised that contaminated groundwater flowing via long and tortuous pathways might ultimately reach a near surface aquifer and thus gain access to the biosphere.

The geological disposal concept cannot eliminate this long term possibility, but relies upon the containment properties of natural host rocks to provide an extremely long delay in the leaching and migration processes. The concept therefore seeks to achieve long-term radiological protection by maximising the delay to the onset of radionuclide release, and thereafter reducing the rate of release to the environment to extremely low levels so that the risks of producing significant radiological doses to man are below the threshold levels of acceptability (186).

As shown in Chapter 5 there are problems in deciding what radiological protection criteria should be applied to future populations who could be potentially at risk, and in deciding upon the probable size and geographical distribution of specific sectors of these populations. However, there are equally significant technical difficulties in modelling extremely long-term migration
phenomena within the underground environment and in ascribing confidence limits to the results obtained.

It is recognised that groundwater will act both as a geochemical agent for the dissolution and release of radionuclides from their point of emplacement, and as a transport vector for their migration through the host rock. Modelling of the long-term processes involved requires the identification of each mechanism involved, its risk of occurrence through time, its rate and the secondary consequences which may arise therefrom (181, 183, 186).

Two principal barriers to radionuclide release and migration have been recognised. These are:

- the waste units; comprising the outer claddings of the waste units and the internal waste immobilisation matrix
- the host rock itself.

The waste units may provide total containment for a period which is significant on the human time-scale. However, archaeological evidence suggests that no human artifact can be relied upon to survive for periods longer than about one or two thousand years. In view of this comparatively short period of containment, it may be argued that the expense involved in packaging the wastes in high-integrity containers is not warranted, particularly for the relatively large volumes of intermediate-level wastes. Most conceptual proposals have adopted this view and have placed the greatest emphasis on the natural host rock as the primary barrier to radionuclide release and migration.
Other important distinctions must be drawn between the disposal requirements for high-level and intermediate-level wastes. Both categories contain significant quantities of long-lived radionuclides of the actinide series, and therefore the criteria for long-term containment may be regarded as virtually identical. However, the activity concentration of high-level wastes is far greater over the first thousand years or so due to the presence of large quantities of fission products and, as shown in figure 10, this also results in significant heat output.

Thus, at the time of disposal and for several hundred years thereafter, high-level wastes must dissipate significant quantities of heat to the surrounding host rocks. The thermal perturbations within the underground environment may be expected to cause considerable acceleration of physico-chemical processes, involving waste/rock and waste/groundwater interaction and could produce irreversible changes which might impair the longer-term performance of the repository system.
PART 2
HOST ROCKS
7. SELECTION CRITERIA

Part 1 has provided a broad background to the problem of high-activity radioactive waste disposal, including the origin and characteristics of the wastes and an indication of the long-term radiological hazards they represent. Chapter 6 has described the essential features of the various waste disposal concepts which have been and still are under consideration; indicating why (in the author’s view) underground disposal on land areas offers the greatest advantages. This part of the thesis, (Chapters 7 to 10), examines relevant characteristics and properties of potentially suitable host rock formations.

The selection of a suitable rock formation is clearly one of the most crucial decisions which must be made in relation to the underground disposal system. There are two fundamental aspects to be considered, namely:

- the containment properties of the rock
- the engineering feasibility of repository design, construction and waste emplacement

The above have formed the basis for the majority of the geoscientific and engineering research which has been carried out to date. However, in addition to the review of these aspects, this thesis will consider three additional elements. These are:

- the uncertainties which are inherent in assessing the containment properties of host rock masses
- the extent to which the construction of a repository system may impair the long-term performance of the
rock mass as a natural barrier to radionuclide release and migration

o the engineering opportunities which may exist for mitigating against these problems and improving the overall performance of the underground disposal system

Host rocks can prevent significant release of radionuclides from buried wastes in two ways; firstly by imposing physical barriers to radionuclide transport in groundwater and secondly by virtue of physico-chemical properties which prevent or inhibit radionuclide release and migration at a molecular level.

However, construction of a deep-level repository and emplacement of radioactive wastes may not only present problems in terms of engineering feasibility; it may also disrupt the physical barrier by introducing new migration pathways and impairing the physico-chemical retention properties of the underground regime.

The choice of a particular rock type for the construction of a radioactive waste repository obviously depends upon the characteristics of formations available within the country concerned. The repository must be sited in an extensive and relatively homogeneous formation with a favourable hydrological setting, characterised by extremely long groundwater flow paths and minimal hydraulic gradients. The area should also be one of minimal seismic risk with an absence of major faults. Furthermore, the location will have to satisfy numerous 'short-term' requirements associated with factors such as waste transport, economy of operation, security and environmental impact.
In practice, these constraints will tend to reduce the range of options for repository siting considerably. Nevertheless, since the primary role of the host rock is to act as a natural barrier to radionuclide release and migration, the intrinsic containment properties of the rock mass are the most fundamental consideration.

Based on the release and migration mechanisms identified in Chapter 6.3, it is evident that extremely low mass permeability is an essential requirement. The material should also be resistant to chemical weathering, solutioning and erosion, and should not have any foreseeable mineral resource potential. An ability to sorb radionuclides by ionic exchange and/or precipitation reactions is a further advantage; and, for high-level waste disposal, temperature stability and good heat transfer properties are desirable qualities.

From the construction viewpoint, the material should be competent (in the engineering sense) at the depths under consideration (typically 500m to 1000m) in order that its containment properties are not impaired by the construction process. It is also desirable that the rock should exhibit a plastic or viscous response to stress change so that transmissive discontinuities are not introduced.

Clearly, not all of these desirable qualities can be found in any one rock type. The search for potentially suitable host rock formations commenced in the United States in 1954, and for a considerable period was focussed upon the bedded saliferous deposits of the Permian salt basin in Utah and the interior salt domes of Kansas (54, 66). However, in the mid-1970's intensive geological research programmes gathered momentum elsewhere, notably in Canada and Sweden and in
the member countries of the European Economic Community (11, 14, 153).

As a result, the scope of international studies was extended to include crystalline and argillaceous rocks, and it is now generally accepted that all three rock types (crystalline, argillaceous and saliferous) can possess a combination of attributes which favour their selection as repository host-media.

Saliferous rocks contain no circulating groundwater, are excellent conductors of heat, and can exhibit remarkable self-healing qualities. These factors suggest that they represent an ideal waste confining medium. However, their susceptibility to solutioning and their future mining potential are obvious drawbacks. Argillaceous rocks, in general, have several important confining qualities. Clay formations, in particular, have extremely low and comparatively uniform permeabilities with high physico-chemical sorption capacities. However, their heat dissipation properties are relatively poor and difficult construction problems may be anticipated at great depths in the unindurated plastic clays. Crystalline rocks are likely to pose few problems in terms of construction, are extremely stable and generally exhibit very low permeabilities, especially at great depth. However, the influence of discontinuities is dominant and the accuracy with which mass properties may be determined in practice is limited.

Table 5 provides an outline comparison of the most salient rock properties. However, broad comparisons of this type generally fail to convey the extreme variability which occurs in the characteristic properties of rocks, and the important differences
between the properties exhibited by small intact samples and those of the insitu rock mass. A more detailed appraisal of relevant characteristics and properties which takes account of these aspects is presented in Chapters 8, 9 and 10 below.
8. GEOLOGICAL AND GEOCHEMICAL CHARACTERISTICS

8.1 Geological Characteristics

Crystalline, argillaceous and saliferous rocks represent a very broad spectrum of geological environments, and the provision of a fully detailed account is beyond the scope of this thesis. However, in the radioactive waste disposal context, the geology of the host formation is relevant in assessing its long-term stability, and problems associated with repository construction. The mineralogical composition, geological history and lithological structure also provide a valuable insight into the significant containment properties of the materials under consideration.

8.1.1 Crystalline Rocks

The crystalline rocks under consideration are predominantly of the granitic type, formed by the crystallisation of alkali-rich magmatic intrusions within the earth's crust. The larger of these intrusive bodies cool slowly, allowing the dispersion of mineralogical concentrations and the development of a porphyritic crystalline structure. The resulting rocks are typically very strong, and exhibit a brittle, elastic behaviour at the depths under consideration. They are relatively homogeneous and are characterised by an interlocking crystalline fabric.

Gneisses (particularly granite gneisses) have also been considered. These differ in terms of mineralogy and structure, according to the nature of the parent formation. They are essentially anhydrous and are relatively brittle, yet they often retain non-annealed discontinuities, mineralogical banding and other
structural features which reflect the degree of re-working to which they were originally subjected. In consequence, many gneissic rocks exhibit considerable anisotropy.

The essential mineralogy of granites and granite gneisses comprises feldspar and quartz, with subsidiary mica or amphibole and various accessory minerals. Depending upon the relative proportions and grain-size distribution of these constituents, granitic rocks may be designated as true granite, adamellite, granodiorite, diorite, tonalite, etc. These distinctions are of little significance in terms of containment properties, although it is important to recognise the corresponding range of chemical compositions which may occur.

Granitic rocks can occur in a variety of geological structures, comprising large discordant bodies (plutons), smaller discordant and concordant sheets (dykes and sills) and lenticular bodies (laccoliths and phacoliths). In contrast, gneisses generally form basement complexes on a regional scale. For the purpose of radioactive waste disposal, the granitic plutons and gneissic complexes are generally regarded as the most favourable geological structures in view of their large size and relative homogeneity in the mass.

Figure 13a illustrates, in simplified form, a typical structural setting of a plutonic crystalline rock formation. In certain circumstances a site could be chosen where there is a considerable thickness of overburden materials overlying the host rock. These would typically comprise a sequence of sedimentary strata, disturbed and altered to some extent by the intrusion of the pluton. However, it is more likely that sites will be chosen in areas where the host
formation occurs at or near the surface, as illustrated. In these cases, the overburden would comprise an upper layer of weathered rock, which, in temperate European climates, would rarely exceed a depth of a few tens of metres.

Below the weathered zone, the fresh crystalline rock would be essentially unaltered and homogeneous. However, the rock mass would be characterised by the presence of discontinuities, usually occurring as three conjugate joint sets. These discontinuities would have varying persistence and would be generally sub-planar, dividing the rock mass into partially separated blocks. Towards the surface, an additional set of joints is usually found, lying more or less parallel to the ground surface. These are sheeting (stress-relief) joints, caused by the gradual removal of overburden through natural surface erosion.

The pattern and frequency of discontinuities plays a dominant role in determining the mechanical and hydrological properties of crystalline rock masses. In general, the available evidence suggests close-spaced discontinuity sets become less frequent, less persistent and may be less open with increasing depth and increasing distance from the lateral margins of the formation. Thus, in the central part of a pluton, the host rock is likely to be 'massive' rather than 'blocky' and sub-horizontal stress-relief joints are likely to be absent.

However, the characteristics of crystalline rocks can deviate in several ways from this general picture of structural and mineralogical homogeneity. Firstly, it must be noted that high-level granitic plutons are
highly discordant, with steeply dipping, sharply defined boundaries. Mineralisation and inclusions of country rocks are likely to occur in the vicinity of the intrusive contact zones. In addition, changes in the discontinuity pattern could possibly give rise to localised sub-vertical ground water flow paths.

In certain areas, heated water and/or steam may have been forced up from lower levels in the original magma; altering the feldspars in the upper levels to form kaolinite, with localised formation of a friable quartz/mica sand within a kaolinite matrix. Similar mineralogical changes are likely to occur on the surfaces of the more open joints, with the formation of kaolinite and possibly chlorite coatings (the latter being largely due to the chemical weathering of mica).

The occurrence of major structural discontinuities due to faulting cannot be discounted, and careful site selection is required to preclude the possibility of badly faulted zones. Where faults do occur, they are likely to have a strong influence upon groundwater flow. Conceptual repository design studies have generally assumed an absence of faults, although it is recognised that tectonic features may be present in the upper levels and around the lateral margins of granitic plutons.

Groundwater conditions in a large crystalline rock mass cannot be predicted on a non-site specific basis, since the mass permeability of the rock is governed almost exclusively by discontinuities; the intact rock itself being virtually impermeable. In consequence, it is generally assumed that the mass permeability of crystalline rocks decreases rapidly with depth, in proportion to the decrease in fracture porosity.
However, master joints, whose spacing may be measurable in tens of metres, are often found to remain open at depth and it would appear that these generally control the volume and velocity rates of deep groundwater flow.

The significance of discontinuities in relation to groundwater flow in crystalline rock formations cannot be over-emphasised. The evaluation of mass permeability based on the concept of an equivalent porous medium is apt to be misleading; since values assigned must incorporate a representative rock mass of sufficient size to account for all transmissive discontinuity sets. The presence of a transmissive fracture of one or two millimetres width implies high localised permeability; and larger-scale persistent fractures are even more significant. Where faults occur, or kaolinitised material has been removed by erosion, a potential exists for significant groundwater flow. Unfortunately, site investigation boreholes cannot guarantee success in delineating the full three-dimensional discontinuity pattern. Thus, the design of a repository should take full account of the potentially adverse effects of naturally-occurring and artificially-induced discontinuities; and the levels of uncertainty associated with current methods of assessing discontinuity data.

8.1.2 Argillaceous Rocks

Argillaceous soils and rocks are the most abundant group of sedimentary deposits within the earth's crust, and exhibit enormous mineralogical and physical variability. However, the term argillaceous simply denotes an appreciable content of clay minerals and it is the
presence of these which distinguishes the group as a whole.

In general, only the lacustrine or marine deposits are likely to include argillaceous strata of the required depth, thickness, lateral extent and relative homogeneity to be considered as potential repository host media. These comprise the more extensive basin deposits, in which continuous deposition over long periods of time and in predominantly low-energy environments has produced strata of 100m or more in thickness, with relatively high and uniform clay contents.

However, even for this relatively narrow range of deposits, the predictive models which describe their composition and engineering behaviour are highly complex. Variations occur in the chemical and physical weathering processes, rates and thicknesses of deposition, salinity and velocity of water during sedimentation, etc., all of which influence the mineralogical composition, texture, grain-size and lateral and vertical homogeneity of the original clay sediments. Post-depositional changes, including consolidation and burial diagenesis, give rise to an even wider spectrum of indurated materials which exhibit great variety in structure, composition and engineering properties.

Considerable attention has been focussed on unindurated clay formations, in view of their low permeability and relatively high sorptive capacity. However, due to their higher strength and superior heat transfer properties, indurated and slightly metamorphosed argillaceous deposits have also been considered (11).
The most important mechanism responsible for the production of clay minerals is hydrolysis. The initial leaching and dissolution of parent rocks results in the selective removal of elements in the order of their solubility, i.e. sodium, potassium, calcium, magnesium, iron, silica and aluminium; leaving a transitional residue of aluminosilicic acid. The latter breaks down and reacts with the residual metallic elements to form clay minerals, accompanied by the release of easily removable colloidal silica.

The types of clay minerals formed during this chemical weathering process depend fundamentally on the concentration of hydrogen (H\(^+\)) and metal (M\(^+\)) ions present. This in turn depends on the composition of the parent material, the rate of rainfall, and the permeability of the surface rocks. A relatively high concentration of M\(^+\) and SiO\(^4^-\) ions tends to form clay minerals such as illite and the smectites (e.g. montmorillonite); particularly where the parent rock is rich in Ca\(^{2+}\), Mg\(^{2+}\) and Fe\(^{2+}\), and the climate is semi-arid with a high rate of evaporation. The pH of the system under these conditions is generally greater than 7, and flocculation of SiO\(_2\) is a frequent occurrence. In contrast, a high concentration of H\(^+\) ions and Al(OH)\(_3\) tends to create minerals such as kaolinite; particularly if the pH is less than 7 (acidic), and the climate is humid, the drainage efficient and the leaching intense.

The bulk engineering properties of argillaceous materials are normally dominated by the clay fraction, and depend to a great extent upon the overall clay content, the moisture content and the surface properties of the individual particles. Clay minerals are comprised of minute, elongate, flaky particles, whose
principal dimension is generally less than $2\mu$. In consequence, their aggregate surface area is very large. The films of adsorbed water which envelope the particles are responsible for the apparent cohesion and plasticity of the bulk material.

The various empirical engineering relationships which are used to describe the combined influence of these factors are summarised in figure 14. Figure 14a shows the effects of moisture content reduction on a sediment which is initially in a state of fluid suspension. As the bulk volume of the material is decreased, it undergoes a series of transitions; from viscous fluid to plastic semi-solid, to friable solid. Finally, a stage is reached where further reduction in moisture content produces no further volume change. In this condition, the material has a relatively hard, brittle consistency; and, if crushed, would be reduced to powder form.

The values of moisture content which correspond to these transitions (consistency limits) are termed the liquid limit (LL), plastic limit (PL) and shrinkage limit (SL), respectively. As a rough guide, each transition point represents about a hundred-fold reduction in strength. Thus, the consistency limits provide an indication of the effects of moisture content variations on strength and deformation characteristics (127).

Most clays exist at an insitu moisture content between their liquid and plastic limits. They are, therefore, normally in a plastic condition and the corresponding range of moisture contents (LL - PL) is termed the plasticity index (PI) of the material. Figure 14b illustrates the standard plasticity classification for fined-grained soils. Potential clay host materials
would lie predominantly within the shaded envelope shown on the diagram.

The influence of clay content and clay mineralogy on the plasticity characteristics of a given argillaceous material are illustrated in figure 14c. In the diagrams, plasticity index is plotted against the amount of the clay fraction for a variety of argillaceous materials. Results for a particular deposit often plot on a straight line; as illustrated for a number of British clays remoulded with varying amounts of sand (127). The slope of the line is termed the activity of the clay, defined by the relationship:

\[
\text{Activity} = \frac{\text{Plasticity Index}}{\% \text{ by weight finer than } 2\mu}.
\]

Activity may be regarded as a measure of the influence of clay mineralogy upon the engineering properties of the material. Figure 14c also indicates the ranges of activity associated with the clay minerals Na-montmorillonite, illite and kaolinite. It is apparent that a relatively small increase in montmorillonite content, for example, would give rise to a relatively large increase in plasticity index, whereas a substantially larger content of illite minerals would be required to produce the same effect.

An important geochemical property of clay minerals is that of base-exchange. When certain varieties (notably montmorillonite) are in equilibrium with the solutions present in a slightly calcareous sediment, they absorb calcium ions to form a calcium clay. If a solution of sodium chloride is then allowed to percolate through the calcium clay, calcium ions are taken into solution and are replaced by sodium ions, thus forming a sodium clay.
Similar replacement by magnesium may also occur. However, reactions with acid solutions, such as water containing dissolved carbon dioxide may remove exchangeable bases, replacing them by hydrogen.

The absorptive and adsorptive properties of clays are significant factors in the selection of argillaceous materials as potential host media; since sorption is an important radionuclide retention mechanism (see Chapter 10.3 below). However, the various natural exchange reactions which occur during deposition also have an effect upon the intrinsic permeability and degree of flocculation of clays. Hydrogen clays, for example, are highly dispersed and relatively impermeable; whereas sodium clays may be partly flocculated. Calcium clays deposited from hard, fresh water, tend to be highly flocculated, and are therefore generally more permeable than either hydrogen or sodium clays.

Changes in pore water chemistry after deposition can also be highly significant. For example, the leaching of salts from the adsorbed layers of a marine clay deposit can give rise to substantial reduction in plasticity accompanied by a loss in strength; as evidenced by the 'quick' clays of Norway. For this reason a stable hydrological environment is a pre-requisite for site selection, and it is clear that the regional topography and depth of burial must be such that the risk of major hydrological changes (due to glaciation for example) is minimal.

In addition to the clay mineral content, a significant proportion of other solid constituents is likely to be present in clay deposits. These comprise finely pulverised rock-forming minerals such as quartz, feldspar and mica; collectively known as rock flour.
The constituents of rock flour are the assorted residual products of physical weathering and mechanical degradation during transport. Due to mechanical sorting processes which occur during the later stages of transport and deposition, their dominant particle-size is likely to lie within the silt range.

However, as previously noted, variations in current velocity during deposition may cause textural and compositional changes within the deposit. In general, high clay contents are likely to be indicative of low-energy depositional environments; although predominantly silty bands or partings of relatively low clay content may occur due to periodic increases in current velocity. It should be noted however, that silt-sized aggregations of clay minerals can often accumulate. Hence relatively homogeneous deposits may form, in which the coarser layers are not depleted in clay minerals.

These differences in composition and depositional environment clearly give rise to a wide variety of argillaceous sediments and, as noted above, the clay content and plasticity indices provide a useful outline guide to their engineering properties. However, the consolidation history of the sediment ultimately determines its consistency, strength and deformation characteristics.

Newly deposited sediments exist in a fluid condition, in which the individual clay particles are encased in sheaths of adsorbed water and are not in direct physical contact with each other. Such sediments are likely to have porosities approaching 90% in the near surface layers. As burial occurs under an accumulation of sediment, the larger pore spaces are reduced in size due
to the expulsion of free water, until the sediment reaches a plastic condition at a porosity of 75% or less. It is at this stage that the sediment attains the consistency of a soft clay.

Continued loading due to deposition produces further expulsion of water, and the void ratio of the material is reduced as adjacent grains are brought into closer contact. The pore spaces at this stage are microscopic, and are capable of imposing considerable throttling resistance to the expulsion of water under more rapidly applied loads. Due to the interaction of surface forces between adjacent clay particles, an apparent cohesive strength is developed which is dependent both on the plasticity of the material and the amount of consolidation which has occurred. Under a given effective overburden pressure, \( p_o \), the clay consolidates until a state of equilibrium is reached. In this condition, the overburden pressure at all points within the deposit is shared in such a way that the pore water pressure equals the hydrostatic head of water in the ground; i.e. all excess pore water pressures have dissipated.

Consolidation theory is one of the fundamental aspects of the science of soil mechanics and it is not intended here to provide a rigorous account of the various descriptive models which have been developed (127). However, since the consolidation history of clay sediments provides the key to an understanding of the factors which influence their engineering performance, it is necessary to provide a broad outline of relevant factors. The general principles involved are illustrated in idealised form in figure 15.

Figure 15a shows the idealised virgin consolidation
curve for a young argillaceous sediment. As shown, the void ratio decreases in an approximately linear fashion with the logarithm of the effective overburden pressure. However, when the material reaches a pore pressure equilibrium at a given overburden pressure, $p_o$, further time-dependent compression occurs at constant effective stress. This process has been termed 'pseudo-consolidation' or 'delayed compression' and is thought to be associated with the mechanical re-adjustment of clay particles which have come into direct physical contact.

In consequence, the ageing of the deposit results in a reduction of void ratio (from $e_o$ to $e_1$). As shown, the shear strength, $C_u$, also exhibits a unique relationship with void ratio. Ageing therefore results in an increase in shear strength from $C_u_0$ to $C_u_1$. The compressibility of the clay is also reduced and any subsequent application of increased load (in the range $p_o$ to $p_c$) produces an essentially elastic response. However, as the applied pressure approaches the value $p_c$ on the projected virgin consolidation curve, a yield point is reached beyond which the material reverts to plastic behaviour. Thus, the idealised theoretical concept predicts that for a clay deposit at equilibrium under the weight of accumulated overburden; strength and deformation characteristics are uniquely defined by its age and the distribution of effective pressure with depth.

The foregoing commentary describes the generalised, stress-time - strength relationships for normally-consolidated clays; i.e. clays which exist at the maximum effective overburden pressure sustained during their formation. However, other post-depositional changes can cause significant
anomalies; and in clay host formations suitable for repository construction, these are likely to be the rule rather than the exception.

Figure 15b illustrates the effect of overburden removal on a 'normally-consolidated' clay. For a clay originally consolidated under an effective overburden pressure $p_c$ and void ratio $e_1$, overburden removal by erosion reduces the pressure to a value $p_0$. This is accompanied by a swelling process which increases the void ratio to a value $e_2$. The clay therefore exhibits an apparently anomalous shear strength $C_u$, and is said to be over-consolidated. Re-loading (in the stress range $p_0$ to $p_c$) produces an initial elastic response and subsequent transition to plastic behaviour, as previously described.

Most clays therefore exhibit an increase in shear strength and a decrease in void ratio and compressibility with increasing depth; the general nature of the relationship depending on the age and stress history of the deposit. The plasticity of the material (which reflects the mineralogy and clay content) largely determines the distinctive relationships which apply to a particular clay.

Figure 15c shows an empirical relationship between shear strength, plasticity and effective overburden pressure which is found to hold true for relatively young, normally-consolidated clays at moderate depth. For pseudo-consolidated and over-consolidated clays, the relationship may also provide an outline indication of shear strength provided that the appropriate values of $p_c$ are substituted for $p_0$. However, the validity of such predictions are not proven, and may be subject to gross error when applied to the more heavily
over-consolidated clays.

Additional factors which influence the characteristics of older clay deposits at depth are:

- cementation and incipient diagenesis due to chemical deposition within the pore spaces of the material;
- fissuring due to stress relief in over-consolidated clays;

The process of consolidation in clays is seldom responsible for reduction in void ratios much below 0.3 (porosity = 25%); at which stage the moisture content of the material will generally lie below the plastic limit. Since repository construction is unlikely to be contemplated at depths less than 200m below ground level, it may be assumed that all clay host materials will have an in situ moisture content close to or less than the plastic limit.

Any further reduction in pore volume is associated with significant diagenetic changes, including re-crystallisation or 'pressure welding' at inter-particle contacts. The latter may be regarded as the most significant factor which distinguishes the argillaceous rocks from the unindurated plastic clays, since chemical diagenetic processes are undoubtedly initiated at a much earlier stage (83).

All argillaceous type rocks are formed as a result of the induration of clay deposits due to burial diagenesis. The most satisfactory basis for description involves a combination of particle-size, fissility and degree of metamorphism. A simplified classification system suitable for present purposes is outlined in

121
The typical range of compositions representative of potentially suitable argillaceous host rocks is given below (218):

<table>
<thead>
<tr>
<th>Mineral</th>
<th>Range</th>
</tr>
</thead>
<tbody>
<tr>
<td>Clay minerals</td>
<td>45% - 60%</td>
</tr>
<tr>
<td>Quartz</td>
<td>20% - 30%</td>
</tr>
<tr>
<td>Feldspar</td>
<td>4% - 11%</td>
</tr>
<tr>
<td>Iron oxides</td>
<td>1% - 5%</td>
</tr>
<tr>
<td>Carbonates</td>
<td>4% - 9%</td>
</tr>
<tr>
<td>Other minerals</td>
<td>2% - 3%</td>
</tr>
<tr>
<td>Organics</td>
<td>&lt;1%</td>
</tr>
</tbody>
</table>

These reflect variations in origin and depositional environment coupled with geochemical changes associated with the diagenetic process.

The inter-particle re-crystallisation (pressure-welding) which accompanies burial diagenesis may reduce rock porosities to values as low as 15%. This process is generally accompanied by an increase in true cohesion, due to chemical cementation and some particle re-crystallisation, to form sub-horizontal cleavage planes. Subsequent metamorphic changes can lead to the formation of slates and schistose rocks and, over geological periods, the process of granitisation may complete the cycle of clay formation (106).

Those rocks with a predominant silt content, or rocks which have been subjected to significant metamorphic change (as opposed to low-temperature, low-pressure diagenesis), will exhibit relatively low sorptive capacities and will tend to exhibit a brittle behaviour; e.g. silt slates, argillites, slates and marls. The
stronger rocks of this type may be regarded as generically similar to the crystalline rocks, in terms of their containment properties (see table 5); although they will be generally weaker and less extensive in the vertical sense.

Grain orientation and the degree of cementation during diagenesis determine the cleavage potential of argillaceous rocks. Thus, shales are characterised by a high cleavage potential, sub-horizontal particle orientation and little cementation. However, all argillaceous rocks are massive under overburden pressure. As overburden is removed, exfoliation occurs and parallel cleavage develops. Argillaceous rocks containing swelling clay minerals generally do not develop fissile characteristics until they have lost appreciable water by dehydration. Nevertheless, they exhibit varying degrees of physical instability when subject to stress release, particularly where water is present; since softening and slaking may occur. These factors are clearly of considerable importance in relation to repository design and construction and the performance criteria for backfilling and sealing systems.

As noted above, the preferential orientation of platey-shaped mineral grains implies a considerable degree of structural anisotropy in shales. However, well-defined horizontal bedding planes are likely to occur in all argillaceous rocks, due to local variations in grain size associated with changes in depositional conditions. Not only is there a likelihood of distinctly different layers occurring within a single argillaceous formation; but lateral gradations or facies changes can also occur, albeit more gradually. Thus, the degree of homogeneity of the formation is a
site-specific factor which could ultimately influence the choice of a particular host rock.

The overburden profile overlying an argillaceous host formation could include virtually all types of sedimentary rocks, from coarse-grained clastic varieties (conglomerates and sandstones) to finer-grained siltstones, mudstones and shales, calcareous rocks (limestones and dolomites) and carbonaceous types (lignite and coal). Whilst it is impossible to predict a typical overburden sequence, beds of sandstone, slate and clay are probable, with the possible occurrence of any other type of sedimentary strata (see figure 13b).

An argillaceous host formation would necessarily comprise a relatively thick and uniform sequence. A high clay content and high plasticity are also desirable qualities since these imply low permeability, high capacity for radionuclide retention by ionic exchange processes, and a tendency for the self-healing of induced fractures. Although true marls are usually associated with high clay contents (low energy depositional environments), their relatively high degree of cementation imparts a brittle tendency which may render them unsuitable.

Due to the wide range of possible overburden materials, groundwater conditions can only be described on a site-specific basis. However, potentially suitable host formations will be characterised by extremely low values of mass permeability. It is important to note that even after consolidation to a rock-like consistency, some argillaceous materials may have a moisture content in excess of 20%. Nevertheless, because of the fine grain-size of the clay fraction, permeabilities are low; especially at depth, where effective (intergranular)
stresses are considerable. The permeabilities of the harder (relatively brittle) argillaceous rocks are likely to be governed by the nature and pattern of discontinuities, as in the case of crystalline rocks. However, in argillaceous rocks the distribution of discontinuities tends to be more predictable, since they are normally related to the bedding. Nevertheless, evaluations of mass permeabilities may be subject to significant margins of uncertainty.

At higher levels, above the host formation, water-bearing permeable strata are likely to occur. These may contain water under artesian or sub-artesian pressure and are likely to be sandwiched between argillaceous or other rock types, which behave as aquitards or aquicludes. Shaft construction may therefore present considerable problems, requiring the use of special methods to seal or stabilise areas of potentially difficult ground.

8.1.3 Saliferous Rock

Saliferous rocks are formed as a result of the chemical precipitation of salts from super-saturated brines within marine or lacustrine basins, in which the overall rate of evaporation exceeds the rate of water replenishment. The principal salts deposited are chlorides, sulphates and carbonates, and the corresponding rock types are characterised in terms of their prevailing mineralogy, as follows:

- Halite or rocksalt (NaCl)
- Anhydrite (CaSO₄)
- Gypsum (CaSO₄ ₂H₂O)
- Calcite (CaCO₃)
Under appropriate conditions, considerable thicknesses of these evaporites can accumulate. With continued precipitation, the basin subsides, allowing further influx of saline water to maintain the depositional cycle. Such conditions exist today in the Caspian and Dead Sea basins (106).

With increasing depths of burial, the lower layers are gradually transformed into more homogeneous beds by a process of re-crystallisation. The fabric of the resulting material is generally dense and micro-crystalline in texture. Complex depositional sequences may also arise, reflecting changes in the chemistry of the original brines. Evaporite sequences may also be separated by varying thicknesses of argillaceous and calcareous rocks, due to changes in depositional environment.

Thus, the thickness and composition of bedded saliferous deposits is highly variable. However, as an example, some of the salt deposits in Germany are over 1200m thick, of which approximately 80% are saliferous rocks.

Formations of this type are termed bedded salt deposits (see figure 13c). However, major changes in the depositional environment may result in their deep burial below other weak sediments. Under such conditions the saliferous rocks may ultimately become mobile, leading to the formation of a salt dome or salt plug which gradually rises vertically towards the surface, displacing the overlying ground materials.

This phenomenon, termed diapirism, is partly attributable to buoyancy (saliferous rocks are typically
about 10% less dense than other rock types). The inherent plasticity of the materials, particularly under large confining pressures, assists the process and enables upward flow to occur from a parent deposit at great depth. However, this mobility is gradually decreased as overlying materials are displaced, and the dome eventually attains a state of equilibrium, with its upper margins at or near the surface.

Figures 13c and 13d illustrate the two distinct types of geological setting which typify saliferous formations. In the case of bedded deposits, any combination of rock types and layer thicknesses may occur within the upper part of the ground profile. However, certain rocks are likely to dominate, notably marls and dolomitic limestones.

Where salt domes occur, circulating groundwater attacks the upper layers by dissolution. Ultimately, a resistant 'caprock' of gypsum and other relatively insoluble minerals is formed, which effectively seals the dome from further groundwater attack. At greater depths, salt domes are less affected by solution and therefore no 'caprock' develops. Overlying sediments may be of any type, although marls, limestones and clays are particularly common.

Rapid variations in mineralogy are possible in bedded deposits since the various salt fractions in a particular brine are precipitated sequentially. Larger scale variations are associated with major changes in brine composition. However, these are likely to be partly obscured by diapirism, since the movement and re-crystallisation which accompanies plastic flow results in a degree of mechanical and chemical re-distribution. Salt domes are therefore generally
more homogeneous, in terms of structure and mineralogy, than bedded salt deposits.

The hydrological characteristics of saliferous formations are closely linked with the geological and mechanical characteristics outlined above. Their micro-crystalline texture and inherent plasticity results in structures which are generally free from discontinuities. Although the material may exhibit a brittle behaviour under rapidly applied loads, subsequent time-dependent plastic flow can result in the healing of discontinuities. In consequence, the intrinsically low permeability of the material, as determined in laboratory and borehole measurements, is often also representative of the rock mass permeability.

Since saliferous rocks are highly soluble, their existence implies an absence of circulating groundwater, and in this respect they represent an ideal host environment. However static groundwater may be present within the host formation in the form of 'connate' water entrapped at the time of formation. The connate water may be present in the form of discrete 'bubbles' or 'pockets' of saturated brine and typically comprises, less than 0.5% when expressed as a total moisture content (31).

Where heat-generating wastes are concerned, research has shown that some localised brine pockets may migrate towards the waste units (31). Accumulation of brine around the waste units would increase the rate of corrosion and would reduce the rate of transfer of heat to the surrounding host rock. A similar situation may arise due to the presence of water entrapped within isolated lenses of non-saliferous materials which may form part of the sequence.
However, any entrapped water is likely to be widely distributed and only small volumes are likely to occur within the zone of influence. This water therefore poses few real difficulties in relation to waste containment. The principal concern is the possibility of major water inflow from adjacent or overlying water-bearing materials and consequent solution effects within the host rock. In the case of bedded deposits, the principal aquifers occur within the overburden and below the formation. In salt domes, water-bearing strata are likely to be present both in the overburden and the surrounding country rocks which have been truncated by the salt mass. It is therefore essential that the bulk excavations within the repository are effectively sealed off from shafts and other penetrations which could lead to inundation and progressive dissolution by circulating groundwater.

8.2 Geochemical Characteristics

The geochemical characteristics of the host rock or, more accurately, the groundwater, will clearly have a considerable influence upon the rate of deterioration of waste unit claddings and the subsequent leaching and migration of radionuclides. The identification of dissolved species within natural groundwater together with a characterisation of potential reaction fields provides a useful means of assessment.

The dissolved species occurring in deep groundwaters are determined by:

- initial exchange reactions occurring within the atmosphere

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o equilibrium and exchange reactions subsequently occurring between the infiltrating waters and rock minerals

o possible additions of saline (connate) water not in equilibrium with other parts of the system

Hydrogen ions are present in infiltrating ground-water due to the dissociation of carbonic acid derived from the dissolution of atmospheric CO₂, according to the equation:

\[ \text{CO}_2 + \text{H}_2\text{O} \rightarrow \text{H}^+ + \text{HCO}_3^- \]  \hspace{1cm} (1)

However, with further infiltration the hydrogen ion content is depleted, due to chemical reactions with the host rock (119). Examples include:

weathering of sodium and potassium feldspars to form kaolinite:

\[ 2\text{NaAlSi}_3\text{O}_8 + 2\text{H}^+ + \text{H}_2\text{O} \rightarrow 2\text{Na}^+ + \text{Al}_2\text{Si}_2\text{O}_5(\text{OH})_4 + 4\text{SiO}_2 \]  \hspace{1cm} (2)

\[ 2\text{KAlSi}_3\text{O}_8 + 2\text{H}^+ + \text{H}_2\text{O} \rightarrow 2\text{K}^+ + \text{Al}_2\text{Si}_2\text{O}_5(\text{OH})_4 + 4\text{SiO}_2 \]  \hspace{1cm} (3)

weathering of orthopyroxene to release iron and magnesium:

\[ \text{MgFeSi}_2\text{O}_6 + 4\text{H}^+ + \text{H}_2\text{O} \rightarrow \text{Mg}^{2+} + \text{Fe}^{2+} + 2\text{SiO}_2 + 3\text{H}_2\text{O} \]  \hspace{1cm} (4)

As a result of these and similar reactions, a series of
cations including Na\(^+\), K\(^+\), Mg\(^{2+}\), Fe\(^{2+}\) and Ca\(^{2+}\) are released in association with the anion HCO\(_3\)\(^-\). Chemical weathering therefore produces a carbonate water containing a variety of cationic species which broadly reflects the mineralogical composition of the parent rocks (119).

The cation population also influences the oxygen content of groundwater. The release of Fe\(^{2+}\) by weathering of ferro-magnesian and other minerals, as for example in reaction 4 above, leads to the oxidation of the Fe\(^{2+}\) ions to the ferric state, in the form of the insoluble hydroxide Fe(OH)\(_3\), according to the reaction:

\[
\text{Fe}_{2+} + 3\text{H}_2\text{O} \rightarrow \text{Fe(OH)}_3 + 3\text{H}^+ \quad \text{............... (5)}
\]

Similar behaviour applies to other metal cations which have variable oxidation states (principally the transition elements; see figure 3). Measurements on groundwater samples from deep crystalline and argillaceous rocks are reported to show increasing Fe\(^{2+}\) content with increasing depth, which is indicative of oxygen depletion and strongly reducing conditions at depths of the order of 1000m (5).

Weathering processes such as those described in reactions 2 to 4 above also result in the production of dissolved silica. Groundwaters in crystalline and argillaceous rocks are therefore likely to have appreciable silica contents, although the saturation concentration tends to be established rapidly so that little variation occurs with depth.

Apart from hydrogen and oxygen, other anions in natural groundwaters include chloride, sulphate, sulphide, fluoride, nitrate and phosphate. In crystalline and
argillaceous rocks these anions are derived principally from the atmosphere.

In general, chloride and sulphate contents are likely to be low, except where relict sea water is encountered. Sulphides may occur by biogenic reduction of sulphates in the root soil and humic substances in deeper groundwaters. Fluoride may be present as a derivative of a variety of minerals, but in granite environments is most readily produced by the dissolution of fluor spar (CaF$_2$). Nitrate and phosphate contents are likely to vary within wide limits, and may be influenced by the root soil composition at the time of infiltration.

The foregoing commentary describes the primary factors which influence the chemical composition of groundwaters at depth and indicates, in a general way, how the solutes in groundwater can vary systematically according to the mineralogy of the host rock. The factors outlined generally result in a relatively limited range of groundwater compositions within crystalline rock formations.

In argillaceous host rocks the potential variation is much wider. In particular, porewaters in argillaceous deposits of marine origin are likely to contain relatively high sodium, calcium and chloride contents. Isolated connate water may also undergo sulphate reduction to produce a sulphide-rich environment with a high pH. This process may result in the diagenetic transport of silica, which has been observed to increase its solubility above a pH of 9 (160). The latter has obvious implications for the leaching of borosilicate glass, whilst the higher chloride contents may enhance rates of metal corrosion in waste unit claddings.
Saliferous rocks are a special case, since circulating water is absent; although brine inclusions may be present as discrete pockets or inter-crystalline fluids, as described in Chapter 7.2.3 above, up to about 0.5% when expressed as a total moisture content. These fluids will be saturated solutions of salts whose composition will reflect that of the host material. Since they are not 'free' circulating groundwaters, their high corrosion and leaching potential is of little consequence in relation to waste disposal, provided that through-flow conditions do not develop as a result of repository construction.

Clearly, actual groundwater compositions vary considerably according to the nature and sequence of the geological profile and the presence or absence of connate fluids. The compositions shown in table 7 illustrate the ranges and relative proportions of dissolved species, representative of each host rock type, which have been assumed during corrosion studies undertaken within the European Community's research programme on radioactive waste disposal.

The groundwater compositions shown in table 7 indicate the availability of reactants within the host rock environments under review. However, it is also necessary to establish the conditions which are likely to govern the equilibria between groundwater constituents, and thence to predict the chemical effects of waste emplacement. The concept of Eh/pH stability fields is found helpful in this respect, as shown by Brookins in his evaluation of chemical equilibria within the ground materials in the vicinity of the Oklo Natural Reactor (32).

The oxidation potential, \( \text{Eh} (\text{mV}) \), is a measure of the
ability of an environment to transfer electrons through redox (oxidation-reduction) reactions. A system capable of donating electrons is termed a reducing environment, whilst one which accepts electrons is termed an oxidising environment. The standard hydrogen electrode is designated as having an Eh equal to zero. Any environment more oxidising than this datum is assigned a positive Eh, whilst a less oxidising environment is assigned a negative Eh.

\[ pH = - \log_{10} [H^+] \]

where the square brackets indicate concentrations in mol/l. On the logarithmic pH scale, a neutral pure water environment has a pH of 7. Values of pH below 7 (decreasing hydrogen ion concentration) are increasingly acidic, whilst pH values above 7 indicate increasing alkalinity.

The hydrogen ion depletion with increasing depth associated with mineral weathering reactions, generally results in decreasing pH (increasing alkalinity). Similarly, the oxygen depletion associated with the oxidation of multi-valent cations such as Fe\(^{2+}\), generally results in increasing negative Eh values (reducing conditions) with increasing depth.

Figure 16a shows a generalised plot of Eh versus pH for natural ground waters. The two dashed lines define the range of Eh and pH values which are theoretically possible. Above the upper line, water is oxidised to liberate free oxygen and below the lower line water is
reduced to liberate hydrogen (160). Since natural groundwaters do not liberate either gas, all groundwater Eh/pH regimes must lie within the limits shown. Superimposed upon the diagram are the broad stability fields for waters in a variety of chemical environments. Those typical of mine waters, deep groundwaters and saline waters are obviously of greatest relevance in the context of the repository environment.

The potential application of Eh/pH stability field diagrams is illustrated in figures 16b and 16c. The diagrams illustrate some of the stable forms of iron and uranium in groundwater, as a function of the prevailing Eh/pH. Since these elements may be introduced into the repository in the form of steel claddings for waste units (Fe) and as radionuclides (U), the relevance of the diagrams is self-evident. The various forms of iron and uranium, as shown, exhibit different solubilities and ionic exchange potentials. The elements may therefore be transported readily by circulating groundwater, form insoluble precipitates, or form ionic associations with other species in the system, according to the prevailing Eh/pH regime and the concentrations of other reactants.

These examples illustrate how the Eh/pH field which characterises a particular groundwater environment, together with the concentrations of chemical species, determine the equilibria attained within the aqueous system. Allard, et al, have utilised the Eh/pH concept to predict the chemical stabilities of various species within a granitic repository (5, 6). Based on the equilibria between redox couples Fe$^{3+}$/Fe$^{2+}$, they suggest that Eh/pH relationship for groundwaters in deep-level granites is represented by:

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\[ \text{Eh} = 0.21 + 0.06 \text{pH} \]  \hspace{1cm} (5)

from which it may be predicted, for example, that uranium leached from the wastes is likely to occur in the relatively insoluble tetravalent state, Uraninite (71). \text{Eh/pH} relationships in argillaceous formations also tend to lie within specific, though wider, boundaries, enabling comparable predictions to be made.
9. CONTAINMENT PROPERTIES

9.1 Permeability

Low permeability is the most fundamental requirement for the prevention of radionuclide release and migration. Restriction of groundwater availability serves to reduce the rate of corrosion of waste unit claddings and subsequent leaching of radionuclides. Restriction of groundwater velocities and volumetric flow rates reduces the first arrival times and peak dose rates for radionuclides transported in groundwater.

The range of representative mass permeabilities of naturally-occurring rocks and soils varies by more than twelve orders of magnitude, as shown in figure 17. For most practical civil engineering purposes, ground materials having permeabilities lower than about $10^{-9}$ m/sec are considered impermeable. However, as described in Chapter 6.3, the slow migration of radionuclides by transport in groundwater is recognised as the most likely means by which radionuclides could return to the biosphere; and in view of the time-scale of the containment problem, small but finite mass permeabilities must be regarded as significant.

As shown in table 5 and figure 17, repository host rocks occur at the extreme lower limit of permeability for ground materials. However, it must be noted that, as in the case of rock mass strength properties, determinations of representative insitu mass permeabilities are subject to significant uncertainty.

For the permeability range of interest in the water supply, civil engineering and mining industries, a measurement accuracy within one order of magnitude is
generally achievable using currently available methods. However, insitu measurements on materials with permeabilities significantly less than $10^{-9}$ m/sec, especially at great depth, are fraught with difficulties (135). For example, minute leakages past a borehole packer can cause gross errors during pumping tests, and fissures which accept flow may be created or modified by the presence of the test borehole and the application of fluid pumping pressure. Hence, for extremely low permeability rocks it is unlikely that insitu measurements can attain an accuracy better than 2 orders of magnitude.

However, for jointed rocks further problems are encountered in attempting to extrapolate the results of borehole measurements to the rock mass as a whole, since permeability values are largely determined by the properties and distribution of discontinuities rather than the permeability of the intact material.

Figure 18 illustrates the influence of scale effects in the assessment of mass properties of fissured or jointed rock masses. It is evident that extrapolation of results from tests in a limited number of boreholes orientated vertically or sub-vertically cannot take full account of the mass properties of anisotropically jointed rock masses on a large scale.

The fundamental distinction between porous flow and fissure flow is also of vital importance in assessing the containment properties of repository host materials. The rate of flow through porous rocks (rocks in which the pore volume is more or less evenly distributed in the form of small inter-connected voids) is governed by Darcy's Law which may be expressed:
\[ Q = A k i \] ................................. (1)

where \( Q \) is the volume of water flowing per unit time
\( A \) is the gross cross-sectional area of the material normal to the direction of flow
\( k \) is the permeability of the material to water
\( i \) is the hydraulic gradient

Equation (1) may be re-written as:

\[ \frac{Q}{A} = k i \] ................................. (2)

in which \( \frac{Q}{A} \) represents a hypothetical Darcy velocity of flow across the gross cross-sectional area \( A \). However, this underestimates the true velocity, since the water only flows through the transmissive pore spaces. Hence, the true velocity is given by:

\[ V_p = \frac{k i}{n_p} \] ................................. (3)

Where \( n_p \) is the effective porosity, defined as the ratio of the total volume of transmissive pore space per unit volume of the material.

In saliferous rocks and clays, the minute pore spaces are widely distributed throughout the material. Hence accurate permeability measurements made on a sufficiently large number of relatively small intact samples should provide a fairly reliable indication of the in situ value. However, anisotropy must be taken into account. Where a preferential orientation of particles occurs, the tortuosity of flow is greatest normal to the larger grain dimension. In consequence lateral
permeabilities in bedded deposits tend to be considerably higher than those in the vertical direction.

However, where fissure flow dominates, as in crystalline and hard brittle argillaceous rocks, the Darcy approach is less appropriate. Whilst the permeability of intact samples may be very low, discrete discontinuities or discontinuity sets have considerable influence upon flow within the ground mass. Figure 19a represents an idealised fissure of width \( b \), bounded by smooth parallel walls. For such a model, Snow (197) has shown that solution of the Navier—Stokes equations for fluid flow under appropriate boundary conditions gives:

\[
v_f = \frac{b^2 \rho g}{12 \eta} \tag{4}
\]

where \( v_f \) is the true longitudinal flow velocity in the fissure;
\( g \) is the acceleration due to gravity;
\( \eta \) is the fluid viscosity.

The above expression may be used as a basis for assessing the rate of flow within a single fissure, noting that in practice the expression must be multiplied by some empirical coefficient to account for friction losses.

Figure 19b represents an analogous series of parallel fissures of average width \( b \), intersecting a unit cross-sectional area normal to the flow direction. The number of fissures within the area under consideration is equal to the discontinuity frequency per unit length \( \lambda \), measured perpendicular to the flow line. In this model, the true mean flow velocity in each fissure is
equal to the value $v_f$ given by equation 4. Hence the total volumetric flow rate $Q$ is given by:

$$Q = \frac{b^2 \rho g \sigma (b \lambda)}{12 \eta} \cdot i$$

or

$$Q = \frac{b^3 \rho g \lambda^2 \cdot i}{12 \eta} \qquad (5)$$

Comparison of equation 5 with equation 1 shows that the permeability of the fissure model expressed in terms of an equivalent porous medium, is given by:

$$k_{equiv} = \frac{b^3 \rho g \lambda}{12 \eta} \qquad (6)$$

Equations 4 and 6 indicate the potential significance of individual fissures or fissure systems upon mass groundwater flow. The true mean flow velocity varies according to the square of the mean fissure width, whilst the rock mass permeability (measured and evaluated on the assumption that the material behaves as a porous medium) is directly proportional to the mean fracture frequency and the cube of the mean fissure width. The practical validity of this cubic permeability law has been verified by Witherspoon et al for a wide range of fissure widths and rock types (222).

The significance of the above in relation to real ground masses may be assessed by comparing the fracture flow equations with the corresponding relationships for porous media. It is particularly instructive to compare the flow rates predicted by the fracture flow model and the equivalent porous medium.

The true mean velocity in a fissure system is predicted
by equation 4. Introducing a coefficient \( c \) to account for friction losses, etc.

\[
v_f = c \cdot \frac{b^2 \rho g}{12 \eta} . i 
\] ........................ (7)

Now consider an equivalent porous medium having a permeability equal to that given in equation 6. The hypothetical Darcy flow through the material is obtained by substituting for \( k \) in equation 2, giving:

\[
q = \frac{b^3 \rho g \lambda}{12 \eta} . i 
\] ........................ (8)

Thus, for a porous material of identical mass-permeability, having an effective porosity \( n \); the true velocity \( v_p \) is given, according to equation 3, as:

\[
v_p = \frac{b^3 \rho g \lambda}{12 \eta n} . i 
\] ........................ (9)

Hence, combining equations 7 and 9:

\[
\frac{v_f}{v_p} = c \cdot \frac{n}{b \lambda} 
\] ........................ (10)

or \[
\frac{v_f}{v_p} = c \cdot \frac{n}{n_f} 
\] ........................ (11)

since fracture porosity, \( n_f = \lambda b \)

Based on the above, a porous medium with a permeability \( k = 10^{-8} \text{ m/sec} \) is equivalent to a single fissure of width 20 \( \mu \text{m} \). Assuming the effective porosity of the porous medium is, say, 15%, then:
\[
\frac{v_f}{v_p} = c. \frac{0.15}{2 \times 10^5} = 7500 \cdot c
\]

Assuming \( c = 0.5 \), then the predicted velocity of flow in the fissure is 3750 times that in the equivalent porous medium. Runchal and Maini (190) have derived a similar, though rather more favourable comparison. The example shows that velocities (and hence travel times) predicted by fracture flow models may be more than 3 orders of magnitude greater than those based on an equivalent porous medium approach.

The above commentary indicates the importance of adopting an approach to ground water flow evaluation which is consistent with the characteristics of the ground mass under consideration. Clearly, the porous medium approach is most appropriate for clays, massive argillaceous rocks and un lithified overburden materials; provided that due care is taken in the measurement and evaluation of representative permeability values.

For jointed crystalline and argillaceous rocks, assessments should ideally be based on a fracture flow model. However, in practice, translation from idealised fracture flow models to the real ground mass presents apparently insurmountable problems. These are associated with the definition of rock fracture geometry on a megascopic scale, as illustrated in figure 18.

Hudson and Priest have shown that three-dimensional fracture frequency distribution may be predicted using statistical 'scan-line' theories, based on data from orientated boreholes, surface exposures and exploratory excavations (108). Given adequate joint trace data, it may also be possible to extend this approach to study the interconnection of discontinuities along a given
plane through the rock mass, and hence to develop a
description of two-dimensional flow paths (109). The
latter would appear to be a valuable analytical tool
provided that adequate joint trace data are available.
Clearly, however, the statistical persistence and joint
width frequency distributions must first be defined for
each joint set and subsequently related to the remainder
in order to provide a reliable model.

In view of the above, it must be recognised that the
very comprehensive data required to develop a reliable
prediction of the three-dimensional stochastic geometry
of rock mass discontinuities at great depth cannot be
obtained in practice, due to the limitations of currently
available exploratory techniques. Moreover, the number
of boreholes which can be carried out without seriously
jeopardising the containment properties of the host rock
is a further limiting factor.

Thus, application of the more fundamental theories will
inevitably be limited by inadequacies in the available
data base, and the 'equivalent porous medium' approach
probably represents the best available method for
describing ground water flow through jointed host rock
masses at present.

Long, Remer et al (134) have shown that permeabilities
expressed in terms of an equivalent porous medium must
be related to a 'representative equivalent volume' (REV)
of rock, in which the fissure population is sufficient to
develop Darcian flow characteristics. However, no
equivalent porous medium permeability will exist when
the REV exceeds the available rock volume or where
transmissive fissures are insufficiently dense. In
consequence, great caution is required in the assessment
of insitu permeabilities and mass porosities in
fractured rocks, and published data must be regarded with circumspection.

The intact permeability of unweathered material is likely to be very small and hence, where the length of test surfaces is close to or less than the mean fracture spacing, insitu measurements will not be representative. It is clear that unless insitu test measurements take account of a representative number of the larger-scale discontinuities, the derived values of equivalent mass permeability will not be representative of the actual rock mass under consideration.

Major features, such as faults and large joints should therefore be accounted for in terms of a superimposed fracture flow analysis, in recognition of their potentially disproportionate influence. Some form of assessment of risk associated with the presence of undetected transmissive features in the host rock should also form part of any repository performance assessment.

Unlike the crystalline and hard argillaceous rocks which have formed the basis for the foregoing discussion, saliferous rocks contain no circulating groundwater under normal circumstances. They are essentially fracture-free, and porous flow may be postulated only on the basis of inter-crystalline fluid movement. Due to their solubility in water and low intrinsic permeability, test measurements on intact samples are generally made using gas as a permeant (135). However, the permeability of the intact rock to groundwater flow is of little relevance; since in the event of a breach in the formation and inundation by groundwater, solution cavities would develop. In this case, the flow theories outlined above are generally inappropriate and groundwater invasion must be described in terms of a
9.2 Physico-Chemical Retention

In the absence of other mechanisms, the rate of radionuclide transport through the host rock would depend solely upon the rate of groundwater flow. However, evidence has shown that natural ground materials are able to retard certain radionuclides, so that their rates of migration are less than those associated with hydraulic movement (217). Three retention mechanisms may be identified, namely:

- diffusive retention, in which dissolved species diffuse into discontinuous fissures or channels within the solid;
- sorption reactions, involving absorptive and adsorptive temporary retention of ions held in solution;
- precipitation reactions, in which dissolved species undergo irreversible chemical reactions with the solid.

The relative significance of each of the above will depend upon the physical structure and mineralogical composition of the host rock. Relevant criteria are outlined below:

(a) diffusive retention;

Diffusive retention occurs where the flow of the solution is concentrated in transmissive discontinuities within a relatively impermeable matrix. In crystalline rocks, the diffusion of...
radionuclides could occur into non-transmissive micro-cracks or grain boundaries adjacent to the principal flow channels. Similarly, in strong, fissured argillaceous rocks, transport of radionuclides by fissure flow may be retarded by transverse diffusion into the porous mass; (see figure 20). In rocks where porous flow dominates, including certain clays and massive mudstones, diffusive retention may also occur due to the presence of occluded pore structures (usually associated with partly-cemented materials).

Diffusive retention mechanisms depend on varying concentration fields within the fluid, and would therefore not be expected to display an affinity for any particular species of radionuclide other than by virtue of their relative concentrations in the wastes or variations in rates of leaching from the immobilising medium. Unfortunately, little insitu data is available for fractured rocks. Since the extent of diffusive retention is entirely dependent on the hydraulic properties (physical structure) of the host, its importance as a natural retention mechanism is likely to be highly site-specific (115).

Diffusive retention may also occur on a molecular scale; where the lattice of the solid contains ports or channels whose dimensions are similar to those of the dissolved ionic species or suspended colloids (micelles). Diffusion of this type is termed 'molecular sieving' and is known to occur in clays and certain layered silicates, such as zeolites (113).

Diffusive retention is an important mechanism, since
it is less dependent on Eh, pH and temperature than other processes. Furthermore, in the absence of reverse hydraulic gradients, its effects may be long-lasting. However, for insitu host media, it is difficult to assess the extent to which retention may occur due to diffusive phenomena as opposed to other mechanisms.

The primary mechanism of diffusive retention in clay host rocks is likely to be molecular sieving. Specific selectivities are not readily identified, although radionuclides having relatively large ionic radii are likely to be retained more readily.

(b) sorption reactions;

Sorption is a broad term used to describe ionic attraction and exchange reactions attributable to electro-chemical forces. All sorption reactions are reversible, and therefore retention is not permanent. Nevertheless, retention times may be significant. Both adsorptive and absorptive mechanisms are well-understood for simple solutions containing a single ionic species. For these systems, sorption capacities and selectivities may be predicted on a quantitative basis. Unfortunately, however, sorptive processes in complex chemical systems (as represented by the repository environment) cannot be reliably predicted on a theoretical basis; and considerable uncertainty exists in the extrapolation of the present limited empirical data-base, as will be shown subsequently.

Adsorption is a surface reaction, and assumes greatest significance in solid materials having a
high specific surface and a net negative charge. Adsorption is therefore almost exclusively attributable to clay materials. Most clays possess a net negative surface charge, due to natural exchange processes within their crystal lattice (83). Thus, adsorption reactions display a marked affinity for cationic radionuclide species; see table 3. However, physical damage to clay particles (edge defects) can also result in a net positive charge imbalance. It has been found that chlorites, for example, can possess a net positive charge and may therefore display an affinity for anion sorption. However, cation selectivity is the general rule and the higher valency (tri-valent and tetra-valent) ions within the groundwater will tend to be preferentially adsorbed.

Absorption involves cation exchange reactions between species in solution and those within the molecular lattice of sheet silicate materials, such as clays and zeolites. The main structural cations which form the silicate layers are strongly bonded to $O^{2+}$ and $OH^-$ ions, and are therefore unavailable for exchange. The exchangeable ions occur at inter-layer positions and are bound by relatively weak electrostatic forces.

A high cation exchange capacity (CEC) is indicative of a high inter-layer cation population; as occurs in smectite minerals and zeolites. Common interchangeable cations are $Ca^{2+}$ and $Na^+$, although $K^+$, $Mg^{2+}$ and other ions may also participate. Exchangeable ions within the mineral lattice are replaced by counter-ions in the solution, until equilibrium is reached.
The affinity for cation adsorption depends upon the relative ionic charge and radius of the species involved. Ions in solution may completely replace similarly charged counter-ions in the solid provided the differences between their ionic radii is less than 15% of the radius of the smaller ion (160). For greater size differences, partial substitution may occur. Under these conditions, competition for exchange sites amongst the species in solution gives rise to an equilibrium condition between ions held in the liquid and solid phases. Another factor which influences the hierarchy for ion selectives is the electro-negativity of the radionuclides involved; a relatively high value being indicative of a more reactive ion (see figure 3).

Due to the relatively high sorptive capacity of clay minerals, argillaceous host formations possess superior sorption properties to those of crystalline rocks. In crystalline rocks, sorption is likely to occur only where clay minerals or other alteration products are present on fissure surfaces. It is therefore significant that adsorption capacity depends on the surface area of the solid in contact with the solution, whereas absorptive capacity depends on the solid volume. The volume of absorptive material present in the fissures of crystalline host rocks is limited. Furthermore, the ability of the material to retain radionuclides depends on the velocity of fissure flow in relation to the rate of lateral diffusion. Hence it is apparent that experimental laboratory data concerning the sorptive capacities of crystalline rocks, based on the use of finely crushed rock samples, must be regarded with considerable caution and cannot be reliably applied in assessing the
insitu retention of radionuclides.

(c) precipitation reactions;

Mechanisms (a) and (b) above involve the retention of radionuclides which are present in ionic form or which exist as colloids (micelles) in the groundwater. In either of these forms radionuclides are potentially mobile. However, precipitation reactions are irreversible chemical reactions which lead to the precipitation of the reaction products as insoluble solids (usually hydroxides). Reactions of this type therefore represent a relatively powerful retention mechanism for specific radionuclides.

The propensity for retention mechanisms (b) and (c) above can be estimated by reference to the ionic potentials of the principal radionuclides leached from the wastes into the groundwater environment. The ionic potential, P, of a particular species of radionuclide is defined by the relationship:

\[ P = \frac{z}{r} \]

(1)

where

- \( z \) is the effective ionic charge (or valency)
- \( r \) is the effective ionic radius

As described in Chapter 8.2, the valency state of leached radionuclides is dependent upon the Eh/pH of the
groundwater; see figure 16. Hence ionic potentials also vary according to the prevailing Eh/pH conditions. Figure 21 shows a graph of effective ionic charge (valency) against effective ionic radius, for the pure water environment; in which the upper and lower limits of ionic potential are defined as 0.03 and 0.12 per picometre respectively (160). The diagram therefore strictly applies to aqueous solutions in which the only anion is OH⁻ (aq). This is clearly a gross simplification of the potential range of natural geochemical environments. However, in the absence of adequate data, it serves to illustrate the relative mobilities of cationic radionuclides in an aqueous system. The anionic species cannot be represented in this way. They must be regarded as relatively mobile, since their negative potential prevents reactions with the majority of species occurring in natural groundwaters.

The principal radionuclides present in 1000-year old high-activity radioactive wastes are shown in Table 3. In Table 8, the same radionuclides are listed to show their ionic potentials for each of the valency states in which they may occur (according to the actual Eh/pH environment). Figure 20 indicates which of these are likely to remain in soluble form and which are likely to be precipitated as insoluble hydroxides.

The classification shows that most of the actinides and many of the fission products can occur in either group, depending on their valency states when leached from the waste matrix. The relevance of Eh-pH stability diagrams, as described in Chapter 2.2, is therefore apparent.

The ability of host rock materials to retain the more
mobile radionuclide species depends on a variety of geochemical factors. For diffusive retention (molecular sieving) the primary factor is the relationship between the ionic radius, \( r \), and the dimensions of ports or channels within the molecular structure of the solid. For sorption reactions, electronegatives and ionic radii exert a controlling influence. Precipitation reactions tend to occur within a limited range of ionic potentials, as shown in figure 20.

The first group of radionuclides to be considered comprises cationic species having low valencies of \( 1^+ \) or \( 2^+ \), and relatively large ionic radii. Since these species will have the lowest ionic potentials, they will form the most readily hydrated (soluble) and therefore the most mobile cations. They include caesium (\(^{135}\text{Cs}\)), rubidium (\(^{87}\text{Rb}\)), samarium (\(^{147}\text{Sm}\)) and tin (\(^{126}\text{Sn}\)) of the fission products; and radium (\(^{226}\text{Ra}\)) and actinium (\(^{227}\text{Ac}\)) of the actinides (actinium has a valency of \( 3^+ \) but its ionic radius is large). These radionuclides tend to have low electronegatives (except for tin) and the main sorbing mechanism will therefore probably be diffusive retention by molecular sieving. Although adsorption and absorption could occur onto metallic oxides and hydroxides present in the rock or the repository backfill, diffusive retention is likely to be more significant due to competition for sorption sites from more reactive radionuclides.

For the retention of these particular radionuclides, the desirable host rock properties are:

- an overall negative charge to promote cation exchange

- a molecular lattice containing channels, ports and
exchange sites large enough to receive diffusing ions with dimensions 200-400 pico metres in diameter.

The second group of radionuclides are those which can occur either as soluble cations or insoluble hydroxides, depending on their valency state (see table 8). In the higher valency states (4⁺, 5⁺ and 6⁺) such ions have comparatively small ionic radii and are therefore relatively reactive. They will detach the hydroxyl (OH⁻) ions from surrounding water molecules, releasing hydrogen ions, and will precipitate as insoluble hydroxides within the back-fill.

It must be noted that retention by chemical precipitation is not necessarily permanent since radioactive decay can produce a change in valency (auto-reduction), due to the emission of particulate radiation; e.g. hexavalent americium (6⁺) will transmute to pentavalent americium (5⁺) by α-emission. Thus, radionuclides initially precipitated as solid compounds, may gradually regain their mobility due to radionuclide decay. However, precipitation reactions are clearly a potentially very important retention mechanism for the long-lived radionuclides species as a whole.

Nearly all the actinides are soluble in their lowest valency states (2⁺ and 3⁺). Due to their relatively high electro-negativities, they tend to be fairly reactive (see table 3) and their ionic potentials normally lie close to the insoluble hydroxide boundary (see table 8 and figure 20). The primary retention mechanism for these radionuclides is sorption by cation exchange reactions within the back-fill.

The third category of radionuclides are those which,
because of their very high ionic potential, can release hydrogen ions from water molecules and form co-valent bonds with oxygen. In certain Eh/pH conditions, these reactions can produce soluble complex ions with other species present in the groundwater. For example, the uranium $6^+$ cation can become $\text{UO}_2^{2+} \text{(aq)}$, and is able to complex again with $\text{CO}_3^{2-}$ to form the aqueous molecule $\text{UO}_2\text{CO}_3$.

All of these forms can be transported in water (see figure 16c).

For this group of radionuclides, it would be desirable to induce and maintain an environment which will promote the reduction of high-valency cations so that they may react with metal oxides and hydroxides. The reducing environment may be produced by the presence of minerals which selectively sorb oxygen; such as vivianite and apatite, which may be present to a limited extent in crystalline and argillaceous rocks.

The last category of radionuclides comprises the anionic species ($\text{M}^-$). For this group, the mechanisms which promote retention of cations ($\text{M}^+$) by absorption or adsorption will not be effective. Some retention may be expected due to molecular sieving, but there is no positive charge to reinforce the retention mechanism. These anionic species therefore tend to remain highly mobile, as ions in solution. The main radionuclides in this category are selenium ($\text{Se}^{79}$), technetium ($\text{Tc}^{99}$), polonium ($\text{Po}^{210}$) and iodine ($\text{I}^{129}$). All are fission products, and their common valencies are $1^-$ to $2^-$. However, in extreme environments they can also occur as cations with valencies of $6^+$ and $7^+$.

The anionic species are relatively mobile in all host
rocks, and physico-chemical retention would require the introduction of a backfill containing reactive cationic elements which will combine with anionic species to form solid precipitates. Phosphorus forms solid compounds with many anions; especially iodine ($^{129}$I). Lead is also known to react with anionic radionuclides and will precipitate solid compounds at very low levels of saturation.

The foregoing commentary outlines the conditions which favour one or other of the retention mechanisms which could operate in an underground repository environment. Although it is generally assumed that no speciation of the radionuclides occurs during the initial leaching process, it has been shown that physico-chemical retention mechanisms can subsequently exert a selective influence. Selectivities are largely based on the ionic potentials, ionic forms and ionic radii of the radionuclides concerned.

However, theoretical assessments based on the assumption of a pure water environment are generally misleading. In practice, the solutions involved are complex and contain many competing chemical constituents, both stable and radioactive. The backfill and host rock materials, through which contaminated groundwater would flow, are complex mixtures of various minerals, and possibly organic compounds. Thus, it is impracticable to make a quantitative evaluation of radionuclide retention in terms of individual chemical reactions. A more empirical view is necessary.

Cation sorption is probably the best understood of the retention mechanisms involved and, for simple solutions, can be evaluated in terms of the following parameters (115):
(a) the cation exchange capacity (CEC) of the solid

(b) the equilibrium constant, \( K \), of the system

(c) the reaction rate, \( r \).

The first of these parameters, (CEC), is a measure of the total cation exchange capacity of the solid, expressed in milli equivalents per gramme (meq/g). An 'equivalent' is defined as the mass of a substance which will displace 8 parts by mass of \( O_2 \), and is obtained by dividing the relative atomic mass by the valency, e.g.

\[
eq \text{equivalent of } K^+ = \frac{39}{1} = 39
\]

\[
eq \text{equivalent of } Ca^{2+} = \frac{40}{2} = 20
\]

Equivalents are used because the valency of the ions in the host environment determines how many are replaceable by counter-ions in the solution.

The CEC may therefore be determined experimentally. Unfortunately, however, methods differ. Hence difficulties arise in comparing data from different sources.

The second parameter, \( K \), is an equilibrium constant which reflects the distribution of replaced and replacing ions. For an exchangeable ion, \( A \), in the solid and a radionuclide, \( R \), in solution, the equilibrium constant, \( K \), is defined by the theoretical relationship:
\[ K = \frac{C_A q_R}{C_R q_A} \] ................................. (2)

Where \( C_A \) is the concentration of ion A (the replaced ion) in the solid

\( C_A \) is the concentration of radionuclide R (the replaced ion) in the solution

\( q_A \) is the equilibrium capacity of ion A on the solid

\( q_R \) is the equilibrium capacity of radionuclide R on the solid.

Equation 2 assumes the same valency in ions A and R and must therefore be modified for ions of differing valencies.

The third parameter, \( r \), is a measure of the rate at which sorption reactions occur. Absorption reactions require diffusion of ions into the solid, and hence, overall sorption rates depend on the rate of flow of contaminated groundwater and the diffusivity of the intact rock. It must be noted that all relationships based on equation 2 assume that the groundwater residence time is sufficient for equilibrium conditions to be attained, and slow rates of groundwater movement are therefore necessary.

However, as previously noted, the theoretical approach is not readily applied for complex chemical systems. A more readily measured empirical coefficient may be derived from equation 2. For the groundwater/host rock system it is reasonable to assume that \( C_A \) and \( q_A \) are
effectively constant, since the concentration of the radionuclide in the groundwater \( C_R \) is likely to be very small. Hence \( C_A/q_A \) may be regarded as a constant, and the coefficient \( K \) in equation 2 may be replaced by an empirical distribution constant, \( K_d \), defined by the relationship:

\[
K_d = \frac{\text{mass of sorbed radionuclide } R}{\text{gm of backfill}} \times \frac{\text{mass of dissolved radionuclide } R}{\text{cm}^3 \text{ of solution}} \tag{3}
\]

The units of \( K_d \) are therefore expressed in \( \text{cm}^3/\text{gm} \), and it is apparent that values may be readily determined by experimental methods. A solution containing a known concentration of radionuclide, \( R \), is first allowed to reach equilibrium with a sorptive backfill material. The fluid is subsequently separated, and the concentrations of the radionuclide within the solid and solution are measured and substituted into equation 3.

The significance of \( K_d \) is that an estimate of the retardation effect of sorption reactions may be made; based on the assumption that chemical equilibrium prevails within the contaminated groundwater/host rock system. Having obtained an appropriate \( K_d \) value it is possible to show that, for a porous medium, the travel time of radionuclide \( R \) is greater than the travel time of the groundwater, by an amount given by:

\[
\frac{T_R}{T_W} = 1 + \frac{K_d}{n} \tag{4}
\]

where \( T_R \) is the travel time of the radionuclide
$T_W$ is the travel time of the groundwater

$K_d$ is the empirical distribution coefficient

$\gamma$ is the density of the rock

$n$ is the porosity of the rock

Where fissure flow predominates, travel times are related by the expression:

$$\frac{T_R}{T_W} = 1 + K_{df} M_s \quad \ldots \quad (5)$$

where $M_s$ is the specific surface of the fissure (ratio of area to volume)

$K_{df}$ is the distribution coefficient for the fissure system

In the above equation, the parameter $K_{df}$ is an analogous distribution coefficient for the fissure flow system (100) defined as:

$$K_{df} = \frac{\text{mass of sorbed radionuclide/unit area of fissure}}{\text{mass of dissolved radionuclide/unit volume of fissure}} \quad \ldots \quad (6)$$

Despite the apparent attractions of the $K_d$ concept, in terms of modelling rates of radionuclide transport, a considerable controversy exists in terms of its application. The derivation of $K_d$ and $K_{df}$, based on equation 2, holds only for reversible reactions in a
dilute solution. In addition, $K_d$ values are valid only if:

- they are insensitive to changes in concentration over the range considered;
- they remain constant after equilibrium is attained;
- the influence of temperature and pH is taken into account.

The $K_d$ concept is widely applied in waste conditioning by ionic exchange methods (113). In addition, some practical experiments have been carried out in the context of radionuclide migration studies. However, no systematic tests have been conducted in conditions which actually represent the repository environment, taking account of groundwater compositions and the chemical forms of the radionuclides of primary concern. Most experiments have been laboratory-based, utilising pure water, spiked with relatively few radionuclides; often $^{137}$Cs and $^{90}$Sr. Since the latter are reasonably short-lived in relation to the probable corrosion-free life of the waste units, the data obtained may be of little relevance; see Chapter 19.

It must also be noted that, as originally defined, $K_d$ applies to sorption reactions only. A more global empirical distribution coefficient would have to be defined in order to account for the additional effects of diffusive retention and precipitation reactions. It would therefore appear desirable to devise standardised test procedures which allow a more general empirical assessment of backfill retention properties. This could
provide a basis for more meaningful comparisons, recognising that the theoretically-based approach is in-appropriate.

In the absence of adequate data, at the present stage, the retention properties and selectivities of host rock and backfill materials can only be assessed on the basis of published values of $K_d$, CEC and selectivity coefficients. The potential for diffusive retention and precipitation reactions must therefore be separately assessed by reference to factors previously outlined.

9.3 Thermal Properties

The thermal properties of host rocks are highly important in the context of repository design and construction. This is particularly so in the case of high-level wastes, since the dissipation of heat generated during the first 1000 years or so is a crucial aspect of the design of the disposal system; see figure 10. However, it is equally important to consider the thermal properties of host rocks from the containment view-point.

Where high-level wastes are introduced in solidified form, e.g. as a borosilicate glass/waste matrix, the integrity of the waste units must be preserved by limiting the maximum induced temperature increase, so as to prevent de-vitrification or melting. The temperature stability of the rock must also be considered, and since rocks comprise an assemblage of minerals, it may be anticipated that differential thermal expansion of mineral constituents could bring about disruption in the rock fabric at some threshold temperature, with melting at some higher value.
Borosilicate glass begins to divitrify at temperatures around 450°C whereas crystalline rocks (which have the lowest melting temperature of the host rocks under consideration) have melting points in the range 700°C to 900°C for depths up to about 5km (42). Since the waste units (which constitute the heat source) undergo a greater temperature rise than the host rock (which acts as a heat sink), the maximum centre-line glass temperature of 450°C will represent the critical design temperature constraint with respect to melting for high-level waste disposal concepts.

However, other aspects of rock/waste behaviour must be considered. Increase in temperature generally results in an increase in chemical reactivity, by altering reaction field conditions and the rates at which reactions occur. Thus, rates of corrosion of materials which may be introduced into the repository (e.g. waste unit claddings and tunnel and borehole linings), and rates of leaching of radionuclides into the groundwater may be expected to increase; see Chapter 19.

The results of laboratory leaching tests on borosilicate glass, reported by Hill and Grimwood (100), indicate that a temperature increase of 40°C could bring about an order of magnitude increase in leach rates. The absolute temperature is the relevant factor, and hence a low geothermal gradient is beneficial in limiting the ambient temperatures prior to disposal, and those which will prevail in the near field after the 1000 year heat production period. Figure 22 indicates the range of ambient temperatures expected with increasing depth, based on typical European geothermal gradients (87), together with the corresponding range of leach rates for borosilicate glass reported by Hill and Grimwood (100).
However, consideration of the temperature stability of the host rocks tends to lead to the selection of lower maximum temperature limit than those associated with melting. For crystalline rocks, micro-cracking due to differential thermal expansion of constituent minerals may develop at temperatures in the range 100°C to 200°C (42). In clays, loss of physically adsorbed water at temperatures above 105°C to 110°C is likely to produce an irreversible reduction in cation sorption capacity (41, 42). For saliferous rocks, release of water of crystallisation may be accompanied by expansion processes at temperatures exceeding 150°C to 200°C and at temperatures approaching 250°C decrepitation has been observed in laboratory tests, where the pressure of brine inclusions exceeds that of the surrounding material (31).

Thus, in general, maximum design temperature limits have been fixed by reference to the temperature stability of the host rock. The upper temperature limits which have been generally accepted are (41, 42):

- Crystalline rocks (granite) ; 100°C
- Saliferous rocks (halite) ; 100°C - 150°C
- Argillaceous rocks ; 100°C

The maximum induced temperature associated with high-level wastes, and their effect upon rock properties may be regarded as near field effects. However, in the long-term, the effect of artificially-induced temperature gradients, superimposed upon the ambient geothermal temperature field, could result in significant perturbation of the groundwater flow system. The concern here is that convective 'cells' could develop which result in a net upward flow of groundwater.
from the waste emplacement zone. This problem is of concern only in relation to crystalline and argillaceous rocks, since they contain 'free' circulating groundwater.

The values of maximum rock temperature and temperature gradients resulting from the emplacement of high-level wastes depend upon the heat output of the wastes, their spatial configuration within the repository, and the thermal properties of the host rock. Assuming that wastes are placed in good thermal contact with the rock (e.g. by means of a dense surrounding backfill) conduction will be the principal mode of heat transfer.

Thermal conductivity is a measure of the rate of linear heat flow across a unit cross-sectional area, under a unit temperature gradient. Thus, for an elemental block of material of length l and cross-sectional area A in thermal equilibrium with a constant heat source:

\[ W = \Gamma A \left( T_2 - T_1 \right) \]  

(1)

where \( W \) is the power of the heat source
\( T_1, T_2 \) are the steady state temperatures at each side of the element
\( \Gamma \) is the coefficient of thermal conductivity of the material

The above may be expressed more generally as:

\[ \frac{dQ}{dt} = -\Gamma A \frac{dT}{dl} \]  

(2)
where $\frac{dQ}{dt}$ is the heat flux and $\frac{dT}{dl}$ is the temperature gradient.

Considering a column of cylindrical high-level waste units, placed in a borehole so that good thermal contact with the rock is achieved; then, referring to figure 23a, the radial heat flow across an annular element of unit length and thickness $\delta t$, at some distance, $r$, from the centre of the waste units is given by:

$$dQ = \int_0^{2\pi} \int_{r_2}^{r_1} d\theta \frac{dQ}{dr}$$

Assuming that, in the equilibrium condition, the rate of heat flow across any such element is constant (i.e. $\frac{dQ}{dt}$ is constant) then:

$$\int_{T_2}^{T_1} \delta T = -\int_{r_2}^{r_1} \frac{1}{r} \frac{dQ}{dr} dr$$

$$T_1 - T_2 = \frac{1}{2\pi/\Gamma} \frac{dQ}{dt} \ln \frac{r_2}{r_1}$$

The form of the radial temperature profile predicted by equation 4 is illustrated in figure 23b, assuming identical thermal conductivities in the waste unit, the backfill and the host rock. In practice, equation 4 should be modified to take account of three-dimensional diffusion effects; particularly with increasing distance from the heat source. In addition, equation 4 cannot be used to predict the temperature profile within the waste matrix itself. Thus, the profile shown in figure 23b is subject to inaccuracies for values of $r$ in
the range $0 < r < r_1$, and $r > r_3$. Nevertheless, it is useful for examining the effects of different thermal conductivities upon near-field temperatures.

Equations 3 and 4 show that the temperature drop across any annular zone in the near field system is inversely proportional to the thermal conductivities of the materials concerned. Thus, the temperature profile within the waste unit itself depends only on the thermal conductivity of the glass. The temperature difference between the edge of the host rock and the centre-line of the waste unit varies with the waste heat output. However, the absolute temperatures in the equilibrium condition depend primarily on the thermal conductivity of the host rock.

Table 9 shows the range of heat transfer properties representative of the principal rock types and waste unit and backfill materials based on published data from a variety of sources (37, 47, 125, 130, 177).

The heat transfer properties of rocks depend on their structure and mineralogy. For many rocks it is found that thermal conductivity varies with the square of the solidity $\Omega$, where $\Omega$ is the complement of porosity $n$; i.e. $\Omega = 1 - n$ (187). This may be said to account for the relatively low thermal conductivity of the clays and argillaceous rocks, which have relatively high porosities. Quartz content is particularly significant in granites and the mass thermal conductivity may exhibit an approximately linear relationship with quartz content according the the equation:

$$K = 2.0 + 4.0 \psi \quad \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdots \cdOTS
The properties shown in table 9 illustrate the relative merits of crystalline, argillaceous and saliferous rock types in terms of thermal performance. It is clear that saliferous rocks are considerably better in this respect than both crystalline and argillaceous types; whilst some argillaceous rocks may exhibit very low conductivities (depending on moisture content). The ranges of values shown reflect the inherent variability within any particular rock type. In addition, it must be noted that thermal conductivity is temperature-dependent; generally decreasing in value with increase in temperature (47, 97).

For a maximum rock design temperature of say 100°C, conductivity values may increase by more than 10% as the ambient rock temperature is reduced to 50°C. Thus, the critical period which governs near-field thermal design occurs soon after emplacement, when the initial thermal equilibrium is attained.

Unlike the host rock mass, the thermal properties of high-level waste units may be determined with a high degree of accuracy. To a lesser extent, this also applies to backfill materials placed around the waste units. It is apparent that borosilicate glass is a poor conductor of heat in comparison with most rocks, and will therefore experience relatively high temperature gradients. However, the waste containers are excellent thermal conductors and very small temperature differences are likely to occur between their inner and outer surfaces; see figure 23).

For comparative purposes, the thermal properties of a range of potential backfilling materials are also
tabulated. The thermal conductivities of most of these materials tend to be lower than those of the host rocks. However, the incorporation of special high conductivity backfills such as graphite or quartz could provide scope for design optimisations. In rock salt formations, it seems likely that crushed salt spoil or anhydrite is likely to provide the optimum material in terms of thermal performance.

However, it must be noted that backfill thermal properties are likely to vary with moisture content and insitu density. For clay fills, in particular, thermal conductivities may vary by a factor of 4 or more, according to the insitu moisture content of the material; and for dry clay fills, values are likely to be as low as 0.3 - 0.5 W/m°C.

Swedish research has indicated that the thermal properties of bentonite can be improved considerably by the addition of fine quartz sand, without excessive impairment of other backfill properties (177). However, the introduction of high conductivity materials is unlikely to produce significant benefits unless sufficient material is introduced to provide inter-particle contact. The design of mixtures and the physical conditions at placement are therefore important areas of research in relation to backfill thermal performance.

Another design aspect is the potentially beneficial effect of modifying the thickness or shape of the waste containers themselves. The provision of a thick metallic overpack may provide considerable improvement in terms of radial heat dissipation and reduction in the temperatures and temperature gradients to be sustained within the system; see Chapter 15.
10. ENGINEERING PROPERTIES

10.1 Introduction

This chapter considers the engineering properties of crystalline, argillaceous and saliferous rocks in relation to repository design and construction. It is important to note that the construction depths contemplated for high-activity radioactive waste repositories are an order of magnitude greater than those encountered in normal civil engineering experience. Except in mountainous areas, tunnels driven for civil engineering purposes seldom attain a depth of cover in excess of 50m, whereas most conceptual design proposals for high-activity radioactive waste repositories envisage depths in excess of 500m.

As will be shown in later chapters, this has resulted in a tendancy for the development of repository design and construction techniques based largely on experience gained from the mining industries. However, the objectives for mine and repository construction are fundamentally different. In a deep mining operation, the primary objective is to achieve low-cost excavation and there is generally no incentive to minimise disturbance to the host rock. Moreover, deep mines are often constructed in the stronger (more competent) rocks where the major ore bodies are found and where low-cost, rapid excavation techniques are possible. With the exception of current experimental shaft-sinking and tunnelling operations at Mol in Belgium, which are being carried out as part of a repository design and development programme, the author is not aware of any previous experience of large-scale underground constructions in plastic clay formations at depths measurable in hundreds of metres. Plastic clay host
formations undoubtedly represent the lower limit of the strength range for repository materials, but it is apparent that deep repository construction the weaker host materials in general is likely to pose many new and difficult technological problems.

### 10.2 Strength, Deformation and Support Requirements

In the design of underground excavations, it is necessary to compare the strength of the insitu rock to the stresses imposed, and thence to determine the resulting deformation response. The definition of failure will then normally depend upon the amount of deformation which can be tolerated in the underground facility without adverse consequences.

The response of an element of rock material to imposed ground loading is influenced by the magnitude of the applied major and minor principal stresses. In unconfined compression, the minor principal stress is zero and the major principal stress is equal to the applied load. This uniaxial loading condition is relatively easy to reproduce in laboratory tests on intact rock specimens and generally indicates a readily defined peak strength value for a particular material. Because of this, the unconfined compressive strength is the most widely used rock strength index (104).

Figure 24 indicates the approximate range of unconfined compressive strengths and deformation moduli exhibited by crystalline (igneous), argillaceous (including argillaceous rocks and plastic clays), and saliferous rocks, based on a review of published data (20, 47, 61, 104, 129, 155).

British standard strength classification limits (in terms of unconfined compressive strength) are indicated
for reference purposes, and values for concrete and steel are also included for comparison. The indicated range of strengths for overconsolidated clays is based on a crude estimate by the author, using the relationship shown in figure 15C and substituting a range of possible pre-consolidation pressures and values of plasticity index PI in the range 10% - 60%. However, it is recognised that, in practice, the overlap in strengths between hard over-consolidated clays and weak rocks does not lend itself to close definition.

Figure 24 serves to illustrate the extreme range of material strengths and stiffnesses under consideration, and it indicates that the range of stability conditions for respository construction in these materials is enormous.

At a particular depth below ground level, the vertical component of the virgin insitu ground stress, $\sigma_v$, is closely approximated by the weight of overburden. Thus:

$$\sigma_v = \gamma d \hspace{1cm} (1)$$

where $\gamma$ is the average unit weight of the overlying rocks

$d$ is the depth to the point under consideration

Although the approximate value of $\sigma_v$ is readily determined by equation 1, the ratio of horizontal to vertical insitu stress is less predictable. The insitu stress ratio, $K_o$, is defined by the relationship:

$$K_o = \frac{\sigma_h}{\sigma_v} \hspace{1cm} (2)$$
where $\sigma_h$ and $\sigma_v$ are the horizontal and vertical principal effective insitu stresses in the undisturbed condition.

For indurated host rocks, elastic theory provides a guide to the factors which influence the variation of horizontal stresses with increasing depth. In an idealised situation, in which the rock profile has not been subjected to post formational or tectonic disturbance, the value of $K_o$ is predicted by elastic theory as:

$$K_o = \frac{\nu}{1-\nu}$$

where $\nu$ is Poisson's ratio.

It is found that rocks which behave elastically over a particular stress range, exhibit values of Poisson's ratio varying from about 0.15 to 0.35. However, corresponding theoretical $K_o$ values (in the approximate range $0.20 < k_o < 0.55$) are often in conflict with the evidence available from sub-surface construction and insitu measurements. The majority of the available data suggests that horizontal insitu ground stresses encountered at moderate depths are normally greater than the vertical stress; i.e. $K_o$ is greater than unity (104). For a given rock formation this may be attributable to the combined influence of several factors, including:

- the method of formation or deposition;
- the nature and extent of post-formational tectonic disturbance;

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the amount and rate of surface weathering and erosion;

the influence of topographical features.

The combined effects of the above cannot be accurately assessed on a quantitative basis. However, the evidence suggests that horizontal stresses are strongly dependent upon the geological history of the formation. For example, in an area of subsidence, the lateral compression in the central regions generates higher horizontal stresses than those experienced at the margins. Similar variations are likely to be associated with significant changes in surface topography, or the occurrence of faults or intrusions.

Whatever their origin, horizontal stresses tend to remain 'locked-in', due to the lateral confinement of the surrounding materials. However, the removal of overburden by erosion generally produces some vertical elastic rebound or exfoliation, as evidenced by the development of sheeting joints in crystalline rocks. This is normally accompanied by a reduction in vertical stress within the upper layers. In consequence, the more brittle elastic rocks tend to exhibit relatively high values of $K_o$ (typically $1 < K_o < 3.5$) at moderate depths below ground level.

However, the more plastic or visco-elastic materials respond differently, and through creep processes are often able to undergo a slow re-adjustment in response to tectonic or geological change. In consequence, values of $K_o$ correspond more closely to those predicted by theory. For plastic and visco-elastic materials, Poisson's ratio approaches 0.5. Therefore theoretical
values for materials of this type (as predicted from equation 3) tend towards unity, implying a hydrostatic insitu stress regime. Most saliferous rocks are characterised by visco-elastic/plastic stress-strain behaviour, even at moderate stress levels. Therefore, hydrostatic conditions tend to prevail. Argillaceous rocks and, to a lesser extent, crystalline rocks, may also exhibit a plastic behaviour at large depths and correspondingly high stress levels. Values of $K_o$ for these materials may decrease with increasing depth and approach unity at some considerable depth below the surface.

Figure 25 illustrates the actual variation of $K_o$ with depth for several rock types, based on data obtained from in-situ stress measurements compiled by Hoek and Brown (104). The data appears to confirm a tendency towards hydrostatic stress conditions at great depths. However, with the exception of saliferous rocks, it appears that $K_o$ values at depths in the range 500 - 1000 m could lie within the range $K_o = 1.0$ to 3.0.

For the clay formations, generalised predictions based on an elastic model analogy are less appropriate. Based on the Mohr-Coulomb relationship:

$$
\tau = c' + \sigma \tan \phi' \\
$$

it may be shown that the limiting values of $K_o$ must lie within the active and passive shear failure thresholds $K_o$ and $K_p$, as illustrated in figure 26 (127).

For normally-consolidated clays, it is found that the value of $K_o$ is closely approximated by the relationship:

$$
K_o = 1 - \sin \phi' \\
$$

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where $\phi'$ is the internal friction angle of the material, in terms of effective stress. It is also found that values of $\sin \phi'$ for normally-consolidated clays vary with the plasticity index (PI), as shown in figure 26b (126). From these relationships it may be shown that the values of $K_0$ for normally consolidated clays will generally lie in the range 0.42 - 0.66.

However, as described in Chapter 8.1, the clay hosts under consideration will invariably have been subjected to delayed compression and/or over-consolidation. Hence larger values of $K_0$ may be anticipated. It is found that the value of $K_0$ increases with increasing values of over-consolidation ratio (OCR) and at OCR values of about 20 to 25; passive failure is induced as $K_0$ approaches the passive pressure coefficient $K_p$.

Below this threshold, the variation of $K_0$ with over-consolidation ratio is found to obey the empirical relationship

$$\frac{K_0 (OC)}{K_0 (NC)} \approx OCR^{0.5} \quad \text{............... (6)}$$

where $K_0 (OC)$ and $K_0 (NC)$ are the values of $K_0$ for the over-consolidated and normally-consolidated states respectively.

Combining equations (5) and (6) gives

$$K_0 (OC) = (1- \sin \phi') \cdot OCR^{0.5} \quad \text{............... (7)}$$

Hence, for given values of OCR and for values of $\sin \phi'$ derived from figure 26b, it is possible to make a tentative estimate of the value of $K_0$ for
over-consolidated clays.

The values of plasticity index for European clays are likely to lie in the range $20 < {PI} < 60$, as shown in figures 14b and 26b. Hence, assuming a limiting maximum OCR value of 25 in the upper layers of the clay formation, the maximum value of $K_o$ calculated from equation 7 is about 3, and will tend towards unity at depths, where over-consolidation ratios are in the range $3 < OCR < 8$.

This conclusion appears to be confirmed by in situ measurements in over-consolidated clays, such as the London Clay, in which $K_o$ values of about 3 are found in the upper layers; decreasing towards unity at depth. However, where an over-consolidated clay is re-loaded to a value greater than the pre-consolidation pressure, $K_o$ rapidly falls below unity to the minimum values for the normally-consolidated condition predicted by equation 5.

Figure 27 indicates the influence of $K_o$ on the intensity of the ground stresses induced around circular underground openings, based on elastic theory (167). For $K_o$ values greater than unity the induced stress field is ellipsoidal, implying distortional loading. Thus, in tunnels, maximum induced compressive rock stresses are likely to occur at crown level and may reach values as high as 8 times $\sigma_v$ for high $K_o$ values. In contrast, for hydrostatic stress conditions, such as those which may occur in saliferous rocks and clay formations at depth (under undrained loading), the predicted stress distribution around tunnels is uniform, implying no distortion. Isotropic stress conditions are also predicted in the vicinity of shafts and drill holes, irrespective of the value of $K_o$; due to their orientation with respect to the vertical stress.
component.

In order to assess the stability of repository openings under specific circumstances, it is necessary to predict the induced insitu stress fields as outlined above and to compare these with the mobilised strength of the insitu rock. Hoek and Brown have developed an empirical relationship to describe the failure criterion for intact rocks in terms of the major and minor principal stresses at failure (103,104). The relationship is:

\[ \sigma'_1 = \sigma'_3 + (m Rc \alpha'_3 + s Rc^2)^{1/2} \] .............................. (8)

where \( \sigma'_1 \) and \( \sigma'_3 \) are the major and minor principal effective stresses.

\( Rc \) is the unconfined compressive strength.

\( m \) is a constant varying from about 0.001 for highly disturbed rock masses to about 25 for hard intact rock.

\( s \) is a constant ranging from 0 for jointed masses to 1 for intact rock.

Guidance for the assessment of suitable values of \( m \) and \( s \) according to geological composition, rock quality and frequency of jointing is also provided by Hoek, based on rock mass classification systems oposed by Bieniawski and Barton, Lien and Lunde (15, 24).

However, in view of the wide range of possible insitu stress levels and stress redistributions, coupled with the range of rock strengths and rock mass quality characteristics which may be envisaged, it is clear that an infinite range of ground stability conditions may be
encountered for repository construction in the three host rock types under review.

An approximate indication of the range of stability conditions likely to be encountered with increasing depth in each rock type may be obtained by the extrapolation of data from past underground construction experience. This has shown that the ratio of unconfined compressive strength to the vertical insitu stress is a useful indicator for most situations.

For indurated rocks, the rock competence factor, $F$, is defined by:

$$F = \frac{R_c}{\sigma'_v} \tag{9}$$

where $R_c$ is the unconfined compressive strength of the intact rock

$\sigma'_v$ is the effective overburden pressure

However, for clays and weak argillaceous rocks, values of $F$ become very small at large depths. For these materials it is more convenient to use the over-load factor, $N$, defined by:

$$N = \frac{\sigma'_v}{C_u} \tag{10}$$

where $C_u$ is the undrained shear strength of the material.

Both $F$ and $N$ are arbitrarily defined stability factors and are simply related by the equation:
The equation is:
\[ N = \frac{2}{F} \] (11)

Figure 28 indicates the likely ranges of F and N for the host rocks and depths under consideration. For the crystalline, argillaceous and saliferous rocks, the assessment is based on the compressive strength data used in figure 24. The author's assessment for clays is more tentative and is based on a prediction of strength/depth relationships shown on figure 15c, and the somewhat arbitrary assumption that overconsolidation ratios will vary from a maximum of 4 at a depth of 100m to 1 at a depth of 1000m.

Figure 28 provides a general indication of probable tunnel support requirements, based on an extrapolation of experience at moderate tunnel depths. Although the rock competence factor, F, cannot represent the wide range of conditions encountered insitu, where discontinuities in the selected host formation are tight and widely spaced in relation to the size of the excavations, the support requirements indicated are likely to be broadly representative (i.e. where m approaches 25 and s approaches unity; see equation 8).

For example, assuming a repository constructed at 1000m in massive crystalline host rock, the maximum anticipated value of \( K_0 \) is about 2; see figure 25. Based on the theoretical stress re-distribution around circular openings (see figure 27), the maximum induced compressive stress in the tunnel crown would be about 5 \( \sigma_v \) (note \( \sigma_1 = \sigma, \sigma_3 = 5\sigma_v \)). Hence a rock competence factor of 10 indicates that the rock could stand completely unsupported with a factor of safety of about 2.
For rock competence values lower than 10 it is found that increasing rock support is required to maintain stable openings. For values less than 2 continuous structural linings are generally required and tunnelling would normally be carried out within a shield.

Figure 28 provides a rough guide to the range of stability conditions likely to be encountered for the range of depths and rock materials under consideration. It demonstrates that the stronger crystalline and argillaceous host rocks are likely to be completely stable, and thus be capable of standing indefinitely without any form of support, even at great depths. At the other end of the scale, the weak argillaceous rocks and clays must be regarded as unstable over the entire depth range, and with effectively zero stand-up time at large depths.

The crystalline and argillaceous rocks between these extremes represent a wide range of stability conditions. These rocks, which may have an essentially elastic brittle response to loading at moderate depths, may undergo a transition to ductile (plastic or visco-elastic) behaviour at greater depths, where confining pressures become sufficiently high. Hoek (103) recommends, as a rule of thumb, that a transition from brittle to ductile behaviour may be anticipated where the induced minor principal stress exceeds the unconfined compressive strength of the material. Thus, in some situations, a zone of plastic disturbance could be generated which extends over a significant distance from the periphery of the excavations.

Figure 29 shows the variation in shape of the zone of disturbance predicted for vertical and horizontal
circular openings for various values of Ko, based on the elastic stress re-distribution theory illustrated in figure 27. For isotropic stress re-distribution (as predicted for Ko = 1 conditions and for vertical openings in general) the zone of disturbance is symmetrical; whereas for horizontal openings, where Ko is less than or greater than unity, distortional loading occurs and the stress re-distribution and associated ground disturbance is ellipsoidal in shape.

The amount of support required depends upon the size and shape of the zone within which the mobilised strength of the rock exceeds the shear strength of the material. Depending upon the manner of transition from elastic to plastic behaviour, and the creep properties of the rock, the timing of installation of rock support can be critical.

Ground-support interaction diagrams can provide a useful qualitative assessment of relevant ground support/rock interaction phenomena (33, 104). Figure 30a illustrates two hypothetical examples, based on the assumption of uniform stress conditions (Ko=1). The line AB represents the load-deformation curve for a rock with a high competence factor (F>10). Prior to excavation, the insitu stress is σv, and equilibrium stress conditions exist within the undisturbed rock (point A on the diagram). Excavation of a circular opening results in the removal of all internal support pressure and causes a radial deformation, u, whose magnitude is given by OB. Since the rock has a high competence factor, the maximum induced stress is below the elastic limit.* Therefore AB is a straight line, the deformation OB is small, and no internal support pressure is required.

*Based on figure 27, the maximum induced stress would be 2σv, for Ko = 1.
The lines ACEG and ACFH represent the load-deformation curves for a rock having a comparatively low competence factor. This could arise where the repository is constructed at a greater depth in the same host material; or where the depths are identical but the host rock is relatively weak; see figure 28. The straight line AC represents the elastic limit of the material. At point C, the peak strength of the rock in the immediate vicinity of the opening is exceeded and relatively large plastic deformations develop.

For a brittle rock, a broken zone would develop around the opening, and if left unsupported, the load-deformation curve would split into two portions; represented by CEG for the side walls and invert and CFH for the crown. The larger deformations in the crown are associated with the additional load due to the vertical 'dead weight' of the broken rock. In the example shown, the unsupported side walls and invert would become stable at point G. However, the roof would not be capable of supporting itself, due to the additional weight of overlying broken material. Continued displacement would produce further lowering and added weight; leading to roof collapse (point H).

Line DEF represents a support reaction line for support installed within the tunnel shortly after excavation (point D). Due to the interaction of the ground and the support system, plastic ground deformations would be limited to $u_1$ in the side walls and invert and $u_2$ in the roof. The tunnel would then exhibit a slight 'squat'. Alternatively, in the example shown, the support could comprise rock bolt reinforcement in the roof only. In this case, the side-walls and invert
would converge more than the roof, and the zone of disturbed rock would be greater in these areas. For a vertical shaft, the load-deformation curve would be a single line as represented by ACEG, since the weight of broken rock would not act in the plane of the shaft section.

The example described has assumed brittle (strain-softening) behaviour in which the rock exhibits a marked post-peak strength reduction accompanied by rock fracture. In practice, progressive strain-softening may occur and the rock may exhibit time-dependent behaviour. These factors allow control over the timing of support installation after the immediate elastic deformation has occurred.

In conventional tunnelling applications, it is often expedient to allow an elapse of time before support is provided. In this case, point D in figure 30a would move to the right and the required support pressure would be reduced. The same effect could be produced by utilising a relatively flexible support system, which would flatten the slope of the support reaction line. These measures allow economies to be made in the design of ground support systems by mobilising the rock strength to the maximum practical extent.

Figure 30b represents the situation where support is installed close behind the face of the excavation, where radial elastic movements are partly restrained. The support reaction line DE is moved to the left and is shown as the line D'E'. In this situation, the elastic limit C is only marginally exceeded in the equilibrium condition. Consequently, the peripheral zone of fractured rock is minimal and is of similar extent in the roof, side-walls and invert. It would also be
possible to install a relatively stiff support system which would steepen the support reaction line and possibly prevent the occurrence of peripheral rock fractures. Clearly, however, the loads to be sustained by the support system would thereby increase.

The above discussion of ground/support interaction in less stable host rocks has considered only the interaction between the ground and a conventional support system. However, in a repository situation, the assessment of support provided by backfill requires a further interactive analysis. This must recognise the fact that linings or other types of artificial support system have very limited effective lives by comparison with the design life of the repository. It is reasonable to assume that conventional support systems can be fully effective during the repository development and waste emplacement phases; but after backfilling and sealing, gradual deterioration due to corrosion will inevitably lead to structural failure within a time period of the order of 100 years or so, at most, for conventional engineering materials.

Thus, for rocks which are not entirely self-supporting, ground stresses will inevitably be transferred to the backfill placed within the repository openings during or after the waste emplacement phase. When stress transfer takes place, a compressible backfill may allow considerable further plastic deformation to occur, as illustrated by the backfill support reaction line VKL, shown in figure 30c. Figure 30d shows the effect of a more controlled transfer of stress to the backfill, which could be achieved by a staged removal of the temporary support system and provision of a relatively stiff (low volumetric compressibility) backfill material.
The foregoing discussion has indicated the extent to which disturbance of the host rock is likely to occur due to the excavation of repository openings. For crystalline and indurated argillaceous rocks having competence factors less than 10, the 'disturbed zone' is likely to comprise a zone of fractured material, since the failure mode is essentially brittle.

However, saliferous rocks represent a special case and their stability or 'competence' at depth is not adequately represented by figure 28. They have a tendency to behave in a time-dependent visco-elastic manner over a large stress (or depth) range, rather than exhibiting a marked elastic-plastic transition as previously described. This is illustrated in figure 31, which shows the effects of increasing loading rates and stress-levels on the strength and deformation properties of saliferous rocks.

It must be noted that the time-scale of the creep curves in figures 29b and 29c has been compressed, and steady state secondary creep may proceed at a very slow rate over an appreciable period. This secondary creep is not generally accompanied by fracturing, except possibly on a micro-scale, and it is found that larger fissures tend to be impersistant and may become annealed with continued creep movement.

Figure 32 illustrates the influence of this time-dependent behaviour on the stress regime around an underground opening. At a distance from the opening, where hydrostatic stress conditions prevail, the ground remains in a state of pseudo-elastic equilibrium. However, within a large area, major and minor principal stress differences produce a visco-elastic secondary
creep as shown in figure 31.

Even at large depths, the zone of visco-elastic 'stable' secondary creep may extend to the periphery of the excavation so that slow inward radial movement is experienced. A plastic zone of fractured rock in the immediate vicinity of the opening develops only where induced stress differences are sufficiently high to promote failure through tertiary creep; see figure 31.

Thus, provided openings in saliferous rocks can be designed so that no plastic zone develops and a low rate of 'stable' secondary creep can be accommodated, no internal support need be provided. Limited rock bolting may be provided to inhibit the formation of a plastic zone, but full structural support, designed to prevent inward movement, is not normally contemplated.

If a rigid form of support were provided, or if an incompressible backfill was placed within the opening, the radial stress component would gradually increase due to ground-support interaction; and the tangential stress component would gradually reduce to ambient levels. Thus, in the long-term, the 'elastic' zone shown in figure 32 would encroach slowly towards the boundary of the excavation and the support system would ultimately have to sustain a load equal to the full overburden pressure.

The latter explains why provision of support to prevent ground movement in saliferous rocks is generally regarded as uneconomic. The provision of over-size excavations and routine re-excavation is regarded as the norm for most engineering applications.

The situation in plastic clay formations is more complex
and less well-understood in general. Construction at the depths contemplated will require shield-driven tunnelling methods and the installation of a continuous lining behind the working face. Face-support may also be required, or ground-freezing methods may have to be considered. At the depths envisaged, the slopes of the ground load-deformation curves are likely to be very flat; see figure 30.

In these situations, it is common practice to design rigid segmental linings to sustain compressive stress rather than bending stress, by providing for some flexibility at the longitudinal joints. For repository construction, such linings would have to be relatively massive in view of the high ground loadings envisaged at great depth.

The re-distribution of stress around repository openings in plastic clays is likely to create a plastic zone of considerable extent. However, unlike the brittle argillaceous rocks, the clays would not develop fractures, and it is unlikely that the permeability of the material would be adversely affected. The undrained shear strength of the material around the openings would be exceeded within a large radius, and complex pore pressure changes could occur over an extended period. In some cases, long term swelling could result in the build-up of lining loads equal to or greater than the over-burden pressure (48). The influence of these pore pressure changes may be significant in relation to the overall hydraulic regime, the implications of which are discussed further in Chapter 16.3.
10.3 Overburden Materials

As described in Chapter 8.1, the geological settings of potential repository host formations differ widely; see figure 13. Although granite plutons may extend close to or even outcrop at ground level, for other rock types it may be anticipated that a sedimentary sequence of overburden materials of substantial thickness will occur above the host formation. In the case of salt domes, the surrounding country rocks at the lateral margins are also likely to comprise a sedimentary sequence, truncated by the upthrust of the dome.

Clearly, the implicitly excellent qualities of homogeneity and low permeability which are to be applied in the selection of the repository host rock will not apply to these overburden materials. The geological setting of argillaceous and saliferous rocks, in particular, implies that the overburden strata may include a complex sequence of relatively weak and permeable water-bearing horizons which could pose a variety of ground engineering problems.

The engineering properties of the possible range of materials which could be encountered defy any generalised assessment. However, access to the waste emplacement horizon requires that overburden materials must be penetrated by shafts and/or drillholes. Their engineering stability during construction may be assessed by methods outlined in 10.2 above, recognising that the induced insitu stress field around a vertical excavation will generally be isotropic, since the major and minor principal stresses in the plane of the opening should be equal; see figures 27 and 29. Thus, although distortional loading should be minimal, stress intensities may be high (especially where high $K_o$ values occur) and the weaker sedimentary strata may therefore be highly unstable.
Apart from associated difficulties in construction, it must be noted that the construction of vertical openings through a stratified sequence of overburden materials can imply the introduction of artificially induced connections between otherwise naturally distinct aquifers. Based on radiological dating experiments, crystalline and argillaceous host rocks are expected to lie within a zone of relatively immobile groundwater, characterised by extremely long residence times; typically ranging from 10,000 to 30,000 years (44, 72, 203). However, at intermediate levels, one or more confined aquifers may be encountered, in a state of artesian or sub-artesian pressure, depending upon the overall geological and topographical structure. The residence times for groundwaters within these zones will be significantly less than those predicted for the host rock.

10.4 Implications for Repository Construction and Performance

The foregoing account provides a basis for assessing the special engineering problems associated with repository construction in each of the host rocks under review; and the impact which the construction and waste emplacement processes may have upon the host rock and overburden materials involved. Problems of repository construction include two important aspects, namely:

- problems of construction feasibility (which predominate in the case of low competence host rocks such as clays and weak argillaceous rocks, and in unstable overburden materials)
impairment of containment properties due to disturbance to the host rock/groundwater system

In conventional applications, it is normal practice to utilise the natural strength of the ground to maximum advantage in devising the optimum method of excavation and support for underground openings. These two aspects are strongly inter-dependent; the manner of excavation being largely dictated by rock support requirements.

An equally important area of concern in most underground construction operations is the control of groundwater inflow; especially at great depths, where water pressures are high. This aspect also influences the choice of techniques for excavation and ground support.

An outline of potential groundwater control problems in overburden materials has been described in 10.3 above. However, for repository host materials, permeability values are expected to be so low that groundwater inflows during the construction process will be insignificant.

Despite the 'ideal' conditions implied by the low permeability of the host strata, it must be recognised that impairment of this vital containment property may be brought about by the construction process, particularly for low-competence rocks. In shafts and tunnels used for civil engineering or mining purposes, the primary objective is to limit inflow from a radial direction, and in permeable rock this can be achieved by localised ground treatment and/or the introduction of an 'impermeable' lining. However, these measures are designed to reduce flows to levels which although 'low' in conventional engineering terms, are nevertheless
unacceptable for a repository situation.

Figure 38 indicates the ranges of applicability of various specialised construction methods and ground treatment techniques in 'permeable' ground. By comparison with figure 17, it is apparent that where localised reduction of permeability occurs in host rock materials, due to ground disturbance, the very low permeability of the undisturbed material cannot be restored using currently available techniques. It must also be recognised that the longevity of many grout formations, especially organic-based grouts, cannot be assured over the performance period required; see Chapter 17.2.

Moreover, in a repository situation, the introduction of longitudinal flow paths (i.e. flow paths occurring parallel to the longitudinal axes of the excavations) assumes an over-riding significance. Any form of continuous peripheral disturbance around the excavations, which results in a localised reduction in permeability, or the presence of transmissive features within the backfilled openings, can constitute a potential radionuclide migration path. One or more such flow paths may lead directly to the biosphere, via access shafts, or may provide 'short-circuit' connections between otherwise non-transmissive natural discontinuities within a jointed rock mass. The latter can apply even in situations where the induced flow path anomaly is of limited extent; see figure 18. The importance of this aspect of repository design and construction does not appear to have received any detailed attention in the available literature, and is clearly not a matter of concern for conventional engineering applications.
Figure 34 identifies three potential longitudinal flow path zones in and around a backfilled repository excavation. The zone of peripheral ground disturbance is associated with dilatant shear movements which exceed the elastic limit of the rock within a zone whose size and shape is defined by the induced stress field; see figure 28. The inter-face areas at the boundary between the rock and the excavation backfill, or at the extrados and intrados of the lining (where present), are possible planes of separation which could promote 'fissure-flow' conditions. Finally, the bulk fill and possibly the lining material could also provide preferential flow paths where their permeability is significantly lower than that of the undisturbed host-rock.

The relative significance of the potential flow zones identified in figure 34 depends upon the engineering behaviour of the ground and the characteristics of the materials introduced into the repository. For obvious reasons, the development of a fractured zone around the openings in otherwise intact rock would be highly detrimental. This could occur in crystalline and indurated argillaceous rocks which are not completely self-supporting. Even in rocks with high competence factors, requiring no support, conventional drill and blast methods of excavation could create sufficient localised increase in fracture porosity to create a significant peripheral flow path. This would suggest that smooth-blasting methods or machine excavation techniques are to be preferred in such materials.

Where a lining system is installed, the problems of ensuring full contact at the ground-lining surfaces assumes considerable significance in brittle rocks which are partly self-supporting. Furthermore, the provision of a fully effective bulk fill within the excavations
presents considerable difficulties, whether or not a lining is present. This applies particularly in the crown areas of tunnels where access for the placement and proper compaction of fills is not available and separation is therefore likely to occur under the influence of gravity. Problems of restricted access in tunnels also prevents a completely uniform compaction of the fill itself and hence a variable permeability may occur within the backfill mass. Consolidating fills can present additional problems, since they would tend to exacerbate any tendency for separation at crown level.

Linings themselves may provide flow paths, as they deteriorate due to corrosion in the long-term. Eventually they must be expected to fail, and could therefore create additional separation and localised rock fracture around excavations in brittle rocks. Linings subjected to bending stresses due to distortional ground loading are likely to be relatively vulnerable, with corrosion leading to failure in areas of high stress concentration.

Since clays and saliferous rocks tend to deform by plastic or visco-elastic 'flow' processes, the problems of peripheral ground disturbance and interface separation at external lining contacts are not likely to create flow paths. However, in clays, the mechanism of transfer of stress from the lining to the backfill (due to lining failure) would be uncontrolled and unpredictable if linings were left in place, and the possibility of interface flow paths could not be discounted. In saliferous rocks, the absence of a lining could prevent such problems. However, a consolidating-type of backfill would be undesirable, since the host rock would be unable to dissipate pore pressures in the backfill generated by imposed ground
loads. It would therefore appear that some form of backfill containing no 'free' pore water is to be preferred.

Based on the above, it is evident that the repository construction process must mitigate against the formation of flow anomalies by the judicious selection of excavation and ground support methods. For brittle rocks, in general, careful excavation by smooth-blasting or machine methods is required; and where the rock is not self-supporting, support should be installed close behind the working face so as to prevent localised rock fracture, recognising that the support system will of necessity be over-designed by conventional standards. Bulk fill materials in these rocks should ideally have significant swelling properties to avoid the occurrence of planes of separation, and must be capable of sustaining the full internal support pressure without significant deformation.

In plastic clays, continuous support using relatively massive linings will be necessary during the waste emplacement phase. Bulk fills having similar properties to those required for brittle rocks could then be placed within the lined excavations. However, recognising the fact that the linings themselves may constitute a potential flow path in the long term, removal or controlled de-stressing of linings, with simultaneous backfilling, should be considered.

In saliferous rocks, careful excavation by smooth-blasting or machine excavation should also be employed, in recognition of the fact that high rates of loading can induce a brittle, elastic type failure, accompanied by localised fracture development; see figure 31. Openings should be designed so as to
accommodate a steady-state visco-elastic creep, generally without internal support during the waste emplacement phase. Non water-bearing fills should be designed to completely fill the excavations and to undergo a long-term ground-fill interaction, leading to the development of full-overburden support pressure.

Requirements for eliminating longitudinal flow paths in shafts are no less onerous, in view of the special problems they pose; see 10.3 above. In general, shafts must be lined and the criteria for linings and backfills will be broadly similar to those described above, depending upon the nature of the overburden materials. Ground treatment in unstable strata must be approached with considerable care. This applies particularly to the use of ground-freezing techniques, since the freeze-thaw process may induce ground disturbance in unconsolidated materials due to the expansion-contraction process; see Chapter 16.5.

It is apparent from the above discussion that uncertainties inherent in the containment properties of undisturbed host rocks are compounded by problems of repository construction which could lead to significant additional uncertainty and loss of containment. As will be shown in Part 3, it appears that inadequate attention has been focussed on means of mitigating against these problems in the international repository design proposals put forward to date. In Part 4, some specific design aspects are examined in detail, in order to determine what scope exists for exercising greater engineering control on the level of containment afforded by high-activity waste repositories.
PART 3
REPOSITORY SYSTEMS
11. OVERVIEW

Part 2 has shown that the underground disposal of high-activity radioactive wastes relies primarily on the natural physical and chemical properties of the selected underground environment to prevent significant contamination of the biosphere. It has also been shown that the repository itself is an engineered structure which must be designed to achieve waste emplacement objectives with minimal detriment to host rock containment properties.

However, the design of a repository should be viewed as one aspect in the development of a fully optimised waste disposal system. Thus, ground engineering objectives will be constrained by factors arising from pre-disposal strategies for the management and conditioning of the wastes; including immobilisation, packaging, shielding and interim storage.

The production of radioactive wastes is a continuous process and the repository must be able to receive wastes delivered over a significant period of production, typically from twenty to thirty years, during which the rate of waste delivery may vary considerably. The various stages in the disposal sequence (repository construction, waste emplacement and backfilling) are strongly inter-dependent, and may even be planned so as to take place concurrently in different parts of the repository. The extent of the interaction between these various activities is clearly an important practical factor which influences the design of the facility as a whole.

Significantly, the majority of the international conceptual repository design studies to date have been
based on the assumption of a self-supporting host rock, presenting few difficulties in terms of underground construction. These comprise the relatively strong granitic and saliferous rocks which remain competent, in the engineering sense, even at great depths; see figure 28.

Indeed, it has appeared to the author that even for this relatively restricted range of materials, many of the generalised assumptions concerning host rock characteristics which have been used as a basis for repository design studies have grossly over-simplified the problems of construction and containment (8, 34). Chapter 12 below describes the various repository design concepts which have been advocated, based on a review of the available literature, and indicates an emerging concensus of international opinion concerning the preferred methods of repository design and construction.

Chapter 13 provides a critical examination of recently published conceptual design proposals, including relevant aspects of waste unit design and pre-conditioning. In order to limit the scope of this account, and to coincide with the author's own background experience in this field (34), Chapter 13 emphasises concepts developed for crystalline (granitic) host rocks. However, certain aspects of the international proposals for repository design in argillaceous and saliferous host rocks are also briefly reviewed for comparative purposes.
12. DESIGN CONCEPTS

12.1 Introduction

The selection criteria for a repository host rock formation and the wide range of properties which must be considered have been described in Part 2.

In attempting to devise the most appropriate underground repository system, the available options are influenced by:

- the form, characteristics and volume of the waste materials involved;
- host rock properties;
- site-specific geological, hydrological, topographical and demographic constraints;
- the availability of suitable underground construction and waste emplacement technology;
- short-term considerations of economy, security and environmental impact;

Clearly, there can be no universal or standard repository design, and the disposal system must be tailored to suit a combination of circumstances which prevail in each region or country concerned. Nevertheless, it is possible, in the first analysis to categorise the various repository design concepts which have been advocated independently from their country of origin and associated geological or industrial constraints.
The various types of underground repository system which have been proposed comprise the following (9):

- very deep drillholes constructed from the surface;
- matrix of drillholes constructed from the surface;
- exploded cavities;
- mined caverns;
- tunnel networks with 'in-room' emplacement;
- tunnel networks with 'in-floor' emplacement;

All of the above have been studied in varying levels of detail over the past twenty to thirty years. However, it must be noted that early research efforts were almost exclusively concentrated upon saliferous and hard crystalline host rocks. Argillaceous rocks, particularly unindurated plastic clay formations, have only recently been seriously considered as potential repository host media.

Furthermore, the majority of detailed conceptual repository design studies have been concerned with high-level wastes; since these have been regarded as posing the greatest technological disposal problems. The adoption of crystalline and saliferous rocks as the preferred host media for these wastes may be attributed to a number of reasons, as follows:

- the vertical extent of the host material and the range of emplacement depth options is generally greater by comparison with sedimentary strata;
particularly clays.

- The relatively high thermal conductivity of crystalline and saliferous rocks enables solidified high-level waste units to dissipate their heat more rapidly over a relatively large volume of rock by comparison with other rock types; so reducing the size of the repository required; see table 9.

- The relatively low melting point of crystalline rocks has allowed rock-melting concepts to be considered as an alternative to solid waste emplacement. Thus, it has been suggested that emplacement of liquid reprocessing wastes in crystalline rocks could result in their conversion to a more stable form within the ground itself; since in the absence of artificial cooling or dilution, the high heat output of the waste would melt the surrounding rock (51, 207). For an acidic rock of granitic composition, melting could occur as the local temperature approaches 700-900°C. The initial rock melt should then increase in size with the continued output of heat and further addition of waste. With subsequent reduction in waste heat output, over many years, the rate of heat dissipation would eventually increase in relation to the rate of heat production and the molten rock-waste matrix would gradually solidify in situ as a stable and inert igneous inclusion.

- The assumption of a strong, competent host rock material minimises the engineering problems associated with all forms of underground construction; enabling repository schemes to be developed which are based on well-established and relatively inexpensive hard rock tunnelling or mining methods. As will be shown later, the evidence
suggests that the nuclear establishment has generally been unwilling to promote concepts which rely on extensive development of new underground design or construction techniques.

These various factors will be considered in detail in relation to design proposals which have been put forward. However, it is sufficient for present purposes to note that the repository concepts described above evolved at a relatively early stage, from consideration of requirements for high-level waste disposal in strong, competent host rocks. As will be shown, the range of repository design options is reduced, and new constraints are imposed when weaker host materials are considered and where intermediate-level wastes are included.

The remaining sections of this chapter review the essential features of the various underground disposal concepts which have been put forward; highlighting the potential advantages and disadvantages of each system. The emphasis is placed on crystalline host rocks, although the applicability of each system to saliferous and argillaceous strata is also described. Finally, section 12.7 provides an overall comparison between the various concepts, and indicates an emerging consensus of international opinion concerning the preferred form and layout of repository systems and methods of waste conditioning and emplacement.

12.2 Very Deep Drillholes

The very deep drillhole concept has been advocated specifically for the disposal of high-level wastes in crystalline rocks. It involves the emplacement of wastes within the lower portion of one or more very deep
drillholes, and the sealing of the upper portion, above a pre-determined level. The principal objective is to attain the greatest possible depth of burial; thereby imposing a more substantial geological barrier to waste migration than could be achieved by tunnelling or mining methods.

'How deep is deep enough' is a question which cannot be answered on a non-site specific basis, since the effectiveness of the geological barrier is strongly dependent upon geological and hydrogeological conditions at a particular site. However, advocates of the concept have generally proposed that holes should be drilled to the maximum depths which can be achieved using currently available or readily developed technology. It has been suggested that drillhole depths could exceed 10km, and the thickness of backfill materials overlying the waste could exceed 2-3km.

The very deep drillhole concept is discussed in a comprehensive American review of radioactive waste disposal alternatives by Schneider and Platt (194), and to a lesser extent in the more recent literature (158, 184, 207). It is a largely unproven technique, which is dependent upon the outcome of considerable research and development in the field of deep drilling and waste handling, but offers the following potential advantages:-

- alternative design options are available involving emplacement of wastes in either liquid or solid form, with or without rock melting;
- in the event that the design concept involves rock melting, the waste can be disposed of without provision for a period of interim cooling;
the great depth of burial virtually eliminates any long-term risk of removal of overburden, either by natural erosion or by catastrophic events;

- very long groundwater migration paths are created by disposal at great depths;

- assuming a general decrease in the frequency and size of discontinuities in indurated host rocks with increasing depth, it is probable that ambient insitu rock mass permeabilities will be extremely low;

- a greater choice of sites is available, since the choice of location is less dependent upon the near-surface geological formation than for other methods of disposal.

Equipment and techniques for the drilling of deep drillholes have been developed mainly within the oil industry, although technical advances were also made during the drilling of deep, large-diameter holes for the underground testing of nuclear weapons in the tuffaceous rocks of the Nevada desert in the United States, and in other formations elsewhere (77, 158, 194). The maximum recorded drillhole depth for oil exploration is 9.16km, with a drillhole base diameter of 0.2m; although several holes have been drilled to depths greater than 6km (158). However, penetration in excess of 10km, at diameters sufficient to allow emplacement of solid waste units, would be dependent upon considerable research and development.

Schneider and Platt (194) have suggested that new equipment could be developed, broadly similar in design
to that used in the oil industry, in order to reach the required depths. They have endorsed a claim that a 0.7m diameter hole could be drilled to a depth of 16km, by adapting and developing conventional oil-drilling rigs. This would involve an increase in power output, drill-table capacity (torque) and drill-rod strength, and modifications in the design of drilling bits and pumps. However, they have also noted that the handling of borehole casings and drillstrings could impose practical restrictions and would probably limit the maximum attainable depth to about 12km.

O'Brien et al (158) noted that, at the end of 1979, four rigs were available in the United States which were capable of drilling to depths of 9km or somewhat deeper, and that geothermal wells drilled to relatively shallow depths (of the order of 3km) had successfully overcome problems associated with drilling in rocks with ambient geothermal temperatures as high as 300°C. They concluded, rather more conservatively than Schneider and Platt, that hole depths of about 11km could be achieved provided that borehole stability is not a problem and the final hole diameter can be reduced to 0.20m diameter. As an alternative, they suggested that fully-cased holes could be drilled to a depth of 9km, with a minimum hole diameter of 0.3m.

In practice, however, the majority of deep hole drilling experience is within sedimentary strata, less hard and less abrasive than the crystalline host rocks envisaged. Predictions of deep hole drilling capabilities in hard crystalline rocks are therefore largely based on an extrapolation of current drilling experience in other strata.

In granitic rocks, a relatively high speed of rotation
would be necessary to advance a drill-hole at a reasonable rate of progress (approximately 500-700 r.p.m.). Equipment such as the 'turbodrill', where the rotative effort is supplied at the base of the hole, rather than by the relatively slow rotation of a drill-table at the surface, would therefore probably be appropriate. However, frequent 'round trips' (involving complete withdrawal of the drillstring and replacement of the bit at the base of the hole) would be necessary for each unit of advance, due to rapid bit wear. A high frequency of 'round trips' at great depth would involve considerable risk of drill-string failure.

The high temperatures associated with very deep drilling impose added difficulties. In the U.K., areas of normal geothermal gradient imply average ambient rock temperatures ranging from approximately 300 to 400°C at a depth of 10km (87). Very dense drilling muds are required for the high temperatures and pressures associated with drilling at such depths; but even high temperature fluids generally become unstable at temperatures approaching 300°C, leading ultimately to drilling bit failure.

It is evident from the above that considerable research and development would be required in order to form and maintain holes of the required diameter to depths of 10km or more; although more modest depths of say 5km may be more readily attainable. The research and development work required to advance holes to great depths in granitic rocks would probably involve the practical advancement of novel rock cutting techniques such as high-energy water jets, thermal spalling, fusion and vaporisation drills, etc. Furthermore, the time required to drill deep holes is significant, and must be added to the time needed for the necessary development
work. Average drilling rates are difficult to estimate, but current deep drilling experience suggests that drilling times and costs would increase exponentially with depth. Assuming an optimistic average drilling rate ranging from 5 to 10m per day, the total drilling time for a 10km hole would then be at least 3 to 6 years (194).

Assuming that problems associated with the construction of deep drillholes can be overcome, waste could be emplaced in either solid or liquid form. Whichever method is adopted, borehole casings would have to be cemented in place to the maximum possible depth to ensure side-wall stability. Venting arrangements would be required at the top of the hole in order to allow release of gas or vapour associated with the high temperatures at depth and to prevent the build-up of excessive 'in-hole' pressures.

For liquid waste emplacement, the behaviour of the waste fluid and waste-rock interactions would require careful evaluation. Following reprocessing of the spent fuel, the liquid waste stream would comprise an acidic aqueous solution; see Chapter 4.2. After emplacement at depth, and in the absence of artificial cooling, the liquid waste would boil, releasing steam and other volatile material. Continued production of heat, combined with the addition of more liquid waste would result in increasing sub-surface temperatures, leading ultimately to rock melting.

Published outline design concepts envisage a degree of control over down-the-hole waste temperatures, by addition and re-circulation of water as a coolant during the emplacement period (51, 194, 207). Surface facilities would be provided for condensing the steam
and re-cycling cooled water to the emplacement hole as indicated in figure 35. Non-heat producing intermediate and low-level wastes could also be added to provide a cooling effect. It should be noted, however, that the vapours and gases released to the surface would include volatile radioactive materials requiring separate treatment.

With the continued addition of waste and the release of volatile material, the emplaced waste would become progressively drier and hotter with depth; leading to the formation of molten rock-waste 'magma'. The aqueous portion of the waste column would have to be maintained at a controlled level by adjusting the rate of emplacement, so as to prevent surging of material towards the surface.

After emplacement of waste to the required level, time would be required for the contents of the upper portion of the material to solidify. The remainder of the hole could then be backfilled with a suitable sealant mixture.

For placement of liquid high-level waste, the site would ideally be located at the reprocessing plant, so as to avoid transporting the waste in mobile form. Placement of solid waste packages, if practicable, would provide greater flexibility. In this event, the process could be engineered so as to induce rock-melting, as in the case of liquid waste emplacement. Alternatively, rock melting could be avoided, either by diluting the waste, reducing the size of the waste packages, or adjusting the vertical distribution of packages within the hole. In this case, the borosolicate glass composites generally proposed would be unsuitable waste forms; since temperatures at depth would cause devitrification...
or even melting; see Chapter 9.3. It would therefore be necessary to use 'Synroc' or some other ceramic material with greater thermal stability (59, 185). However, whatever the waste form, the non-melting case obviously implies a reduction in waste disposal capacity of the drillhole; see figure 35a.

For the hole depths and diameters envisaged, solid waste packages would have to be relatively long and thin. For a 0.3m diameter hole, for example, the maximum practicable canister diameter would be of the order of 0.2m. O'Brien et al (158) have suggested that canister lengths up to 9.1m (30 feet) would be most convenient; since this corresponds to the length of standard U.S. drill pipe. However, in practice, problems associated with waste handling and shielding at the surface would probably require the use of shorter lengths.

In the non-melting case, it is difficult to envisage how adequate waste-rock thermal contact could be established to provide for adequate heat conduction into the rock mass in the short term. At the depths envisaged, the only viable technique for providing a filling of the annulus around the waste packages probably involves the emplacement of wastes so as to displace a cementitious slurry, which subsequently hardens around the units. Such a measure would also seem desirable to prevent the development of over-stressing of the canisters due to the self-weight of the waste column and, in practice, structural platforms comprising suitably rigid backfill material would probably be required between successive units. However, although the problems associated with waste emplacement and backfilling at great depths appear formidable, they do not appear to have been considered in the available literature.
The concept of emplacing liquid high-level wastes in deep holes so as to induce rock-melting involves additional risk, since the increase in volume which accompanies rock-melting may result in fracture of the surrounding rock. Ingress of water as a result of such fracturing, with intersection of water-bearing discontinuities, could lead to considerable vapour release and possible geysering. In addition, where relatively dry conditions occur at depth, it would be necessary to ensure careful cementation of the casing to prevent ingress of water to the emplacement zone from higher level aquifers (see figure 35b).

Further unresolved engineering and operational difficulties must be considered, namely:

- Whichever waste form is adopted, the integrity of the hole must be maintained throughout the duration of the waste emplacement programme (assuming that a single hole will accommodate all generated waste).
- In practice, it would be impossible to guarantee the long-term integrity of uncased sections.
- Short-term retrieval must be regarded as impracticable for either solid or liquid wastes. For solid waste packages, severe difficulties would occur in the event of packages becoming jammed before reaching their emplacement depth.
- For liquid waste emplacement, it would be necessary to install independent pipework for waste delivery, release of gas and return of condensate. This pipework would be subject to high temperature and pressure differences, and would require a virtually maintenance-free design covering an exceptionally
long operational life; extending considerably beyond the total period required for waste emplacement.

- Comprehensive ground investigations, involving direct methods of exploration, would probably be impracticable and therefore considerable reliance would have to be placed upon geophysical techniques. These methods involve considerable difficulties in interpretation, and the possibility that a deep hole may have to be abandoned, due to adverse ground conditions, therefore presents a considerable risk.

- The design of an effective hole sealing system must provide positive assurance of its ability to contain any gas pressures developed at the end of the emplacement phase, and to prevent the preferential upward flow of contaminated groundwater. If casing is left in place, it must therefore be designed as an integral component of the sealing system, and account must be taken of the effects of corrosion of the casing upon the potential for radionuclide migration. If the casing were to be removed, even over short lengths, it would be necessary to provide assurance that the hole would remain stable during the sealing process.

In view of the various uncertainties and engineering difficulties described above, the deep drillhole concept is seen to pose very severe practical development problems. These have remained largely unresolved, and little known recent research effort has been expended in developing the original idea.
12.3 Matrix of Surface Drillholes

The construction of a repository comprising a matrix of emplacement holes drilled to relatively moderate depths (of up to about 3km) is discussed in some recent comparative studies (67, 194). The concept is a modification of the deep drillhole method, utilising shallower drillhole depths, more compatible with the present state of the art; although still beyond those normally envisaged in tunnelling or mining practice. The system involves the drilling of an array of vertical holes, and the emplacement of solid waste units in the lower portion to achieve isolation from the biosphere.

Like the 'deep hole' concept, it is envisaged that only high-level wastes would be disposed of in this way. However, rock-melting methods are not discussed in the literature, and it is generally assumed that a solid waste form would be adopted. The essential features of the system are illustrated in figure 36. The borehole grid and vertical distribution of the solid waste units would be determined so as to achieve dissipation of heat without mineralogical alteration or structural disturbance within the surrounding rock mass.

The principal features of the matrix of drillholes concept are similar to those described for the deep hole method. Firstly, all construction and emplacement operations would be carried out from the surface; although most of the operating facilities could be contained within underground 'cut and cover' structure if required. Secondly, the solid waste packages are emplaced as far as practicable below the ground surface, with minimal disturbance of the surrounding rock mass during excavation (since drilling methods generally
result in less disturbance to the surrounding rock than other excavation techniques).

Deep drilling, at diameters in excess of 1.0m. has been achieved using special equipment, (notably for nuclear weapons testing at the Nevada test site). The maximum depth of drilling for a given site would be determined, inter alia, by consideration of borehole stability, ambient geothermal temperatures, and an optimisation of costs in relation to the depth of drilling. This in turn would determine the number of holes required in the repository. It is clear, however, that increasing the number of emplacement holes would also increase the number of potential vertical migration paths for radionuclide release to the biosphere; see Chapter 10.3.

The choice of site would be influenced by surface geology and topography to a greater extent than for the deep hole concept, and would require a relatively large and level site. Due to thermal interaction, it would also be necessary to make allowances for possible drillhole deviations in evaluating the optimum borehole spacing.

Figure 36(a) illustrates a possible arrangement for surface handling and emplacement of waste canisters into an array of drillholes. Boreholes would be lined as necessary, and would certainly require lining in the upper sections to prevent ingress of water from strata penetrated at higher levels during drilling. It is envisaged that waste units would be emplaced to within approximately 1km or less of the surface and the upper sections of the holes would be backfilled with a low permeability sealing mixture.
As with the very deep drillhole concept, the system offers economy in excavation combined with a large depth of burial; larger than might be envisaged using conventional tunnelling or mining methods. It may be considered for both crystalline and saliferous rock formations; but since the wastes are to be emplaced in long vertical columns, it is seen to be generally unsuitable for argillaceous rocks, whose vertical extent is relatively limited.

The following advantages are apparent in comparison with the construction of a single very deep hole:

- Direct methods of site investigation up to the emplacement depth are feasible, with the added possibility of utilising cross-hole geophysics as a means of extending the level of ground information between emplacement holes,

- A number of drilling rigs could be used simultaneously, so decreasing overall construction time,

- The loss of a single emplacement hole, through breakage or jamming of equipment, collapse or movement of borehole walls, etc, would be less catastrophic.

However, despite its apparent advantages in terms of simplicity, the method does share certain disadvantages in common with the deep hole alternative. These are:

- Jamming of canisters during placement could prevent full use of emplacement holes,
short-term retrieval of waste units would probably be impracticable,

it would be difficult to ensure borehole stability during waste emplacement and backfilling operations.

In addition to the above, it is apparent that a relatively large number of penetrations of the host rock are required, each of which would constitute a potential pathway for migration of wastes to the biosphere. The latter generally confirms that the concept must be deemed inappropriate for the disposal of intermediate-level wastes; since large volumes of material are involved. For similar reasons, the concept is unlikely to prove attractive for the disposal of high-level wastes in countries where waste arisings are large.

For high-level wastes, logic would also suggest that saliferous rocks represent the preferred host material, due to their relatively favourable thermal properties. At depths of the order of 2 km, geothermal temperature could be as high as 60°C to 80°C; see figure 22. Since the thermal conductivity of saliferous rocks is approximately twice that of the crystalline types, saliferous rocks would be better able to accommodate heat generating wastes in an end-to-end vertical stacking configuration, without causing a rock temperature increase which could cause devitrification. If crystalline rocks were to be used as the host material, it may prove necessary to utilise an immobilising material of higher thermal stability, such as 'Synroc'; as in the case of the very deep drillhole concept.
Although the general reasoning outlined above is not found in the available literature, it would appear that the conclusions are borne out in practice. Denmark is the only country to have developed detailed conceptual engineering proposals for radioactive waste disposal using a matrix of surface drillholes concept (67). The facility is intended for the disposal of high-level radioactive wastes in saline rock, and it is notable that Denmark has the lowest predicted volume of radioactive waste arisings of all the countries actively engaged in underground waste disposal research; see Chapter 13.

12.4 Exploded Cavities

Some consideration has been given to the formation of deep underground repositories by means of conventional or nuclear explosives. The application of this technique was considered by workers in the United States in the early 1970's, based upon experience gained during underground nuclear weapons testing at the Nevada site (66, 77, 194).

The potential advantages which have been claimed are largely economic, since large capacity, deep-level cavities may be created using an explosive device at relatively low cost, when compared with deep drillholes or mined repositories. Furthermore, an exploded cavity may be formed with relative ease at depths beyond conventional tunnelling experience.

However, from an engineering viewpoint, the concept appears less attractive. An underground explosion produces a rubble-filled cavity with a peripheral zone of fused and damaged rock having high fracture porosity.
and mass permeability (77). It would therefore be necessary to ensure that a substantial boundary layer of sound rock was available to provide the necessary geological confinement outside the affected area.

Therefore, the concept is applicable only to liquid waste emplacement, leading to rock-melting. In theory, the large contact area presented by the broken rock within the cavity would rapidly lead to the incorporation of a considerable volume of rock within the final rock-waste matrix, as shown in figure 37. However, peripheral fractures could also carry liquid waste or rock/waste 'melt' into surrounding country rocks, which may be relatively permeable. In addition, the increased rock fracture could result in induced peripheral flow paths, causing groundwater invasion into the emplacement area, accompanied by the generation of high-pressure steam and possible geysering.

For these reasons, the exploded cavity concept is clearly applicable only to regions where very wide and deep-seated plutonic rock masses are known to occur, or to desert regions above the water-table. Furthermore, the site would clearly have to be a very remote one, as is the case with the Nevada test site.

As with the deep drillhole concepts, liquid emplacement would require the provision of a pipework system for waste delivery, vapour release and return of condensate. The latter would be difficult to install in the original access hole and may require drilling of separate holes after the construction of the main cavity; figure 37b.

The exploded cavity concept is generally not regarded as a serious contender in comparison with other proposals, and is not widely discussed in the more recent
literature. Apart from the engineering drawbacks outlined above, obvious political, environmental and social factors suggest that the method is generally unacceptable and therefore not worthy of detailed technical consideration.

12.5 Mined Caverns

For the purpose of the present discussion, a mined cavern repository may comprise either a purpose-built underground disposal system, or a disused mine which has been specially adapted. In common with repository systems based on construction of tunnel networks, it is envisaged that the caverns would be located at depths of several hundred metres below ground level. The term 'cavern' implies relatively large openings having roof spans typically in excess of 15m, and possibly up to 40m, depending on the properties of the host rock. Thus, the concept is only applicable to relatively competent rocks, and would not be a viable solution in relatively weak argillaceous rocks and clays.

The mined cavern concept enables large quantities of waste to be emplaced within a relatively compact repository system. However, because of this, the characteristics of high-level and intermediate-level wastes impose different constraints upon the design of the system.

12.5.1 High-Level Waste Disposal in Mined Caverns

The concept of using mined caverns for high-level waste disposal has found very few committed advocates. Three reasons are suggested to account for this.
Firstly, the construction of a cavern, as opposed to a tunnel, results in a relatively large excavated volume per unit surface area of exposed rock. As a consequence, filling such a cavity with heat-generating wastes creates a relatively high heat flux at the waste/rock interface. Therefore, unless rock-melting is intended, a more dispersed disposal system is to be preferred in which a relatively large surface area of rock is available for heat removal per unit volume of emplaced waste.

Secondly, since cavern dimensions are relatively large, the magnitudes of elastic displacements induced by excavation are greater, with a corresponding increase in the tendency for loosening and inward movement along discontinuities. This situation would be exacerbated by thermal expansion effects, due to high induced rock temperature gradients in the near-field. In consequence, an extensive fractured zone could develop around the periphery of the cavern, providing a pathway for radionuclide transport. Although it may be possible to prevent detachment of loosened material and to ensure long-term structural integrity of the excavation during the emplacement phase, by provision of artificial support, the propensity for relatively extensive fracture development in brittle rocks is an obvious disadvantage.

The third difficulty concerns the mechanics of waste emplacement. If the waste is in solid form, emplacement in an ordered three-dimensional array would require the provision of a complex and costly internal structure within the cavern. Yet such an arrangement would be necessary to ensure a uniform heat flow, and to allow placement of backfill to provide good waste/rock thermal
contact. During the period required for waste emplacement, it may be seen that the combination of radiological hazards and heat production would create severe operational difficulties, requiring complex shielding and interim cooling arrangements.

In view of these difficulties, all proposals involving the use of mined caverns, for high-level waste disposal must involve either rock-melting, prolonged interim cooling or emplacement of solid waste having very low fission product concentration per unit volume of waste material. However, each of these expedients poses a number of technical and economic difficulties.

Figure 38 illustrates some of the design features involved in the rock-melting concept, in which the cavern is lined with stainless steel or other high integrity material. Liquid waste may be introduced and artificially cooled, by the re-circulation of condensate and introduction of additional liquid to provide dilution. The steel lining would thus fulfill a function similar to the surface storage tanks currently in use at reprocessing plants (see chapter 4.2) although the problems of providing maintenance-free cooling would be considerably greater. Cooling of the liquid waste would continue until the maximum emplacement volume is reached, or possibly longer. Thereafter, cooling would be terminated and the waste temperature allowed to increase; ultimately causing the cavity lining to fail, with subsequent melting of the surrounding rock followed by solidification to form an insitu rock-waste matrix.

Alternatively, waste may be emplaced in solid form. In this case, interim cooling would be provided by circulating a fluid around the canisters. The fluid
would remain uncontaminated by radionuclides, provided the waste containers do not fail. On completion of waste emplacement, the cooling process may be terminated, leading ultimately to failure of waste containers and rock-melting, as in the case of liquid waste emplacement.

In common with other concepts involving rock melting, a two-pipe system would be required for delivery of waste and coolant and discharge of steam and gas; together with surface facilities for re-circulating the condensate and treatment of non-condensible components.

In such a system, it is postulated that by maintaining the cavern at atmospheric pressure, groundwater flow within the rock would be directed into rather than out from the waste emplacement chamber (207). At some stage, the inlet hole would then be sealed and the liquid waste allowed to boil dry. In the absence of further addition of water, rock melting would occur and, with increase in the size of the 'magma', the corresponding increase in rock contact area would lead to a higher rate of conductive heat loss. Radioactive decay would also diminish the level of total heat output, and the rock/waste 'magma' would begin to solidify when the rate of heat loss into the ground exceeds the level of heat output.

It has been suggested by Cohen et al (51, 207) that provided the outlet pipe is kept open during the rock melting and solidification phase, the 'magma' would remain immobile. However, this assumes minimal fluid migration in the surrounding rock and the absence of significant gas production as a result of heating and melting the rock. It is also postulated, in apparent self-contradiction, that the heat generated within the...
emplacement zone would keep groundwater at bay. However, this notion seems unjustifiable, since whilst the cavern is vented to the atmosphere further steam is likely to be generated from groundwater in the vicinity of the cavern, accompanied by continued groundwater replenishment from lower levels by thermal convection.

It has been calculated (207) that a reprocessing plant handling 1500 MT of spent fuel per year would require a cavern with 6000m$^3$ liquid high-level waste capacity over a period of 25 years. On terminating the circulation of coolant, the rock/waste magma would develop to its maximum size after about 65 years, and would solidify completely after approximately 200 years.

The foregoing proposals are seen to pose numerous technical difficulties and uncertainties. If rock-melting is seen as a viable waste-disposal concept, the placement of solid waste, with temporary cooling during the emplacement phase, would appear to provide the greatest degree of control. However, in the author's view, the rock-melting methods are particularly unsatisfactory when applied to mined cavern concepts, as opposed to borehole concepts. This is because the volume of the solidified rock/waste matrix would inevitably be much less than the equivalent volume of virgin rock originally present. Thus it must be anticipated that the solidified rock-waste composite would be surrounded by a rubble-filled void which would be extremely difficult if not impossible to seal.

The problems associated with the disposal of high-level wastes in mined caverns, without rock melting, are typified by the WP-cave system illustrated in figure 39. The WP-cave system (4) has been proposed by a Swedish Company, as an alternative to the 'KBS project'
established by Swedish nuclear authorities; see Chapter 13. The method involves the emplacement of solid waste within a cavern constructed in strong granitic rock, in a manner which does not induce rock-melting and does not require long-term cooling.

The underground disposal space within the WP-cave system comprises a 40m diameter spherical cavern with an 11m diameter cylindrical concrete 'heat stack', constructed at its centre. The space surrounding this heat stack is filled with plain 3.6m spherical concrete balls, and the whole mass therefore contains numerous interconnecting voids.

Cylindrical canisters of solidified waste are cast into similar concrete balls, placed in the lower half of the 'heat stack'. The upper half of the stack is then filled with additional plain concrete balls. It is postulated that after emplacement of the waste and plugging of the access shaft, a natural and continuous air convection through the 'heat stack' and concrete fill will dissipate the heat emitted by the waste; with uniform distribution over the interior surface of the cavern. However, it is not clear how the heat is to be dissipated during the short-term period whilst waste emplacement is in progress. Presumably, some form of mechanical ventilation is envisaged for this purpose, although no details are provided in the literature.

The system as described, is designed to remain structurally stable for up to approximately two hundred years, during which it is claimed that short-term retrieval would be possible. Thereafter, gradual structural deterioration inside the cavern is anticipated, with eventual cessation of the convection mechanism. This is claimed to be acceptable, on the
premise that at this time thermal output of the waste would be relatively low, and conduction mechanisms would be adequate to disperse the residual heat. Pressure grouting would be carried out at this stage, in order to fill the voids and provide continuous thermal contact.

A significant feature of the system is the proposed construction of a 5m thick bentonite-sand barrier to envelope the entire repository, in order to provide a high degree of confinement, both by inhibition of water flow and by chemical sorption of radionuclides. Excavation of spiralling tunnels, floor-excavation, and simultaneous backfilling is intended to provide a continuous bentonite 'blanket' as shown in figure 39. The provision of this barrier is claimed to allow use of the system in a wide variety of ground conditions, without risk of environmental contamination. Such a claim appears very naive in view of the cavern dimensions envisaged. However, it is less so when viewed exclusively in the Swedish context, since the entire country is underlain by Pre-Cambrian granites and gneisses of the Fenno-Scandinavian Shield. The range of ground conditions envisaged, in practice, is therefore far from 'wide'.

The construction of an extensive bentonite envelope around the repository appears highly attractive in terms of waste containment. However, in practice, construction of the barrier would cut free an enormous volume of rock to bear directly upon compressible bentonite clay fill at the base and sides. Extensive fissuring, rock-bursting and tunnel collapse may be anticipated during the latter stages of this construction; with large mass-movement and squeezing out of the backfill. In the author's opinion, the possibility of actually constructing the bentonite
barrier to achieve the desired effect is therefore remote; irrespective of the ground conditions.

For these reasons alone, the WP-cave system does not appear to warrant serious practical consideration. However, more fundamentally, it should be noted that the volume of waste which could be incorporated in the cavern is equivalent to only one year's nuclear energy production in Sweden. This is due to the effective dilution of the solid waste inside the central cavern, by the inclusion of numerous non-waste bearing concrete balls in order to dissipate the total heat output. The volume and complexity of excavation and backfilling required to achieve this level of disposal is clearly disproportionate; and several similar repositories would be required to accommodate the total national waste arisings over a prolonged period of production; involving the selection of numerous sites. This aspect serves to illustrate the general disadvantage which applies to any concept involving cavern disposal of solidified high-level wastes.

In view of the foregoing discussion, mined cavern concepts are considered unlikely to meet the overall technical and economic requirements for high-level radioactive waste disposal, except where an extremely long cooling period can be provided to reduce heat outputs to minimal levels prior to disposal. In this case, the problems associated with disposal become more or less identical to those for intermediate-level wastes, as described in 12.5.2 below. However, mined caverns are undoubtedly worthy of consideration for provision of temporary underground storage, with artificial cooling, in lieu of surface storage vaults. Here, the concept of disposal does not arise, and the construction of an underground facility is intended simply to provide
security against sabotage or intrusion rather than permanent isolation from the environment.

12.5.2 Intermediate-Level Waste Disposal

The problems associated with heat dissipation in large caverns, as outlined above, do not arise in the case of intermediate-level wastes. Indeed, mined cavern concepts have appeared particularly attractive in view of the relatively large volumes of intermediate wastes foreseen.

However, despite the obvious economic advantages of achieving a high emplacement density (expressed in terms of excavated rock volume), the author considers that there are a number of operational and technical drawbacks to the concept of using large caverns for intermediate-level waste repositories.

The operational difficulties are due to radiological problems during the waste emplacement process, and the subsequent backfilling of the caverns. At intermediate stages of emplacement (prior to backfilling), it is evident that a large cavern full of waste units presents a relatively high direct radiation hazard to workers entering the emplacement area. Since workers are exposed to a larger number of units over a longer period, the chances of receiving radiation from a defective waste unit, for example, must be increased. In relation to backfilling, experience also suggests that there would be difficulties in ensuring proper penetration of backfilling and sealing materials around a large stack of units by comparison with a small one. This particular aspect of intermediate-level waste repository design will be considered in greater detail
As previously described, the use of large excavations in brittle rocks could be detrimental in terms of radionuclide migration, due to the development of a relatively large zone of disturbance. However, for a high-density emplacement configuration of intermediate-level waste units, it is also evident that where one or more induced pathways exist around the excavation, the volume of waste which may act as a source of contamination increases in proportion to the square of the mean excavation diameter. For randomly spaced natural discontinuities, the probability that the excavation will intersect a given number of transmissive pathways also increases in relation to the transverse dimension of the openings. Economies gained by increase in the size of openings (to reach cavern proportions), may therefore be offset by increasing the probability of loss of effective containment by the host rock.

Notwithstanding size considerations, disused mines have attracted considerable interest as intermediate-level waste disposal sites. Two important advantages are apparent in the selection of a disused mine. Firstly there is an obvious economic advantage in taking over a site where all necessary underground excavation work has been completed. Secondly, site investigation requirements are minimised; since the geological and hydrological characteristics of the formation are generally well-established prior to site selection. The technical advantages of the latter are self-evident and should not be undervalued. However, from the promotor's viewpoint, the avoidance of site investigations may also be a significant political factor, since it could reduce the amount of public consultation required during the site selection and licensing procedures. To the
jaundiced eye, it may be envisaged that this factor could lead to the expedient choice of a disused mine for largely non-technical reasons.

Thus far, disused mines have been identified as potential intermediate-level waste repository sites in France, Germany and Britain. The Asse salt mine in Germany has already been used as a disposal site for intermediate-level wastes, and in Britain and France, it has been made clear by nuclear authorities that they intend to establish industrial-scale disposal facilities within disused mines in the near future. Although detailed proposals for the development of these sites have not yet been put forward, the author has a number of reservations concerning the use of disused mines which may be described in a general way.

Firstly, it is apparent that many more disused mines are available in salt formations than in any other potentially suitable host materials under consideration. This is simple prima facie evidence that salt formations are regarded as an attractive mineral resource. Yet according to the general principles of geological disposal, risks of inadvertent human intrusion should be minimised in selecting the disposal site. It would therefore be contrary to declared intentions to select a disused salt mine simply 'because it is there.'

Secondly, although there are likely to be relatively few disused mines available in the crystalline or argillaceous rocks envisaged, due to their comparatively low mineral resource value, the selection of a disused mine in these formations could have adverse consequences in terms of waste containment.

Problems associated with peripheral disturbance in
brittle rocks have already been described, and it is
noteable that mining practices do not generally strive
to minimise this aspect. However, a further
consideration is the effect of a mine on the chemical
environment. As shown in Chapter 8.2, deep groundwaters
in low permeability rocks tend to be characterised by a
chemically reducing environment. Clearly, the presence
of an extensive mine over appreciable periods, in
water-bearing rocks, creates a drainage condition which
influences ground water movement on a large scale and
may lead to relatively oxidising conditions in the near-
field.

In addition, the size and layout of mines is dictated by
the logistics and economics of the original ore-winning
exercise. The concept of avoiding unnecessary
disturbance to the host rock and avoiding fractured
zones implies that minimisation of repository size, and
careful orientation with respect to fracture geometry,
are pre-requisite factors which should be taken into
account at the design stage and are therefore unlikely
to be embodied in an existing mine system; see Chapter
15.

In practice, the volume capacity of a dis-used mine
selected for repository development is likely to exceed
the available waste volume. However, as will be shown
in Part 4, the cost of effective backfilling may exceed
excavation costs by a factor of 4 or more; see Chapter
17.3. Hence, even the economic advantage claimed for the
selection of a dis-used mine could be invalidated in
many instances.
12.6 Tunnel Networks with In-room Emplacement

The 'in room' concept simply involves the emplacement of solidified radioactive waste units within an underground system of tunnels, which are subsequently backfilled to seal the repository. The resulting system differs from the mined cavern concept principally in terms of the size of the emplacement chambers. This in turn influences the repository layout, and most proposals envisage a series of long parallel emplacement tunnels linked by access tunnels; the whole system being connected to the surface by a number of shafts.

Because the openings are relatively small (typically 20m² to 50m² in cross-sectional area), the concept is applicable to all of the host rocks under consideration, and may be based on the use or adaptation of currently available sub-surface design and construction technology.

12.6.1 High-Level Waste Disposal

The 'in-room' system for high-level waste disposal involves the emplacement of waste units in a linear configuration on the tunnel floors. Figure 40 shows three possible emplacement systems; although numerous other minor variations have been proposed (17, 51, 66, 194, 195). Waste units could be placed directly on the tunnel floor at regular spacings, and either bolted or grouted in a pre-determined configuration. Alternatively, they could be placed horizontally at regular intervals on 'shelves'.

The minimum lateral spacings between canisters would be determined by the heat output of the waste at the time of disposal, and the thermal properties and maximum allowable temperatures in the proposed backfill and host
rock materials. Opportunity therefore exists for reducing canister spacings by allowing a period of interim cooling, in which the heat output of the waste is allowed to decay to a relatively low level, before backfilling takes place. This cooling could take place at a separate site, with backfilling shortly after emplacement in the final repository. Alternatively, canisters could be positioned in the final repository at the desired spacing, with provision for in situ cooling by forced circulation of air for a prescribed period prior to backfilling. It is apparent that some temporary in situ cooling by forced air circulation is necessary, unless waste units are to be immediately surrounded by backfill after emplacement.

Since it is generally envisaged that the tunnel network would be located at a single level, the repository would occupy a relatively large plan area in comparison with other disposal concepts. The 'in-room' tunnel emplacement technique may, in effect, be considered as a direct contrast to the concept of waste emplacement in a single deep drillhole, where the waste is distributed in a vertical rather than a horizontal mode.

On initial examination, the method appears to offer the advantage of great simplicity, and involves the use of well-proven conventional construction methods for creating underground space for waste disposal. In addition, by careful positioning of canisters within the tunnels, it would be possible to provide a substantial thickness of backfill to surround each unit by comparison with drill-hole emplacement methods. Opportunity therefore exists for the creation of major 'engineered' barriers to radionuclide migration by suitable formulation of tunnel backfill characteristics.
However, despite these apparent advantages, the method is open to fundamental criticism in that it poses several practical operational problems. Firstly, since the high-level waste units are positioned in an exposed fashion with the tunnels, all emplacement operations require the extensive use of remote handling or mobile radiation shielding equipment, as illustrated in figures 40a and 40c, due to the relatively high intensity of $\gamma$-radiation associated with high-level wastes. The same difficulty would occur in carrying out other operations within the emplacement tunnels, such as monitoring or retrieval of waste containers or completion of engineering remedial works prior to backfilling.

Such technical difficulties are not insurmountable, since the principles involved in the design and use of remote handling and shielding equipment are well-established in the nuclear industry. However, a substantial cost penalty is incurred as a result of the need for sophisticated mechanical handling and shielding equipment for a multiplicity of sub-surface operations.

Backfilling of the emplacement tunnels also presents serious technical problems. The use of remote-controlled equipment for backfilling could present difficulties in the event of a mechanical failure. The most suitable method therefore probably involves pumping a backfill through vertical stop-ends, followed by pressure grouting. However, if stop-ends are constructed at the tunnel extremities, the length of the emplacement tunnels would have to be short (say 30m) to provide the necessary confidence in the effectiveness of the operation. The only alternative is to place a series of stop-ends at regular intervals, with backfilling of each tunnel in a series of short sections as emplacement proceeds. Considerable conflict could
then occur between emplacement and backfilling operations.

The difficulties encountered due to the need for protection against radiation could be avoided by the provision of a fully-shielded overpack for each container. However, the high cost incurred by adopting this method could be prohibitive, both in terms of the unit cost of the overpacks themselves and the increased handling costs brought about by the additional weight of the overpack materials.

Further difficulties are associated with the ventilation of the repository during the period prior to backfilling, so as to remove the heat emitted by the wastes. A fail-safe ventilation system would be required throughout the operational life of the repository, with sufficient capacity to dissipate the heat from the maximum number of canisters exposed in the emplacement tunnels at any particular time. This requirement is additional to the ventilation necessary for all construction and emplacement operations.

12.6.2 Intermediate-Level Waste Disposal

Since intermediate-level wastes do not emit significant quantities of heat at the time of disposal, the problems of heat dissipation during the emplacement phase, as outlined in 12.5.1 above, do not arise. It is apparent also that the problems of radiation exposure to repository workers and possible reduction in containment properties, as described in 12.4.2 for disposal in caverns, are significantly reduced. The 'in-room' disposal concept therefore offers considerable advantages for intermediate-level waste disposal.
Most proposals envisage disposal of cylindrical or cuboidal waste units emplaced in the most dense stacking arrangement possible; see figure 40d. Waste emplacement and backfilling would generally take place in stages, either by backfilling and compacting fill over successive horizontal layers of units and then pumping fill into the crown area, or by pumping in and then pressure-grouting a fluid-form fill so as to fill the voids in a complete stack.

However, as will be shown in subsequent chapters, it appears that little detailed attention has been given to these aspects. The influence of waste unit design, package shielding, stacking configuration, and backfilling materials and procedures upon operational aspects and long-term containment performance are examined in Part 4.

12.7 Tunnel Networks with In-Floor Emplacement

The 'in-floor' concept is a combination of some of the ideas previously described. It is intended for the disposal of solidified high-level wastes only, and is devised so as to resolve the problems associated with heat dissipation and radiological exposure within the repository during the operational (waste emplacement) phase.

The concept involves the emplacement of high-level waste units in boreholes drilled in the floors of emplacement tunnels. As with the 'in-room' concept, it is generally envisaged that the tunnels would be arranged in a parallel fashion, linked by access tunnels and shafts.
Figure 41 provides a generalised illustration of the concept, including several possible variations in the configuration of tunnels and emplacement holes. Numerous alternative geometrical arrangements can be produced by varying the following basic parameters:

- borehole spacings
- borehole depths
- borehole inclinations
- number and vertical separation of waste canisters in each hole
- distance between adjacent emplacement tunnels

In addition, the above parameters may be varied by changing the size and shape of the waste units themselves, and the heat output of each unit at the time of disposal. For any given waste unit size and shape, a reduction in borehole and tunnel spacings may be obtained by cooling the units in an interim storage facility prior to disposal, or by lowering the initial concentration of fission products incorporated during the immobilisation process; see Chapter 4.2. It should be noted, however, that whilst interim cooling results in a reduction in the overall size of the final repository, a lowering of the waste concentration does not have the same effect, since an increased number of waste units is required to incorporate the same total quantity of waste.

The concept of combining horizontal tunnels with vertical or sub-vertical boreholes allows considerable versatility in determining the overall shape of the
underground repository. Increasing the depth of the boreholes allows the emplacement of a relatively large number of units per hole, with a corresponding reduction in the length of tunnelling required. Conversely, the use of shallow boreholes, with fewer units per hole, requires an increase in the lateral extent of the tunnel network.

Two extreme limits may therefore be envisaged. The first of these involves the construction of shallow holes, each containing a single waste canister, with all canisters emplaced at the same level, as shown in figures 41a and b (a planar configuration). In this case, the plan area occupied by the repository is theoretically identical to that required for the 'in-room' concept, described in 12.5.1 above. The other extreme is represented by a single, very deep hole extending from the surface, as described in 12.2. An optimisation of the geometrical arrangement between these extremes produces a cuboidal configuration which allows the exploitation of a given rock mass, in three-dimensions, having due regard to its vertical and lateral extent and in situ properties; see figure 41c.

Apart from this geometric flexibility, the 'in-floor' method presents a number of special operational advantages for high-level waste disposal. As with the 'in-room' concept, emplacement of the waste units involves a series of transportation and handling operations below the surface which require the use of shielded transfer flasks or remote-handling equipment. However, after deposition of the waste units within the boreholes, shielding is provided by the surrounding rock, and artificial shielding against radiation is no longer required, other than the provision of a plug at the mouth of each hole. Manned access into the
emplacement tunnels for other tasks is therefore possible throughout the operational life of the repository.

Following the deposition of an individual waste unit within a hole, immediate backfilling of the annular space around the unit may be carried out in order to provide thermal contact with the host rock. Thus, the repository ventilation system is not required to cool the waste after emplacement, and the period of exposure prior to emplacement is relatively short. This provides a greater security against possible contamination of air by radiation and a reduction in the level of dependence upon mechanical plant for safe operation of the facility.

In determining the optimum repository configuration, full consideration must be given to the relative cost and difficulty of tunnelling and borehole drilling, and the handling operations and maintenance requirements associated with any particular scheme. There is an incentive, in certain geological formations, to adopt a design incorporating long emplacement holes in order to accommodate the total quantity of waste within a repository of relatively small plan area. In addition, it may be considered advantageous to drill large diameter boreholes with an extra wide annulus to surround the canisters, so as to allow placement of a substantial 'engineered barrier', comprising a suitably formulated backfill material.

However, the blind-drilling of large diameter boreholes, within the restricted confines of a tunnel, imposes practical restrictions upon the diameters and depths of the holes. In addition, the difficulties of achieving safe and accurate emplacement of canisters and high
density backfill to ensure good waste-rock thermal contact increase with borehole depth and inclination from the vertical.

Whichever system is adopted, sealing of the emplacement tunnels (assuming these are not used for the 'in-room' disposal of intermediate-level wastes) is seen to be a relatively easy operation, since floor space in the emplacement tunnels is not restricted and shielding is not required after emplacement. A greater confidence in the efficiency of the tunnel backfilling is therefore possible by using direct methods of fill emplacement.

The foregoing discussion has centered mainly upon construction and operation aspects of a particular choice of 'in-floor' repository configuration. However, the shallow borehole (planar) configuration offers certain distinct advantages in addition to the relative ease of borehole drilling and waste emplacement. These are related to the heat transfer characteristics of the repository as a whole.

For planar arrays, the total heat-generating capacity of the emplaced waste is spread over a greater plan area, resulting in a greater dispersal of heat within the overlying rock. In consequence, the potential for setting up convective groundwater movements towards the surface is reduced, in comparison with the more compact cuboidal emplacement configurations.

Clearly, in view of the many options involved, the designer has a considerable scope for developing alternative repository schemes, based upon the concept of a tunnel network system with 'in-floor' waste emplacement. These may vary in terms of the cost and complexity of construction techniques used, the reliance
placed upon mechanical systems, the ability to conform to the geometrical restraints imposed by a particular site, etc. For all these reasons, the concept undoubtedly offers considerable overall design flexibility.

12.8 Summary

The foregoing account has outlined the essential generic engineering features associated with each of the design concepts identified in the international literature; with critical comment upon the viability of each method and its important design variations. To the author's knowledge no comparable reviews are available; although several conceptual design studies have been published, based on one or more of the disposal methods described; see Chapter 13.

Elsewhere, the author has provided a comparison between the various concepts, based on a ranking of various attributes with specific reference to high-level waste disposal (9). However, it is probably more appropriate to discount some of the concepts altogether, and a more intuitive assessment will suffice to provide a comparative summary.

Based on the premise that the disposal system should minimise the level of risk and uncertainty of release of radioactivity in the long term, those concepts which induce extensive disturbance or unquantifiable changes in rock properties should be relegated to a low level of priority for research and development. This applies generally to concepts which involve emplacement of liquid wastes and/or insitu rock melting. Concepts which offer little chance of gaining public acceptance,
or which imply excessive risk to operatives, or grossly disproportionate costs may also be excluded from serious consideration.

Based on the discussion of each concept in preceding sections, concepts which (in the author's view) suffer from one or more of these disadvantages include:

- the very deep drillhole concept
- the matrix of drillholes concept (applied to large waste volumes)
- the exploded cavity concept
- the mined cavern concept (applied to heat-generating wastes especially, but to all types generally)
- the in-room concept (applied to heat-generating wastes)
- the in-floor concept (applied to intermediate-level wastes)

Hence, by process of elimination, the following alternatives appear to be the most favourable:

- the matrix of drillholes concept, applied to the disposal of high-level wastes, provided predicted waste arisings do not require a large number of drill-holes
- the in-room concept applied to intermediate-level wastes, or high-level wastes which have been allowed to cool to a minimal level of heat-output
the in-floor concept, applied to high-level wastes having significant heat output at the time of disposal.

Generally, this conclusion appears to be endorsed by the nuclear establishment; since the majority of detailed proposals put forward in recent years have been based on one or other of the above concepts.

However, the disposal of intermediate-level wastes into mined caverns has been advocated by several authorities; see Chapter 13. A number of potential draw-backs to this system have been highlighted by this author in section 12.4.2 above; particularly those associated with operational radiation exposure, backfilling and peripheral ground disturbance. Assuming that operational exposure problems can be successfully overcome, the author considers that disposal of intermediate-level wastes into mined caverns is potentially suitable only for saliferous rock formations. This is based on the premise that peripheral disturbance can be prevented in saliferous rocks and that backfilling requirements in the immediate vicinity of the emplaced waste units may be minimised. Further consideration of these aspects is provided in Part 4.
13. INTERNATIONAL DESIGN PROPOSALS

13.1 Introduction

This chapter provides a comparative review of a number of recent international design concepts for the construction and commissioning of high-activity radioactive waste repositories. Emphasis is placed on concepts for the disposal of high-level reprocessing wastes into crystalline host rocks; based on detailed accounts available in the published literature. However, essential features of proposals involving argillaceous and saliferous host rocks are referred to for comparative purposes, and concepts for the disposal of intermediate-level wastes are also briefly considered.

Most nuclear power-producing countries now have active research and development programmes for the underground disposal of high-activity radioactive wastes, and many have developed conceptual repository design proposals for reference purposes. The critical assessment contained in this chapter reflects the essential elements of the principal repository engineering design proposals which have recently been put forward in all of the major western nuclear power-producing countries.

Elsewhere, the author has provided a detailed description and engineering evaluation of the conceptual repository design proposals for disposal of high-level wastes into crystalline rock formations put forward by each of the following countries (9):

- Canada
For the sake of brevity, it is not proposed to reproduce a detailed critical account of each scheme. Instead, the important features of these schemes will be described in the light of preceding chapters. The objectives are:

- to indicate the common features of the proposals and to show how these reflect an emerging consensus within the international nuclear community concerning the preferred layout of repository systems and methods of waste conditioning and emplacement

- to assess the extent to which the proposals take account of possible loss of containment associated with construction and waste emplacement processes

- to assess the reliability of the operational methods advocated for repository development, waste emplacement and backfilling; and the effect of the interaction between these activities during the repository development and commissioning phases

- to assess the influence of pre-disposal strategies, including waste conditioning and storage and waste unit design, upon the performance of the repository systems
to assess the extent to which the basic design assumptions made, and the proposed construction methods, procedures for backfilling and creation of engineered barriers, are adequate in meeting the objectives for long-term waste containment.

The review will concentrate primarily on details of the schemes proposed by the five countries listed above. As previously noted, these deal exclusively with crystalline rocks. To provide further insight, specific features of reference schemes put forward by authorities in other western countries and for other rock types will also be described. These are:

- Belgium; plastic clay
- Denmark; saliferous rock
- Netherlands; saliferous rock
- West Germany; saliferous rock

Generally, the conceptual design proposals put forward in a particular country have not been assembled in a single document, but consist of separate publications from which the author has assembled an overall picture. Often the published documents are found to contain rather bald statements, illustrated by diagrams and drawings; and in some instances it has been necessary to deduce or assume the reasoning behind certain aspects of the proposals. Sections 13.2 to 13.9 below provide a critical assessment of relevant aspects, and provide a background for the further development of repository design principles described in Part 4.
13.2 Disposal Systems and Site Selection

Table 10 indicates the disposal systems for high-level and intermediate-level wastes which have formed the basis for detailed conceptual repository design studies in each of the countries concerned. Also indicated are the nuclear authorities or government bodies responsible for commissioning the studies, and defining both the pre-disposal strategies and the basic assumptions and precepts for repository design. The organisations responsible for the development of repository engineering and waste emplacement proposals, and the corresponding reference documents, are separately identified.

Irrespective of rock type, almost all countries have adopted the tunnel network system with in-floor waste emplacement for the disposal of high-level wastes. Denmark is the sole exception, and has proposed a system based on the construction of a matrix of drill holes. Significantly, the volume of waste arisings predicted for the Danish proposals is less than for any of the other countries; see 13.3 below.

Although increasing attention has been focussed on intermediate-level waste disposal in recent years, comparatively few detailed repository design proposals have been published. Those which are available favour the mined cavern or in-room concept for non-heat generating intermediate-level wastes and the in-floor concept for heat-generating intermediate-level wastes, such as spent fuel claddings.

Thus, in general, the international concensus appears to confirm the general conclusions presented in Chapter 12. However, the proposed use of mined caverns in the
Netherlands, Sweden and West Germany, for the disposal of intermediate-level wastes, contrasts with the author's views expressed in Chapter 12. The Netherlands' proposals involve the use of large bunkers into which intermediate-level waste units are lowered by remote-control, from an overlying access tunnel, so as to produce a random stacking arrangement (92). The West German proposal involves the use of a dis-used salt mine at Asse (70, 188); and the Swedish proposal involves the construction of large caverns below the Baltic Sea (161, 192).

Although details of other schemes have not yet been published, reports indicate that in the United Kingdom, the Nuclear Industry Radioactive Waste Executive (NIREX) is also contemplating the disposal of non-heat generating intermediate-level wastes in a dis-used salt mine and it is reported that the French authority, ANDREA, is developing similar proposals involving the adaptation of a dis-used mine.

As shown in Table 10, certain countries have considered more than one host rock type. The available choice has clearly been dictated by geological circumstances. For some countries, the choice has been strictly limited or clear-cut in outcome. The whole of Sweden, for example, is underlain by Precambrian gneisses and granites of the Fenna-Scandinavian shield, and there is no alternative to the selection of a crystalline host rock (122, 123); see figure 42. In Canada, the nuclear energy industry has been centered within the province of Ontario, which is underlain by Precambrian crystalline rocks of the Canadian shield. Although initial research by the Geological Survey of Canada during 1973-74 included the investigation of rock salt and other deposits, emphasis has since been placed
almost exclusively upon the crystalline basement rock formations within Ontario province (161); see figure 43.

In the United States, initial studies were concentrated exclusively upon the bedded salt deposits of the Permian basin and Interior Salt Domes of Kansas, as shown in figure 44C. However, as a result of major policy changes during the latter part of the 1970's (9, 34), the scope of the studies was extended to include basalts and granite; see figures 44a and 44b.

The remaining countries identified in Table 10 are members of the European Economic Community. Since 1975, these countries have participated in jointly-funded research programmes partly directed towards the identification of suitable repository host formations in Europe (52). Britain and France, because of their relatively varied geology, have not made a firm commitment to the choice of any particular host rock; although their contribution towards the EEC programme for the identification of potentially suitable formations and corresponding high-level waste repository design proposals has been concentrated on crystalline rocks (11); see figures 45 and 46.

For the remaining European countries (Belgium, Denmark, Netherlands and West Germany), the available choices have been relatively limited and related studies have been concentrated on a single host rock. In Belgium, an inventory of suitable major rock formations indicated that only clays and shales were worthy of detailed consideration (11). Although seven potential sites were identified, it was found that the 'Boom' clay (a sub-horizontally bedded, stiff clay some 100m thick (sandwiched between water-bearing sand strata)), underlay
the CEN research establishment near Mol, in north-east Belgium. Because the material was considered potentially suitable, and to avoid difficulties in obtaining planning consents elsewhere, detailed site investigations and repository design studies have since been concentrated exclusively upon the Boom clay formation at the Mol site (52).

In Denmark, two deep-seated Permian salt domes (the Linde diapir and the Mors diapir), in Jutland, have been identified as suitable host rock formations (11). Although high-level waste repository feasibility design studies were initially based on a tunnel network system with in-floor waste emplacement, more recent studies have shown the matrix of drill holes concept to be economically attractive, in view of the low predicted waste quantities in Denmark (67).

In the Netherlands, three potentially suitable salt-dome formations have been identified in the north-eastern area of the country, which form a continuation of the Zechstein salt basin of north-west Germany (94). However, due to difficulties in obtaining planning consents for exploratory investigations on land, additional studies have been focussed on salt domes within the continental shelf, under the North Sea (161). In 1980-81, this work was supplemented by studies undertaken in collaboration with West Germany concerning the mechanical properties of rock salt at the Asse mine. However, conceptual repository design studies have hitherto remained non-site specific (92).

In West Germany, the Zechstein salt basin has been designated as the preferred disposal formation for high-level wastes; although high-level waste repository design studies have also been essentially non-site
specific (188). The Asse salt mine has been established as the preferred disposal site for intermediate-level wastes; although additional investigations within the disused Konrad iron ore mine have been carried out at a subordinate level (52).

In Britain and France, high-level waste repository design studies have been essentially non-site specific, and have been set against a background of considerable difficulties in obtaining planning consents for exploratory drilling work. In Britain, eight areas of crystalline rock were identified, as shown in figure 45; but, to date, drilling has been carried out only in Area 4 (the Strath Halladale Granite); whilst difficulties in obtaining planning consents have prevented exploration in the remaining areas.

In France, studies implemented by the Bureau de Recherches Geologiques et Minières concentrated upon the Massif Amorican and the Massif Central in western and southern-central France, respectively; which are the two principal areas in which massive intrusive igneous rock formations occur (52). It is reported that three sites have undergone more detailed study; and, although locations are not described in the literature, it is thought reasonable to deduce that a site close to the Cap de la Hague, on the edge of the Massif Amorican has been favoured. The use of a coastal repository site in this area of France would appear to have logistic advantages, in view of the proximity of the reprocessing facilities at la Hague, and the emphasis placed on obtaining foreign reprocessing contracts involving shipment by sea.

In both Britain and France, high-level waste repository design and development programmes have recently lessened
in pace, and no active site exploration work appears to be in progress. Repository concepts therefore remain non-site specific. However, intermediate-level waste disposal is now receiving priority, due to the pressing problem of waste storage; and in this context it is notable that, in Britain, NIREX has compiled an inventory of potential repository sites which is limited to those owned by the nuclear establishment or large corporate bodies. Such a strategy presumably represents an attempt to avoid some of the problems of public opposition associated with the more widespread approach to site selection previously adopted for high-level waste disposal studies.

In Canada, Sweden and the United States, conceptual repository design proposals have also been essentially non-site specific. This also reflects the universal difficulties experienced in obtaining planning consents for exploratory works. In Canada, public opposition and disagreements between the Federal administration and the Provincial Government of Ontario forced AECL to abandon its wide-spread field investigations in 1977. Whilst continuing to seek planning permission for sub-surface exploration at various sites, AECL then commenced exploratory investigations within the limits of their own nuclear premises at Chalk River, Ontario, and the Whiteshall Laboratories in south-east Manitoba. As a result of their investigations, an area adjacent to the Whiteshell Nuclear Research Establishment, underlain by the Lac du Bonnet batholith was leased from the Provincial Government in 1980 for a period of 21 years. The pluton is to be used for the development of an experimental underground facility for basic research into the underground disposal of high-activity radioactive wastes, and construction and development work is currently in progress (161).
In Sweden, geological investigations in support of the KBS high-level waste repository design proposals were mainly focussed on southern and eastern areas of the country. All the sites are owned by the nuclear utility companies participating in the KBS programme; thus avoiding difficulties associated with obtaining planning consents. The principal sites which have been investigated are at Karlshamn (site of an oil-fired power station), Finnsjo (adjacent to the Foresmark nuclear power station) and Krakemala (adjacent to the Oskershamn nuclear power station); see figure 42.

KBS have published numerous technical and scientific reports since the project was commenced at the end of 1976. But in December 1977, the first of two major design reports was produced, providing detailed proposals for the management and geological disposal of vitrified high-level reprocessing waste, based upon the assumption of an 'idealised' granite host rock. This report, known as 'KBS-1', (122) was submitted in support of fuelling applications for two new reactors, and a reprocessing contract was negotiated between the Swedish Nuclear Fuel Supply Company (SKBT) and the French corporation COGEMA.

Following domestic and foreign reviews of the 'KBS-1' report, the Swedish Government reached a decision in October 1978. It was accepted that the proposals were generally satisfactory, but it was held that KBS had failed to prove that host formations of the required size and engineering properties were available in Sweden.

In response to the Government's decision, KBS undertook supplementary site investigations, including eight
further borings to depths from 500 to 770m. New applications for permission to fuel the reactors were then submitted to the Swedish Parliament in February 1979, based upon a report of the supplementary site investigations. In examining the new application, the Government inspectorate called for independent geotechnical advice from eight consultants. Seven out of eight of these consultants reported unfavourably upon the supplementary report, but the inspectorate still recommended Government approval for the applications. Government approval was then given in June 1979, giving rise to considerable political controversy.

Repository proposals based on the disposal of spent fuel were also developed by KBS; and a report 'KBS-2' (123) was published in 1978. An independent review by a special sub-committee of the United States National Academy of Sciences (145), was generally favourable but avoided the issue of absolute safety; see Chapter 1. The review also concluded (as in the case of KBS-1) that the availability of a relatively fracture-free, large crystalline rock mass at a depth of 500m or more had not been proven, and that further site investigations were required.

The lack of unreserved support for the 'KBS-1' and 'KBS-2' schemes caused a dilemma for the Swedish Government. In May 1980, a national referendum was held which confirmed the public's opposition to the granting of new reactor fuelling licences, and a change of Government subsequently took place. The new parliament then confirmed that no further nuclear reactors would be built, other than those already in operation or at advanced stages of planning. This effectively limits the number of nuclear reactors in Sweden to twelve, and means that all of these should be shut down by the year
2010; i.e. at the end of their planned operating lives (161).

In the United States, a mixture of non-site specific and site specific repository design proposals have been evolved over a substantial period of time. During 1963-67, the Oak Ridge National Laboratories (ORNL) undertook a programme of in situ experiments in a disused salt mine near Lyons, Kansas, using irradiated fuel elements as a simulant for calcined solidified wastes. These field studies were followed by a conceptual repository design study for the Lyons site, known as 'Project Salt Vault'.

As a result of both technical and political opposition, the Kansas State Authorities refused permission for further development work, and 'Project Salt Vault' was abandoned in 1972. The principal reason for the cancellation of the project was the reported presence of unsealed well-borings in the area. In direct response to this event, a programme of borehole sealing research was initiated by the Atomic Energy Commission (AEC), in 1972. Further impetus was provided by the knowledge that the presence of unsealed borings could also be anticipated in most of the remaining potentially suitable saliferous deposits in the United States.

Thus, largely for political and circumstancial reasons, the 'Borehole Plugging Programme' formed the focus for a major research effort on methods for backfilling and sealing deep boreholes from 1972-78. Related studies were carried out by numerous contractors, managed by Oak Ridge National Laboratories (ORNL), under auspices of the Union Carbide Corporation Nuclear Division (UCC-ND).

ORNL then selected an alternative site for study in the
Permian salt basin, at a military establishment some 50 km east of Carlsbad, New Mexico (see figure 44C), and completed exploratory and engineering studies with a view to constructing a repository for low-level and intermediate-level solid radioactive wastes (73, 196).

In 1975, the study was transferred from ORNL to Sandia Laboratories, and was re-named the 'Waste Isolation Pilot Plant' (WIPP) project. Large-scale in situ tests were carried out and a conceptual design was evolved for the disposal of long-lived transuranic (TRU) waste arising from the U.S. military defence programme. Whilst the results of studies associated with the WIPP project have been disseminated to the civilian nuclear industry and the general public, the site itself has been ultimately designated for military use only. Meanwhile, the decision regarding its possible use as a pilot facility for disposal of wastes arising from the civilian nuclear power industry has remained in abeyance.

During the latter part of the 1970's, a series of major reorganisational changes took place, under President Carter's administration. Federal responsibility was firstly transferred from the AEC to the Energy Research and Development Administration (ERDA), and subsequently to the Department of Energy (DOE). The activities at ORNL were also expanded and re-organised under the auspices of the Office of Waste Isolation (OWI) and were subsequently transferred to the Battelle Memorial Institute under the Office of Nuclear Waste Isolation (ONWI).

In addition to these administrative changes, a new programme was developed to include the study of basalt, granite and shale in addition to salt deposits. In
relation to the crystalline rocks, various in situ studies were put in hand, including:

- evaluation of the Columbia River Basalts at the Hanford reservation nuclear site, by Rockwell Engineering Company (56, 93)

- evaluation of the granites at the Nevada nuclear test site by Lawrence Livermore Laboratory (13)

- participation in Swedish-American co-operative studies at the Stripa mine in Sweden by Lawrence Berkeley Laboratories.

It is noteworthy that, as in the Canadian research programme, the two American study sites were selected because they were Government-owned, rather than because they presented ground conditions which were inherently more favourable than elsewhere.

In April 1977, a policy statement was issued by President Carter, which placed a commitment upon ERDA to locate a number of potential repository sites, including two in salt deposits and others in alternative formations, including granite and basalt. As part of the same policy statement, it was announced that the development of facilities for the reprocessing of spent nuclear fuel would be postponed as part of a policy of non-proliferation of plutonium, pending the outcome of the International Nuclear Fuel Cycle Evaluation Study (INFCCE); see Chapter 3.3. As a result of these decisions, the study of igneous formations in the United States became as intensive as those previously devoted to salt deposits. Figures 44a and 44b show the principal areas of igneous rock which have been under general consideration.
The scope of conceptual design studies was also expanded to consider the disposal of un-reprocessed spent fuel as an alternative to vitrified waste. However, since it was generally recognised that the reprocessing of spent fuel could be re-adopted by a subsequent administration, studies by ONWI continued to make allowance for the possibility of handling and disposal of solidified high-level reprocessing wastes (54, 156, 166).

A series of Generic Environmental Impact Statements (GEISS) concerning the various waste disposal options was commissioned by ERDA in 1977, and the task of reporting upon geological disposal was allocated to ONWI. The technical material published in support of these impact statements includes important accounts of American proposals in relation to repository design in granite (159), which are used as a basis for the comparative review described in later sections of this chapter.

In February 1980, following the completion of an Inter-agency Review Study, a comprehensive new waste management programme was announced, which involved the cancellation of WIPP and the evaluation of four or five potential repository sites, to include salt and igneous crystalline formations. Under this programme, in situ studies are to continue at both the Nevada and Hanford sites, together with possible further experimentation at the Stripa mine. American workers have also taken great interest in the results of the Swedish KBS project, particularly in relation to the design of engineered barriers such as overpacks, backfill materials, etc.

Since the commencement of the Reagan administration in 1981, further policy changes have been introduced. In
particular, the ban on reprocessing has been lifted and America is anxious to re-establish its position as a world-leader in the international reprocessing industry. In terms of waste disposal it now appears that the disposal of spent fuel is unlikely to be considered further.

During the past ten years, therefore, the American studies have continued to be broadly-based, whilst policies have remained in a state of flux, and no clear strategy has evolved. Numerous reviews and scientific and conceptual design studies have been completed and, in terms of breadth of research, there is no doubt that the United States has a clear world lead. Fortunately, the results of American studies have been widely disseminated and, despite vississitudes in American research policy, other countries have benefited considerably from various American technical research initiatives.

It is apparent from the above, that no country has found it possible to select a site on a purely technical (geological) basis. It appears likely that many countries will be forced to adopt the expedient of selecting a site underlying a nuclear or Government-owned establishment. Clearly, therefore, 'idealised' generic assumptions concerning host rock properties contained in published non-site specific repository design proposals must be viewed with considerable caution. Any critical evaluation should take into account both the inherent variability of natural rock mass properties and the fact that 'ideal' host formations used as a basis for some conceptual design studies may not be available in practice.
13.3 Waste Form and Characteristics

Tables 11 and 12 provide a comparison of the quantities and characteristics of the high-level waste units assumed in the various international conceptual design studies for crystalline rock repositories. Clearly, a wide variation exists in the preferred combination of waste characteristics. In addition, differences in nuclear generating capacities and in the assumed design production periods make direct comparisons difficult. However, it is worthwhile to attempt to rationalise these differences in order to determine the parameters which appear to have had the strongest influence upon the reference repository design concepts.

In the author's view, the waste characteristics which have the greatest influence upon the design and performance of an underground repository are the following:
<table>
<thead>
<tr>
<th>Waste Characteristics</th>
<th>Main Influence on Repository Design</th>
</tr>
</thead>
<tbody>
<tr>
<td>total heat emitted by the waste units to be buried within the repository at the time of disposal (number of waste units multiplied by individual heat output)</td>
<td>overall repository size/volume consistent with maximum rock temperature constraints</td>
</tr>
<tr>
<td>heat output of individual waste units at the time of disposal, in combination with overall canister size, shape and thermal conductivity</td>
<td>minimum allowable spacing of emplacement tunnels and boreholes consistent with near-field rock temperature constraints</td>
</tr>
<tr>
<td>provision of overpacks and concentration of radionuclides within the immobilising medium</td>
<td>provision of shielding facilities during transportation and emplacement; degree of ultimate confinement</td>
</tr>
<tr>
<td>size, shape and weight of waste units</td>
<td>size of excavations and type and capacity of plant required for handling and emplacement</td>
</tr>
<tr>
<td>number of waste units and rate of disposal</td>
<td>size and layout of distribution shafts and tunnels</td>
</tr>
</tbody>
</table>

Table 13 summarises the values of the more important waste parameters which have been adopted for design purposes, together with parameters derived by the author to indicate their influence. In making comparisons, it
is important to note that variations in assumed rock properties produce superimposed differences. Therefore, it is assumed for present purposes that all countries have adopted identical 'idealised' rock properties, so that the direct influence of waste form may be examined. The effect of variations in assumed rock properties is described separately in 13.5 below.

Table 13 is divided into columns containing figures derived from the data shown in Tables 11 and 12. The parameters shown in columns e, f, g and h are of prime interest and, for assumed constant rock properties, may be related to characteristic engineering features of the reference design concepts.

The total heat output at disposal (column g) influences the relative overall sizes of the repositories (total volume of rock required for disposal), and is a function of the total quantity of heat-generating waste and the period of interim storage. Clearly, on the basis of the waste arisings and characteristics adopted, the United States and Canada both require relatively large repositories to accommodate the production quantities foreseen. France too requires a fairly large repository, whilst Sweden and Britain need relatively small total rock volumes for disposal.

The values entered in columns f and g are not in constant proportion for all countries, due to differences in the periods of cooling prior to disposal (except in the case of Canada and the United States, where identical storage periods have been adopted). Thus, the United Kingdom requires a marginally smaller total rock volume than Sweden, despite the fact that the total mass of fission products to be accommodated is considerably larger. This is due to the relatively long
period of interim cooling assumed for the British design concept.

The heat output per unit mass of fission products at disposal (column h) is indicative of the relative overall degree of compactness of the repositories (due to the combination of waste characteristics), and removes the influence of variations in waste production figures.

It is interesting to note that Britain has adopted a combination of waste characteristics which allows the greatest degree of overall compactness. At the other end of the scale, Canadian and American designs are remarkably similar, and compactness does not appear to have been given priority consideration.

The heat output per unit surface area of each waste unit (column e) reflects the required density of waste distribution within the repositories. For a given value of maximum allowable rock temperature, a relatively low rate of surface heat flux allows a close waste unit spacing, e.g. Canada and Sweden; whilst a high value of heat flux requires larger spacings, e.g. France and the United States. Reduction in waste unit spacings may be achieved by increasing the period of interim storage, reducing waste concentration, or increasing the curved surface area over which heat is transmitted to the surrounding rock. The emphasis placed on each of these factors is seen to vary considerably, as summarised below:

<table>
<thead>
<tr>
<th>Design Emphasis</th>
<th>Country</th>
</tr>
</thead>
<tbody>
<tr>
<td>long interim storage</td>
<td>United Kingdom, Sweden,</td>
</tr>
<tr>
<td></td>
<td>France</td>
</tr>
</tbody>
</table>
low waste concentration  Canada, Sweden

large canister surface  United Kingdom, Canada

area

In addition to its influence on repository size and configuration, the waste form also affects requirements for mechanical handling, and, more fundamentally, the long-term containment of the wastes after burial. Reduction in waste concentration results in reduced concentration-levels for mobile radionuclides arising from the leaching action of groundwater, and an overpack provides a further engineered barrier. However, an increase in waste unit size and weight also increases the difficulty of waste handling.

It is apparent that benefits gained by varying one waste characteristic may be offset by advantages in another area. Table 14 provides a general summary of the various factors outlined above, for each of the countries examined. A simple numerical ranking is used to indicate the emphasis placed by each country upon the different aspects of waste form and packaging, in relation to repository design.

The distribution of emphasis conforms with the general remarks above. Particularly surprising, however, is the relatively large numerical rankings in all areas which are assigned to the United States proposals. This is thought to endorse the author's view, that despite considerable research effort to date, comparatively little progress has been made in tackling the problem of optimising the design parameters involved. It is thought that the same criticism may be made of the remaining international proposals to a greater or lesser
Table 15 indicates the essential characteristics of high-level waste units for concepts involving disposal into argillaceous and saliferous rocks. The relatively slim waste units proposed by the Netherlands and West Germany are thought to reflect difficulties foreseen in emplacing waste units in deep boreholes in salt formations. Due to the creep processes, described in Chapter 10.2, which are accelerated under the influence of heat, it is evident that boreholes must be suitably over-sized in relation to the waste units; especially where long boreholes are considered (due to the comparatively long time required to fill the borehole). Because of practical limitations on the maximum diameter of boreholes, it would appear that it has generally been found more expedient to adopt a relatively small waste unit diameter.

However, for a given heat output per unit, reduction in diameter increases the surface heat flux considerably. The initial surface heat flux of the West German unit is particularly high; see Table 13. However, it may be noted that saliferous rocks have relatively high thermal conductivities in comparison with other rock types; see Table 9.

Little or no information is available concerning the detailed design of intermediate-level waste units for any of the countries considered. However, because of the insignificant heat output of the majority of these wastes, the principal factor affecting the design of repository facilities is the shape and size of the unit; since these characteristics influence waste handling and stacking arrangements. It appears that cubes and cylinders are the preferred shape, presumably due to the
relative ease of fabrication; and the majority of units are likely to be 1m or less in maximum dimension. However, where shielding is provided and where decommissioning wastes are included, larger waste units may be required. Further consideration of the influence of waste package design for intermediate-level waste disposal is included in Part 4; see Chapters 18 and 19.

13.4 Interim Waste Storage

Tables 12 and 15 show that the periods of interim storage contemplated for high-level wastes are considerable. Depending on size, shape, composition and required maximum heat output at disposal, the periods proposed vary from 10 to 60 years. Storage periods of such magnitude can only be contemplated for immobilised and packaged wastes, and the design of suitable facilities clearly forms an essential part of the overall waste management and disposal concept. Interim storage facilities for heat-generating intermediate-level wastes are of lesser concern, because the periods required for loss of significant heat output are relatively small; see Chapter 4.3.

Of the five principal countries examined, only Sweden and Britain have produced conceptual design proposals for interim high-level waste storage at the repository site. It should be noted that Canada, France and the United States have developed independent design proposals for separate interim high-level waste storage facilities, to be constructed at the surface, close to the site of the reprocessing plant. In France, these facilities are already in full operational use at the la Hague reprocessing site, as described in Chapter 4.2.
In these countries, it is envisaged that temporary storage facilities will be contained in surface structures. However, Canada has evolved a unique proposal, involving storage of individual waste units within cylindrical concrete shield blocks, exposed to the atmosphere so as to allow heat dissipation by natural convection. The United Kingdom Atomic Energy Research Establishment has developed an outline proposal for a similar scheme, which has been reviewed by the author in terms of economic and engineering viability (10). However, this method of storage does not form part of the recently published conceptual design proposals.

Both the Swedish and British proposals envisage interim storage facilities constructed at shallow depth (30 to 50m) below ground level; primarily for security and environmental reasons. The Swedish concept involves construction of caverns in solid rock (122), whereas the British proposal involves construction of a series of storage tunnels (144). In this context, it must be noted that Sweden has considerable experience in the design and construction of rock-cavern storage systems and, in comparison with Britain, suitable competent rock formations are readily available at the depths envisaged.

Despite these differences, no real problems are foreseen in the design and construction of sub-surface interim storage facilities and the provision of associated waste handling and ventilation plant. Construction of such facilities is well within the scope of current civil engineering practice, and the associated waste-handling technology is also well developed.

It is significant that none of the countries has
favoured use of circulating water as a cooling medium. Forced circulation of air appears to provide the optimum solution, and has been well demonstrated in France. There appears to be a general consensus that air-cooled interim storage facilities could be provided, with an operating life in excess of one hundred years, if required.

13.5 Repository Depth and Host Rock Properties

No common rational basis appears to have been developed for determining the optimum repository depth. The selection of burial depth is strongly site-dependent and involves consideration of the following factors:

- overall geological environment
- geothermal gradient and induced rock temperature increase
- local variations in the nature of discontinuities and mass permeability with increasing depth
- size and shape of the host formation
- thickness of overburden and depth of weathering
- surface topography
- ground water regime
- virgin in situ stress field and rock competence
- proposed geometrical configuration of the repository
In view of these site-specific influences, none of the international conceptual models can be considered to be definitive. However, a number of general points may be made:

- construction of shafts and tunnels to depths of up to 1000m is well within conventional civil engineering and mining experience

- variations in construction depth in the range 300-1000m are likely to produce relatively small differences in overall costs, since the costs of shaft construction will generally represent a small proportion of the whole

- geothermal gradients in the older (e.g. Precambrian) formations will be relatively small

- disturbance to the rock cover due to long-term geological and climatic changes could occur to depths of about 100 to 300m (81, 122).

- in general, an increase in burial depth results in an enhanced level of confinement, due to increase in the length of potential migration paths to the biosphere.

Thus, there is a general incentive to adopt a substantial depth of burial, consistent with restraints imposed by ambient rock temperatures and problems associated with construction; see Chapter 10.4.

For high-level waste repositories in deep crystalline rock formations, thermal parameters have the greatest overall influence, since these affect the choice of
waste unit spacings, and hence the overall size and configuration of the sub-surface facilities. Mechanical and hydraulic rock properties are also highly significant in relation to ground stability and degree of confinement, for a given geometrical configuration, stress field and groundwater regime.

The principal thermal design constraint is the maximum allowable rock temperature, which in turn influences the maximum glass temperature developed within the waste units. As shown in Chapter 9.3, borosilicate glass becomes unstable at temperatures approaching 400°C, and therefore an adequate factor of safety is required to reduce the risk of devitrification. However, since physical and chemical changes could occur in certain rocks, at temperatures as low as 100°C, the threshold rock temperature is generally regarded as the more critical parameter.

Table 16 indicates the repository construction depths and corresponding thermal design constraints adopted for the crystalline rock repository concepts for high-level waste disposal, put forward by the five principal countries under review. Where a range of depths is indicated, the depth interval corresponds to the maximum length of emplacement holes envisaged.

As shown, the maximum assumed permissible rock temperatures generally lie in the range 80-150°C. However, proposals put forward by the United States envisage rock temperatures of up to 260°C, with waste unit surface temperatures approaching 375°C (159). Under these conditions, thermal cracking of the rock would be expected to occur in the borehole walls, extending for a distance of approximately 150mm. It is notable that steel sleeves are proposed for lining the
emplacement holes and these may be intended to prevent detached rock fragments from jamming the waste units and preventing short-term retrieval.

The acceptance of these high temperatures allows a close spacing of waste units in relation to their heat output at disposal; but the American design appears to provide inadequate thermal stability in the near-field. This is recognised in the design report, where it is stated in conclusion that a revision of waste unit spacings would be desirable (159).

Maximum induced rock temperatures depend upon the pre-existing ambient temperatures, determined both by the depth of burial and the local geothermal gradient. In this respect, the French and British design proposals are constrained to a greater degree by the relatively high anticipated geothermal gradients, in comparison with other countries where older (Precambrian) host rocks are available.

Assumed values of thermal conductivity for the crystalline rock types used as a basis for the international studies lie within the range 2.5–3.0 W/m°C (falling towards the lower-bound limit of the available data; see Table 9). For initial design purposes, a value of 2.5 W/m°C is considered to be reasonably conservative for an 'idealised' homogeneous, isotropic granitic material, whilst a value of 3.0 may be more appropriate to a gneissic host rock (e.g. Sweden). However, it must be noted that values of induced rock temperature increase are particularly sensitive to variations in thermal conductivity (75). Furthermore, real values of thermal conductivity will vary according to the anisotropy of the host rock. This effect is likely to be most apparent in banded gneissic rocks,
where values of thermal conductivity parallel to the mineralogical banding may be twice the values for the perpendicular direction (47). The value assumed for conceptual design purposes is therefore largely conjectural, unless related to a specific site. To this extent, values of thermal conductivity assumed by Canada and Sweden allow relatively close spacings for given maximum allowable rock temperatures.

Table 17 indicates the range of construction depths for all the countries under consideration. As shown, proposals involving disposal into saliferous rocks generally involve greater depths; notably the matrix of drillholes concept advocated by Denmark. This reflects both the higher temperature stability and thermal conductivity of saliferous rocks, and the need to exploit the vertical dimensions of the formation; particularly in salt domes of limited lateral content.

In contrast, the Belgium proposals envisage a depth of only 220-235m. In this case, the depth range is limited by the available thickness of Boom Clay; see 13.2 above. Clearly, however, ambient geothermal temperatures will be comparatively small, and this may be regarded as advantageous in view of the relatively poor thermal conduction properties of clays; see Table 9. Although the Belgian proposals currently represent the only available concept for a clay host formation, it may be envisaged that the available depths would be relatively small for clay hosts in general.

Table 17 also serves to illustrate the range of ground stability conditions implied by the published repository design proposals. The assumed rock strengths are considered to be reasonably representative of those available for unweathered rocks at depth; see figure
24. However, it is interesting to note the corresponding stability conditions implied for the depth ranges envisaged. Several of the proposals for crystalline rock repositories imply competence factors of less than 10, and as shown in Chapter 10, this could imply a need for partial support and risks the possibility of generating localised ground disturbance in the vicinity of the excavations; with possible adverse consequences in terms of containment properties.

For an assumed unconfined compressive strength of 150 MN/m$^2$ and a mean rock density of 30 KN/m$^2$, the maximum allowable construction depth for a rock competence factor of 10 is 500m. Even this depth limit may be excessive where relatively high $K_o$ values occur, and thus unless the rock is stronger than the assumed values, construction depths approaching 1000m in granitic rocks may increase the risks of inducing flow path anomalies; see Chapter 10.

It is also interesting to note the high over-load factor implied by the Belgian proposals. The predicted over-load factor is greater than five; which for tunnelling at moderate depths would suggest the need for continuous support, using a shield and face-support (e.g. by means of a slurry plenium chamber or earth-pressure balance chamber). However, the in situ stress of approximately 4.6 to 5.0 MN/m$^2$ implies ground loadings an order of magnitude greater than those hitherto experienced in materials of the same type. Severe problems associated with ground ’squeeze’ may therefore be anticipated, and it is clear that the feasibility of repository construction will be strongly dependent upon further research and development.

For the proposals involving saliferous rocks, the
predicted rock competence factors are low. However, as shown in Chapter 10, the use of competence factors is less appropriate in saliferous rocks, in view of their distinctive deformation properties. However, it is evident that deep emplacement holes may have limited effective stand-up time and the concurrent phasing of construction and waste emplacement activities appears inevitable.

A comparison of assumed mass permeabilities also provide a valuable insight into the anticipated performance of the 'idealised' host rocks assumed for conceptual design purposes. Values assumed for crystalline host rocks by the five principal countries under review are:

- Canada $10^{-8} - 10^{-11}$ m/sec
- France $10^{-8}$ m/sec
- Sweden $10^{-13} - 10^{-9}$ m/sec
- United Kingdom $10^{-12}$ m/sec
- United States $10^{-12}$ m/sec

All values are predictably low (see Table 13 and figure 17). However, it is significant that the range of generic values used as a basis for conceptual repository designs varies over five orders of magnitude, reflecting the basic uncertainties inherent in the assessment of rock mass permeabilities.

13.6 Repository Configuration

Although the various published repository design
concepts vary in engineering detail, proposed configurations all appear remarkably similar (with the exception of the Danish matrix of drillholes concept). The concepts based on tunnel network systems are all rectangular in layout, comprising parallel waste emplacement tunnels linked by access and ventilation tunnels, generally all at a single level, as illustrated in figure 47.

Figures 48 to 52 illustrate the crystalline rock repository layouts for the high-level waste disposal schemes put forward by Canada, France, Sweden the United Kingdom and the United States, based on an adaptation of drawings presented in the published literature. Figure 53 illustrates the Belgian repository scheme, largely for comparative purposes.

For high-level waste disposal, based on the in-floor concept, the depth of emplacement holes influences the plan area required. Of the five crystalline rock repository concepts studied, Canada, Sweden and America have all adopted planar configurations, where waste units are emplaced singly in holes drilled to depths of up to 6m. In contrast, France and Britain have selected cuboidal configurations for their reference concepts, with holes up to 300m deep, in which waste units are stacked vertically.

Few engineering investigations have been carried out to evaluate the options available between these two extremes. Some optimisation studies have been carried out in France and Britain (27, 144), but other countries appear to remain committed to planar configurations only. It is thought that, in practice, the configurations adopted are partly a reflection of differing geological and geographical circumstances;
Canada, Sweden and the United States having relatively extensive crystalline basement complexes available for exploitation.

The planar configuration is seen to be more compatible with current engineering construction and operational technology; since emplacement holes are drilled to relatively shallow depths, and waste unit emplacement and backfilling are relatively straightforward. In addition, facilities for waste monitoring and retrieval are more easily provided. However, the cuboidal arrangement (in which waste units are stacked in a vertical waste column) allows considerably more flexibility in the three-dimensional exploitation of a given host formation.

With respect to thermal characteristics, the cuboidal configuration induces longer-lasting rock temperature rises in the near-field, and a greater tendency for convective groundwater movements in comparison with planar configurations; see Chapter 9.3. This implies a need for greater emphasis upon the provision of overpacks to provide short-term containment, in the case of cuboidal designs. Surprisingly, this is not evident in the international proposals, where only the Swedish proposals (planar configuration) envisage provision of an overpack.

As described in 13.5 above, cuboidal configurations have generally been proposed for saline rock repositories. Here, the problem of convective groundwater movement does not arise, and there is an incentive to maximise hole depths as a means of attaining greater distance from the regions of potential groundwater inundation at the margins of the formation, where water-bearing rocks are encountered.
As shown in figure 53, the Belgian proposals also envisage a cuboidal emplacement configuration for high-level wastes, and inclined holes are envisaged as a means of exploiting the limited available thickness of the clay. It must be noted however, that the stability problems referred to in 13.5 above apply equally to the construction of boreholes, and the feasibility of forming inclined holes of this type, without the extensive use of ground freezing, must be questioned.

Partly due to differences in the predicted total quantities of waste arisings and the choice of high-level waste emplacement configuration referred to above, the overall sizes of the repository systems vary considerably. As would be expected, systems based on planar configurations tend to be larger. However, it is evident that there has generally been little attempt to limit either the number or size of excavations, or the overall areal extent of the repositories.

As shown in figure 48, the proposed Canadian repository system extends over a rectangular area some 0.87Km wide and 3.65Km in length; with twin exhaust shafts located at one extreme end of the site. The system has been based on the 'room and pillar' mining concept, with five separate longitudinal tunnels for haulage, and a downcast ventilation shaft in addition to the twin exhaust shafts.

The American system illustrated in figure 52 is remarkably similar to the Canadian scheme, although it occupies a somewhat larger area and both intake and exhaust ventilation shafts are located at the same end of the repository. It is also extensive; the area for high-level waste disposal measuring over 2Km square.
The Swedish repository configuration as shown in figure 50 is also similar, though relatively small (measuring 1Km square). Four parallel access tunnels are provided, one of which is designated only for ventilation, and there are two ventilation shafts in addition to the two access shafts.

The French repository scheme, illustrated in figure 49, also occupies an area measuring over 2Km square, but unlike the Canadian, Swedish and American proposals, the access tunnel system runs continuously around the periphery, with no central division of the facility. The British concept illustrated in figure 51 is broadly similar to the French scheme. However, largely due to the high fission product concentration of the waste units as shown in Table 12, the repository is relatively compact, measuring about 0.5Km square.

It is surprising to note that the rectangular layout adopted for the crystalline rock repository proposals has also been utilised as a basis for the Belgian scheme shown in figure 53. In the author's view, a rectangular layout of this type, involving multiple tunnel junctions, is not favourable for tunnelling in clay; since enlarged junction chambers would be required at the ends of each drive, creating severe ground stability problems.

The saliferous rock repositories, which are not illustrated, follow a broadly similar configuration to those described above, involving a parallel arrangement of tunnels with a series of inter-linking tunnels and shafts for access and ventilation.

Since differences in waste production figures tend to
obscure repository size comparisons, it is perhaps more constructive to examine the volumetric utilisation of space for the concepts under review. Table 18 indicates the volume of excavation for each scheme, in relation to the volume of emplaced waste. The particularly low volume utilisation for all the in-floor high-level waste disposal concepts is self-evident, and is due to the non-utilisation of tunnels overlying the emplacement holes. Clearly there would be considerable incentive to utilise these overlying tunnels for intermediate-level waste disposal; although none of the international waste disposal programmes appear to allow for this. Notwithstanding the latter observation, the extreme variability in effective volume utilisation generally confirms that no efforts have been made to optimise the volumetric efficiency of the repository designs.

13.7 Construction and Operation

With the sole exception of the British proposals (144), all the concepts under review for crystalline and saliferous rock repositories envisage concurrent repository construction and waste emplacement. The proposals also generally envisage the use of conventional drill and blast techniques as the principal method of excavation at the emplacement horizon and, as noted in Chapter 11, the majority appear to be based on mine development principles.

13.7.1 Canadian Proposals

In the Canadian high-level waste repository concept, illustrated in figure 48, the principal excavations are termed longitudinal drifts, panel drifts and emplacement
rooms. The central drift divides the plan area of the repository into two sections, each 450m wide. Separate rock haulage and backfill haulage drifts run along the length of the repository and are located 30m below and above the central drift, with a series of vertical raise-bored connections for transfer of materials to and from the repository level. Thus, the three central drifts provide access for all construction, spoil handling and backfill distribution operations, whilst the two outer drifts are used only for transfer of waste.

Tunnel heights are determined by the headroom requirements for the drilling of emplacement holes, using crawler-mounted drills. As shown in figure 54, a minimum clear headroom of 4m is specified for this purpose, with a clear height of 5m at roof mid-span. Emplacement rooms are 188m long and 7.5m wide. Holes are drilled at 1.5m centres in rows of four, providing 500 emplacement holes in each room. Each hole is 0.61m in diameter and 4.7m deep, providing space for a single canister per hole.

The spacing between holes is reported to be the minimum practicable separation, based on consideration of drilling operations, and is apparently not the minimum value dictated by thermal loading criteria. No attempt is made here to make a critical assessment of the thermal analysis used to derive these spacings. However, it is apparent that the Canadian proposals produce a very close spacing of waste units compared with other countries, even allowing for the relatively low fission product concentration; see Table 12. In view of this close-packed arrangement, it would be interesting to examine whether potential cost savings would arise by construction of a continuous trench as an alternative to
the grid of drillholes, with placement of canisters into holes precast into a mass concrete backfill. It is noted also that the proposed spacing between drilled holes is less than 1m, and it is not apparent how emplacement and backfilling vehicles would traverse the perforated floor space, since considerable manoeuvring of the trackless emplacement vehicles is evidently required.

Widths of drifts are not specified, but a nominal width of approximately 5m has been inferred from other data provided. Passing points are provided in all longitudinal drifts at intervals of 200m, involving localised widenings.

Pillars of 23.5m width are provided between adjacent emplacement rooms, providing adequate separation from a structural point of view. This is quoted as an equivalent 25% extraction ratio, although the dimensions given suggest a higher value. The pillar widths which separate the three longitudinal drifts from the nearest emplacement rooms are approximately 50m. This width is greater than can be accounted for by the need for passing points, and the author considers that the widths of the longitudinal pillars could be reduced, without affecting stability, in order to reduce the overall area occupied by the repository by approximately 10-15%.

No provision is made for lining of tunnels or screeding of floors, except in the emplacement rooms, where a level concrete pad is to be placed over the area required for emplacement holes. It is assumed that nominal rock bolting will be required in the crown of the tunnels. However, all shafts are to be concrete-lined.
Drill and blast excavation has been assumed throughout for both shafts and tunnels. The construction proceeds in three phases over a period of approximately sixty-five years, comprising:

- access and demonstration (4 years plus 6 years demonstration)
- primary development (5½ years)
- panel development (45 years)

The access and demonstration phase involves the construction of the two exhaust shafts connected by an 850m long drift, to be used initially for demonstration purposes. Both shafts are to be 3.65m diameter. The first shaft is to be sunk using normal drill and blast excavation from the surface, and the second using raise boring techniques. One of these shafts is to be used as a temporary hoisting shaft until completion of the primary development phase.

From the interconnecting drift between the shafts, a pilot tunnel is then to be driven along the alignment of the proposed central drift, to the location of the service shaft. The period allowed for this critical phase of construction is four years, and a further six-year period is allowed for demonstration activities. As noted in 13.6 above, it is intended that the longitudinal pilot tunnel will be used for exploration of ground conditions, although no details of techniques are described.

The primary development phase begins with the construction of the main 7.3m diameter service shaft. A pilot raise bore is first drilled from the pilot tunnel
to the surface and the shaft is subsequently enlarged to full-size by excavation from the surface, with spoil removal through the pilot hole at the bottom, where it is transported along the longitudinal pilot tunnel to the temporary skip hoisting shaft (see figure 48b).

On completion of the service shaft and installation of hoisting equipment and rock bins, the adjacent waste shaft and intake ventilation shaft are both raise-bored to the surface (3.6m and 4.6m diameter respectively). The five longitudinal drifts are then excavated full-face to the opposite end of the site, using the service shaft for hoisting of spoil. A total period of five and a half years is allocated for this primary development phase, with completion of the central drift by enlargement of the pilot tunnel after four and a half years.

Panel development then commences at the exhaust shaft end of the repository, and proceeds towards the service shaft end. Lateral drifts are first excavated from the central drift, and emplacement rooms are then constructed by simultaneous working of several faces with concurrent drilling of emplacement holes. During the panel development stage, tunnel rock spoil is removed by front-end loaders to rock passes in the central drift, where the material is transferred down vertical holes to the rock haulage drift below. In the rock haulage drift, electric locomotives are used to haul the material in muck skips to a loading pocket at the base of the service shaft for hoisting to the surface.

On completion of each panel, a security grill is placed at the intersection of the panel drift and the central drift, to prevent further access by construction plant
Emplacement of waste units begins as soon as the first half-panel is complete; see figure 54. Construction and emplacement activities then proceed in parallel, with completion of construction after a period of forty-five years from the start of the panel development phase.

The maximum rate of emplacement is likely to be 14 waste units per shift, based on a double shift system. Transporters are to be rubber-tyred, four-wheeled articulated diesel mine vehicles, each with a single, tilting shielded transfer flask at the rear. The waste units are loaded into the transporters at the waste shaft in the vertical position, and, after tilting to the horizontal, are driven to the emplacement room. Vehicles are then reversed into the emplacement room and after raising the flask to the vertical position and adjusting the alignment, a shielding sleeve is dropped and the waste unit is lowered into the hole on to a prepared concrete plug at the base. The transporter then returns to the shaft area to commence a new cycle.

Initial studies (1) assumed a one-hour emplacement cycle, involving a 30-minute travel time and a 30-minute emplacement time. If this cycle time is assumed for the reference scheme, then it appears that only two transporters are required, each completing up to 8 cycles in a single shift. Thus, it would appear that transporters need not pass one another, since they can operate in separate halves of the repository. This would imply that passing points are in fact unnecessary, and further reductions could be made in the size of the repository access tunnels and hence the overall plan area of the repository.
Figure 55 illustrates the ventilation flow system for the repository. The air flow is uni-directional, towards the unmanned sections of the repository. The air intake shaft is equipped with a single 1080kW fan, and a surface structure above each of the exhaust shafts houses an axial flow fan, damper and filter section. During normal operation the exhaust air by-passes the filter system but, in the event of an emergency involving contamination of air, by-pass dampers would close to exhaust air through a filter bank.

The ventilation system is designed to provide an adequate supply of fresh air in the main drifts, allowing for the operation of diesel-powered plant in the repository. However, for each operating panel, auxiliary fan installations and ducting are required to ventilate individual rooms during construction, waste emplacement and backfilling activities.

13.7.2 French Proposals

In the French high-level waste repository layout, illustrated in figure 49, a series of 82 parallel emplacement tunnels, each 2285m long, are constructed at 26m centre to centre spacings. Three parallel distribution tunnels, A, B and C run from the shaft area and merge into a single peripheral service tunnel, D. As in the case of the Canadian scheme, it is intended that construction and emplacement operations will proceed concurrently, but with physical separation within the repository. Service tunnel A is therefore used for construction access, whilst tunnels B and C are used for transfer of waste units and emplacement equipment and personnel. The overall plan area of the repository measures 2.3 km by 2.44 km.
A total of six shafts is envisaged, comprising two 5m diameter shafts and four 4m diameter shafts. Three shafts are required for construction, and three for operational use, as summarised below:-

<table>
<thead>
<tr>
<th>Diameter</th>
<th>Construction Shafts</th>
<th>Operational Shafts</th>
</tr>
</thead>
<tbody>
<tr>
<td>5.0 m</td>
<td>Spoil removal</td>
<td>Waste transfer</td>
</tr>
<tr>
<td>4.0 m</td>
<td>Personnel and services</td>
<td>Personnel</td>
</tr>
<tr>
<td>4.0 m</td>
<td>Materials and ventilation</td>
<td>Services and ventilation</td>
</tr>
</tbody>
</table>

The shafts are located at the end of the service tunnels A, B and C, and are within the area designated as the surface site.

Each emplacement tunnel contains 74 boreholes drilled into the tunnel floor at 30m centres. The holes are 1.0m diameter, drilled to a depth of 100m, and waste units are emplaced at vertical intervals of 20m, with 5 units per hole. Thus each emplacement tunnel is able to accommodate 370 waste units.

At the base of the waste transfer shaft, shielded transfer flasks are conveyed by self-propelled electrical rail-mounted locomotives to the emplacement holes. After the transfer flask is aligned above the hole, a temporary shielding plug is removed, and the unshielded waste unit is lowered into place, guided by centralising devices. After the shielding plug has been replaced, the empty transfer flask is transported to the shaft area and returned to the surface, whilst a separate rail-mounted vehicle completes the hole backfilling process.
It is intended that full-face shaft-boring machines will be used to sink the six vertical shafts. This is presumably intended to reduce the potential for vertical groundwater migration, by reducing the risk of peripheral disturbance associated with conventional drill and blast methods of excavation. All shafts are initially excavated at 4m diameter, and the two 5m shafts are subsequently reamed-out to full size. Tunnels are to be driven using drill and blast excavation using smooth-blasting patterns for shot firing.

Following initial development of the shafts and service tunnels over a period of twelve years, the emplacement tunnels are to be excavated three at a time from both ends (6 operating faces). Tunnelling is to proceed on a 3-shift system, 5 days per week, with rotation of specialist crews for drilling, blasting and spoil-removal cycles; and a period of twenty-four years is envisaged for the construction of the 187 km of emplacement tunnels within the repository.

This represents an average progress rate of less than 2m per shift. Based on U.K. experience, these progress rates appear slow, and would indicate an uneconomic rate of construction.

Drilling of emplacement holes is to proceed simultaneously with the construction of the emplacement tunnels. A total of 6216 holes is required, and it is envisaged that these will be drilled at an average rate of 222 holes per year, using six rigs. A pair of rigs will operate in each tunnel, with each drill averaging one hole per week. Drilling will therefore require twenty-eight years for completion.
Emplacement operations will commence fifteen years after the start of construction (i.e. on completion of the first three emplacement tunnels), and will continue over a thirty-year period. Thereafter, the tunnels and shafts will be gradually backfilled, after removal of equipment and services. The period stipulated for this phase of the operation is thirty-five years, comprising thirty years for the filling of the emplacement tunnels and five years for the service tunnels and shafts. Thus, access to each emplacement hole is to be maintained over a thirty-year period, allowing for monitoring and retrieval (if required) over this period.

Based on the above, the overall programme for construction, emplacement and sealing covers a period of 81 years, and is made up as follows:

- **Years 1-12** Construction of shafts and service tunnels;
- **Years 13-36** Construction of emplacement tunnels;
- **Years 14-41** Drilling of emplacement holes;
- **Years 16-45** Emplacement of waste units;
- **Years 46-81** Backfilling and sealing

### 13.7.4 Swedish Proposals

In the Swedish high-level waste repository scheme, illustrated in figure 50, the initial development comprises the sinking of the four shafts, the principal transportation tunnels, and an experimental waste emplacement area, within a total period of about four years. Emplacement tunnels are to be constructed subsequently in four 'quarters', with waste emplacement commencing after completion of the first quarter.
As for the other proposals, drill and blast methods of excavation are envisaged with minimal ground support in tunnels, and provision of continuous in situ concrete linings in the shafts. Figure 56 indicates the principal dimensions of the emplacement tunnels and boreholes. It is notable that the tunnel dimensions are considerably smaller than for the concepts previously described. However relatively large (1.0m) diameter emplacement holes are specified to accommodate the 0.61m diameter waste units, which incorporate a lead-titanium overpack; see Table 11.

The planned period for construction of the emplacement tunnels is thirty-three years, involving the construction of some 39 km of tunnels. Thus, the average rate of tunnel excavation envisaged is approximately 23m per week. In order to complete the 9,000 emplacement holes at the same rate, it would be necessary to complete an average of approximately 5 holes per week.

As in the case of the proposals previously described, it is difficult to envisage a realistic working schedule compatible with this rate of progress. For example, a single-shift system with two working faces operating on a five-day working week, would achieve the design rate with only one excavation cycle (drill, blast, muck-out) for each working face per day. A single drill could proceed at the rate of approximately 1m per hour with an allowance of approximately half a shift to move between successive hole positions.

As with the Canadian proposals, the ventilation system is designed to avoid the flow of air from areas of waste emplacement to areas of construction activity. The proposed ventilation system is shown in figure 57. Fans
at ground level blow air down the ventilation shaft, and the lateral distribution of air is regulated by dampers fitted at entrances to the emplacement and transportation tunnels. Air is first directed around the peripheral transportation tunnels and thence towards the central tunnel, where small diameter vertical shafts connect to a separate exhaust air tunnel immediately above. The exhaust air tunnel is connected to the rock hoist shaft which also acts as an exhaust ventilation shaft. Movement of air is assisted by additional fans inside the exhaust air tunnel and at the top of the exhaust shaft.

As shown in Table 13, the design capacity of the system is 9000 waste units; a relatively small number by comparison with other proposals. Based on a 20-year emplacement phase, this implies a mean delivery and emplacement rate of 300 units per year, or one unit per day. This relatively low level of activity explains why tunnel dimensions have been kept comparatively small and (with the added regulation by dampers in un-manned sections) ventilation system demands are relatively modest.

The waste units are transferred from a sub-surface temporary storage facility, close to the surface, via a waste transfer shaft as illustrated in figure 58. Due to the partial shielding provided by overpacks, the waste transfer flask is relatively light-weight. On arrival at the shaft bottom, the flask is to be drawn through the tunnel system on rails, using electric locomotives and is then positioned above the emplacement hole. The waste unit is then lowered into the hole, using a hoist inside the transfer flask and a temporary shielding plug is placed over the hole as shown in figure 58. A separate rail-mounted vehicle subsequently
removes the temporary shielding plug and places backfill within the hole.

13.7.5 United Kingdom Proposals

The British conceptual high-level waste repository scheme, illustrated in figure 51, differs from the remainder in terms of the proposed sequence of development. The relatively large 'Harvest' canister and high fission product concentration of the vitrified waste, as shown in Tables 11 and 12, result in a total planned waste unit production of only 4000 units. Hence, the assumed waste production period of 20 years implies an average disposal rate of only three units per week; and, on this basis, simultaneous phased construction has been shown to be uneconomically slow (144). The scheme therefore envisages that all construction will be completed in advance of the waste emplacement phase.

In general, the scheme is less fully developed in engineering detail than for the other countries under review, and represents one of a number of alternative high-level waste repository feasibility design studies which have been considered in the United Kingdom (17). As in the French proposals, relatively deep emplacement boreholes are considered (varying in depth up to 300m) so as to create a relatively compact cuboidal arrangement; see 13.6 above.

Smooth-blasting techniques are proposed for repository construction, and in view of the relatively shallow repository depth of 300m and the compact configuration adopted, it is estimated that construction could be completed within approximately 3 years, based on conventional rock tunnelling progress rates. Because of
the sequence of development, the need for duplication of shafts and tunnels (to avoid any physical conflict between construction and waste emplacement activities) is avoided.

13.7.6 United States Proposals

Predictably, the American high-level waste repository design proposals are based on comparatively large quantities of waste. The total quantity of units is 85,300, which is less than for the Canadian proposals. However, each unit contains a significantly higher proportion of fission products; see Tables 12 and 13.

As for the Canadian proposals, the repository system is based on concurrent construction and waste emplacement, and complex arrangements of tunnels are employed to ensure physical separation of associated activities. Construction is also based on conventional drill and blast methods, and diesel-powered plant is employed for all mobile equipment and haulage vehicles.

As shown in figure 52, the emplacement tunnel network is based upon a panel system which extends laterally from three central distribution tunnels. A fourth tunnel runs approximately 20m below the three main distribution tunnels and is used for haulage of broken rock from the excavations, with a series of vertical chutes at designated loading areas connecting to the overlying tunnel development.

Each panel comprises 40 parallel, 174m long emplacement tunnels at 23.7m centres, connecting to a single 960m transverse branch tunnel. The opposite end of each emplacement tunnel connects with a perpendicular
exhaust-air branch tunnel to convey the spent air from emplacement rooms towards the peripheral exhaust tunnel system; where it is directed, via a series of filters, to the exhaust-air shaft.

Figure 59 shows the arrangement of emplacement tunnels and boreholes. Within each emplacement tunnel, boreholes are drilled to 6.1m depth, in rows of two, with lateral spacings of 1.8m and longitudinal spacings of 3.05m.

A single waste unit is to be emplaced in each borehole (planar configuration). Therefore, a total of 85,300 emplacement holes are required, with 104 holes in each full-length emplacement tunnel. Two 'half-panels', of 87m width, are scheduled to be constructed first, so as to allow emplacement to commence shortly after the initial panel development. Thus, 19 full-panels and two half-panels are to be provided.

Borehole and tunnel spacings have been determined from thermal design criteria although, as previously described, the corresponding design value of maximum rock wall temperature is undoubtedly excessive. It is apparent that, like Canada's conceptual model, the waste units are closely-spaced, especially in view of the comparatively high initial heat output of the waste units.

All five shafts are to be sunk simultaneously, within a construction period of approximately three years, and are to be provided with nominal 0.3m thick concrete linings. Jumbos, with boom-mounted electric-hydraulic drills, are proposed for tunnel development, with an expected advance of about 3.5m for each complete excavation cycle. Diesel-powered, front-end loaders are
proposed for mucking the rock, with 35-tonne diesel dump trucks to transport the spoil to loading chutes, offset from the main distribution tunnel. Six faces are to be worked simultaneously, with rotation of specialist crews for drilling, blasting and mucking operations. A fourth crew is used for scaling-off loose rock and rock-blasting, as necessary. Additional construction workers are required to operate the materials handling system in the low-level tunnel, where rail-haulage and conveyor systems are to be used to remove tunnel spoil to the shaft area during the panel development phase.

The five shafts are to be located in a central part of the repository, and at one end of the high-level waste disposal area. The sizes and functions of the shafts are to be as follows:

<table>
<thead>
<tr>
<th>Shaft Internal Diameter</th>
<th>Purpose</th>
</tr>
</thead>
<tbody>
<tr>
<td>4.3 m</td>
<td>Transfer of high-level waste units</td>
</tr>
<tr>
<td>3.0 m</td>
<td>Transfer of drums containing low-level wastes</td>
</tr>
<tr>
<td>8.2 m</td>
<td>Transfer of men and materials</td>
</tr>
<tr>
<td>8.5 m</td>
<td>Ventilation air intake</td>
</tr>
<tr>
<td>8.5 m</td>
<td>Ventilation exhaust</td>
</tr>
</tbody>
</table>

The three parallel horseshoe-shaped main distribution tunnels are each 9.1m in width and 7.6m maximum height. These comprise:

- a central tunnel for transport of construction
equipment and personnel and delivery of fresh air supply for all construction operations

- a waste transportation tunnel for distribution of waste units to all parts of the repository and supply of fresh air for emplacement operations

- an exhaust air ventilation tunnel, solely for the removal of dust-laden air and fumes from construction areas.

It is significant that, as in the Canadian conceptual design proposals, no mention is made of special techniques to reduce peripheral rock fracture, such as smooth-blasting, either for shaft or tunnel construction; and it may be noted that the use of diesel-powered equipment with a multiple-face working system will place an exceptionally heavy demand upon the repository ventilation system.

Following completion of the shafts, a two-year period is allowed for development of service and distribution tunnels within the vicinity of the central shaft area, and for completion of the first half-panels in each repository replacement zone. Waste emplacement is thus scheduled for commencement within five years after the start of construction. The subsequent phase comprises the completion of the main distribution tunnels and the peripheral ventilation tunnels, followed by the development of panels, with simultaneous waste emplacement.

Surprisingly, no provision is made for exploring ground conditions at an early stage, by extending the central and peripherals tunnel system, before making a commitment to panel development. Provision for
exploratory works of this nature is considered an essential part of a repository construction programme, but has been absent from all the published international proposals, except the Canadian scheme as described above.

Following construction of each emplacement tunnel, hole drilling is to proceed using a number of electrically-powered units, each capable of drilling the 6.1m deep, 0.5m diameter holes at a rate of one hole per shift. Each hole will receive a 0.4m diameter mild-steel sleeve, with a closed bottom, as shown in figure 59c. The sleeves are to be formed from two 3m long pre-fabricated cylinders, welded during installation, with centralising spacers on the outside. The function of the sleeves is not specified, but they may be intended to facilitate short-term retrieval or to prevent rock spalling due to heating effects. Presumably, it is also intended that a close-fit will be provided between the waste units and the sleeves, in order to ensure good thermal contact. However, there are clearly practical difficulties with this arrangement, and evaluation cannot be made without access to further technical details.

Within the emplacement room, the vehicle is to be positioned over an empty drill hole and the transfer flasks are to be returned to the vertical position. After precise alignment is complete, one of the waste units is lowered into the hole. The vehicle is then re-aligned over the adjacent hole to emplace the second waste unit. After emplacement, a concrete plug is lowered to provide radiation shielding. The transporter vehicle then returns to the shaft area to commence another emplacement cycle. Details of the proposed rate of waste unit delivery and cycle times for emplacement
are not provided, although it is stated that five transporters are required.

The basic ventilation circuit has already been described. However, it is noted that all sub-surface plant, with the exception of the rock-waste haulage system, is to be diesel-powered. In consequence of this and the extent of the underground activity, the required ventilation air throughput would be considerable. Substantial fire precautions would also be necessary, since a large underground fuel storage capacity is also envisaged, and it is not clear how the ventilation system would cope with a major incident involving fire.

Eight centrifugal fans are to be used for ventilation, supplying a total of 55,000 cu m of fresh air per minute during periods of peak activity, with two back-up units for use in the event of a mechanical failure. At the base of the ventilation shaft, the air flow is divided into the separate air supply systems by controlling dampers. Controlling doors admit air into panels where emplacement operations are in progress, preventing flow elsewhere. Nevertheless, it is noted that the three main distribution tunnels are made considerably larger than the emplacement tunnels, purely to reduce air intake velocities to acceptable levels.

Exhaust air from emplacement and construction areas is separated within the exhaust shaft by a dividing wall. Air from emplacement areas is filtered continuously, whereas air from construction areas is treated only if contamination is detected.
13.7.7 Other Proposals

The foregoing account indicates the importance characteristics of the proposals for construction and operation of high-level waste repositories in crystalline rocks, within the five principal countries under review.

The proposals for saliferous rock repositories put forward by the Netherlands and West Germany are similar in several respects. Conventional drill and blast excavation methods are envisaged, with little or no rock support. Because of continued creep movement, underground transport by rail-systems is generally discounted, and the use of diesel-powered rubber-tyred vehicles is envisaged. Also, because of the relatively rapid convergence of emplacement holes under the influence of heat emitted by high-level waste units, the effective stand-up time is limited and concurrent construction and waste emplacement programmes are regarded as inevitable.

The Belgian proposals for a repository in the Boom Clay as illustrated in figure 53, provide few details of the proposed construction and waste emplacement process. It is evident that the basic feasibility of constructing the facility has not yet been fully established. It appears that current ideas are based on the use of ground-freezing throughout, since the proposed rectangular configuration with multiple tunnel junctions does not lend itself to shield tunnelling methods. Massive segmental nodular cast-iron linings are envisaged throughout the tunnel system, and it appears that these are to remain in place after backfilling and sealing. Not surprisingly, in view of the foreseen construction difficulties, tunnel dimensions are relatively small (4.5m and 3.5m diameter; see figure 53).
As described in Chapter 10.3, saliferous and argillaceous rocks are likely to be overlain by comparatively unstable, water-bearing strata. Corresponding repository design proposals therefore place considerable emphasis on special techniques for sinking and lining the shafts. This aspect assumes over-riding importance in the case of saliferous rock repositories, where shaft penetration through the cap-rock and into the body of the host rock may provide a route for groundwater invasion.

Proposals put forward by Belgium, the Netherlands and West Germany all involve the use of ground-freezing techniques and the installation of massive 'water-proof' shaft linings. An illustration of the proposed shaft lining construction for the West German repository scheme is shown in figure 60. The installation of the composite shaft lining is claimed to provide a completely impermeable barrier, extending well in the host rock. However, whilst the transverse permeability of the lining is likely to be very small, it is not clear how the lining will eliminate vertical groundwater flow paths in the region of the rock-lining interface. It is well known that ground-freezing produces an expansion of the affected zone, followed by contraction when the coolant is no longer circulated. It therefore appears likely that vertical flow paths could be induced, leading to deterioration and increased flow in the vicinity of the plaster packing at the extrados of the lining. This aspect also appears to have been neglected in the proposals put forward by Belgium and the Netherlands.

The Danish proposals, involving the construction of a matrix of drillholes, are unique. The proposed system
comprises 8 drillholes, completed to depths of approximately 2500m. The upper portion of each hole is to be cased over a length of 950m or 200m into the salt dome, whichever is the greater; and the remaining lower section is to remain un-lined, as shown in figure 61.

The hole construction procedure involves four stages as follows (67).

(a) A 1915mm steel caisson is sunk through the upper loose, water-bearing Quaternary strata, to a depth of 30m.

(b) From the base of the caisson, a 185mm diameter hole is drilled to a depth of 200m through Cretaceous strata, using reverse mud-circulation. A 1320mm internal diameter casing is inserted during the drilling process and, on completion, the casing is cemented into place by forcing cement along the outer annulus to displace the bentonite. A cement retarder is used to allow 4-5 hours for this operation.

(c) From the 200m level, the hole is extended at 1200mm diameter, using reverse mud-circulation, and an 812mm diameter casing is installed. This section of the hole is completed to a depth of 950m below ground level or approximately 200m into the salt (i.e. 200m below cap-rock); see figure 61a. The second-stage casing string is then cemented-in as described in (b) above.

(d) The lower portion of the hole is drilled using direct mud-circulation without any casing. Initially, a pilot hole of 445mm diameter is drilled to a depth of 2500m. The hole is then
reamed-out to its full diameter of 750mm.

After the completion of the drilling process, each hole is full of bentonite slurry. This is subsequently displaced by brine, using a tremie-pipe. It is envisaged that a volume of brine equal to 2-3 times the volume of the hole will be required to remove all the drilling mud, so as to leave a clean hole.

The waste units are to be 0.66m in diameter and 6.0m in length. These are to be lowered into the brine-filled hole at a rate of about 415m/hour and each unit is fitted with three centralisers to ensure proper alignment. The annular space around the units (nominally 45mm) is filled by pre-batching a suitable quantity of cement slurry to the base of the hole. During the final stage of their travel, the units are pressed into the cement, forcing it into the annular space between the units and the borehole walls.

The emplacement hole fill comprises standard 'deep-hole' cement, with added sodium chloride, and a retarder which maintains the material in a fluid condition for about 50 hours. It is envisaged that about 5-10m$^3$ of cement fill will be placed per batch, allowing some 5-10 waste units to be placed in each cycle. Each hole is designed to receive 200 units within the depth interval 1200-2500m.

On completion of waste emplacement, each hole contains an empty upper section of about 1200m depth. This section remains filled with brine, and is cased and cemented to a depth of 950m, but is unsupported with a 'clean' salt wall from 950-1200m. The sealing system envisaged for the whole of the empty upper section is shown in figure 61b.
A 50m long plug of cement containing sodium chloride as an additive, is placed immediately above the emplacement zone. The plug is placed using a tremie and, after curing, the overlying brine column is removed by air-lifting.

Above the concrete plug, crushed saliferous spoil is placed over a thickness of 180m, to a level of some 20m below the bottom of the cased section of the hole. The remainder of the hole, to a depth of about 2m below ground level, is to be filled with 63% bitumen and 27% pulverised limestone.

The 20m thickness of asphalt within the unlined section of the hole is intended to provide an effective water seal, based on the fact that the pressure of the overlying asphalt exceeds the maximum possible hydrostatic water pressure which could ultimately develop in the overlying casing cement. It is also assumed that the crushed salt plug will anneal under ground pressure; and will re-crystallise to form a homogeneous salt mass. The latter assumption merits some serious investigation in relation to the backfilling of saliferous rock repositories in general; see Part 4.

It would appear that Elsam/Elkraft have assumed that the casing cement will ultimately deteriorate with time and could therefore create a vertical flow path. However, the deterioration of the cement introduced around the waste unit column is not mentioned. This cement slurry will be subjected to heating during the curing process, and it may be anticipated that some steam will be generated. By implication, it appears that deterioration in the near-field cement backfill is considered to be of little consequence, since
significant borehole closure may be anticipated after placement.

13.8 Backfilling and Sealing

With few exceptions, the international proposals for repository backfilling and sealing, described within the design literature, are poorly developed. All of the high-level waste disposal concepts under review rely upon the immediate placement of a borehole backfill to provide rock/waste thermal contact and short-term radiological shielding protection. Borehole backfilling materials and procedures have therefore received priority consideration. However, many of the borehole filling proposals are open to serious criticism and several of the proposals for the bulk filling of tunnels and shafts indicate that this aspect of repository design has received comparatively little serious consideration by the authorities concerned.

The proposals involving the in-floor disposal of high-level wastes, described in 13.7 above, generally involve the use of special mobile transporter units capable of centering themselves above the emplacement hole to lower the waste units into place. A separate mobile unit then delivers a pre-batched quantity of borehole fill and compacts it into the annular space around the waste units.

13.8.1 Canadian Proposals

In the Canadian design proposals, a specially designed vehicle is used for the backfilling of emplacement
holes, based on the following cycle:

- the vehicle lowers a cylindrical shield over the emplacement hole and removes a temporary shielding plug from the top of the hole.
- a batched quantity of crushed sand is placed in the hole to fill the annulus between the cylinder and the wall of the borehole.
- the upper 1.5m of the hole is backfilled with a compacted sand/bentonite powder mixture.

No information is provided concerning the exact method of placement and compaction, or the moisture contents and gradings of the materials used. Emplacement and drillhole backfilling operations take place concurrently in different rooms, to avoid any interdependence between the two operations.

The sand/bentonite mixture is presumably intended to generate a significant swelling pressure on ingress of water from the surrounding rock. However, no specific measures are described to provide the necessary restraint against the possibility of swelling in the short-term.

Individual emplacement rooms are backfilled when waste unit emplacement and hole sealing is complete. Thus, emplacement rooms are gradually backfilled in parallel with the waste emplacement operations, working progressively from the area of exhaust ventilation shafts towards the service shafts. When all emplacement rooms are backfilled, backfilling of lateral and longitudinal drifts takes place, followed by sealing of shafts and decommissioning of surface facilities. These
final backfilling operations are scheduled to take place over a period of five years; see 13.7 above.

The backfill used for filling rooms, drifts and shafts is intended to provide a relatively impermeable barrier to the flow of water in the rooms and drifts and a conductive medium to transfer heat from the waste to the surrounding rock. The proposed backfill comprises:

- 55% crushed rock coarse aggregate
- 35% pit-run sand and fine gravel aggregate
- 10% bentonite

Mention is also made of the need to prepare the rock surface in emplacement rooms to ensure than an impermeable rock/backfill interface is created. No specific proposals are made, but presumably some form of rock-scaling operation is envisaged.

Aggregates for bulk backfilling operations are to be prepared at a surface crushing and batching plant and are to be delivered, via a pipe in the service shaft, to a storage bin at the backfill rail haulage level. The aggregates are then off-loaded at passes under the backfill haulage drift, via vertical holes, to the underlying central drift.

Proposals for the placement of fill in the rooms and drifts are based on pneumatic (stowing) methods. Bentonite and aggregates are combined at the mixer, with addition of water at the nozzle, although the moisture content and residual swelling potential of the emplaced material are not indicated. Either pneumatic or mechanical placement methods are envisaged for the filling of shafts, with breaking-out of the concrete lining at specific locations to key the backfill to the
It is apparent that the Canadian backfilling and sealing proposals, as reflected in the conceptual repository design proposals, are at an outline stage (35). However, a series of specialist studies is also in progress to evaluate alternative backfilling material formulations.

Geochemical assessments of various backfill materials are being carried out by the Research Chemistry Branch of the Whiteshell Nuclear Research Establishment (25). Eight potential backfill materials are being investigated, namely:

- Bentonite clays with crushed rock and quartz sand
- Synthetic swelling compounds such as magnesium oxide and silica gel
- Metallic compounds including oxides, sulphides, carbonates and phosphates of copper, lead, iron and manganese
- Clay minerals such as kaolinite and illite
- Graphite
- Zeolites
- Anhydrite
- Crushed rock spoil

In addition, backfilling and sealing studies are being carried out under the direction of Ontario Hydro, in
association with the Canadian Centre for Mineral and Energy Technology (CANMET). Published literature suggests that current efforts are principally directed towards the development of procedures for the sealing of shafts, and emphasis is placed upon ground treatment of the peripheral rock mass (63).

13.8.2 French Proposals

In the French high-level waste repository design proposals, the 1.0m diameter high-level waste emplacement boreholes provide a 0.3m wide annulus around the waste units. Following emplacement of a waste unit, a backfill comprising 15% dry bentonite and 85% crushed granite is poured over the unit and is compacted by special vibrators drawn upwards through the material. Although not specifically stated, it is assumed that the backfill is to be brought up to the level required to achieve the desired vertical separation between units. Water is then added to allow the bentonite to swell and fill the voids within the material. The upper 2m of each hole is subsequently filled with pure bentonite to provide a seal at tunnel floor level.

It is assumed that (as for the Canadian proposals) the high crushed rock aggregate content of the fill is intended to provide adequate thermal conductivity in the near-field, compensating for the low thermal conductivity of the bentonite, and the bentonite is introduced in order to promote a tight annular seal by virtue of its swelling properties. However, any wetting of the fill between successive waste unit emplacements would promote an upward expansion within the hole, in the absence of constraint, and should thus be expected to alter the vertical spacings between waste units. This
aspect of the proposals therefore appears questionable.

The proposed tunnel and shaft backfilling techniques are considerably different from those described elsewhere. A 300mm layer of compacted bentonite is placed on the tunnel floors, and pre-formed 300mm thick dry bentonite 'segments' are placed to form a tunnel 'lining' around the walls and crown. Transverse bentonite walls are then formed across the tunnels at 100m intervals, using pre-formed bentonite blocks, and the cavities between successive walls are filled by pumping a bentonite-aggregate fill, comprising 10% bentonite and 90% crushed granite.

Shafts are to be filled in a similar manner, using pre-formed bentonite lining segments with placement of a mass-infilling of bentonite aggregate. Layers of pure bentonite are placed at 100m intervals to form transverse barriers, as in the horizontal tunnels.

Clearly, it is intended that the outer 'skin' of bentonite will ultimately exert a swelling pressure or undergo an increase in volume to ensure a dense infilling of the bulk excavations. Although no specific reference has been found, it may also be intended that the swelling bentonite will seal any significant peripheral fractures in the rock.

Since the publication of the French 'Geostock' study report, considerable geoscientific research has been undertaken in relation to backfilling and sealing, and particular efforts have been made to identify materials having favourable ionic sorption properties (182, 183).

Eight naturally occurring minerals have been studied in order to assess their cation sorption capacity, namely:
o bentonite
o illite
o kaolinite
o vermiculite
o attapulgite (palygorskite)
o bauxite
o clinoptilolite
o sepiolite

Of these, kaolinite, sepiolite and vermiculite have been rejected and the remainder are to be subjected to further detailed investigation.

Similar geochemical research has been implemented to identify potential anion sorbants. Various compounds of iron, copper and lead have been identified as the most promising candidates. The long-lived anionic species iodine and technitium have been used as a reference for these studies, and chalcopyrite, galena and siderite have been identified as potentially beneficial in terms of their anion retention properties.

13.8.3 Swedish Proposals

In the Swedish high-level waste repository design proposals, although backfilling of the emplacement holes takes place immediately after the positioning of each waste unit, backfilling of the tunnel network does not commence until the repository is completely full. An allowance is also made for a substantial period of monitoring prior to backfilling. Thus, it is intended that the repository tunnels will remain open for at least thirty-five years, and perhaps considerably longer, before sealing.
Each drill-hole hole is to be prepared prior to waste emplacement. The hole is first drained of any groundwater. A 300mm bed of sand-bentonite (90% sand, 10% bentonite) is then placed and tamped into position, titanium irrigation pipes are installed, and a mobile radiation shield is positioned at the mouth of the hole.

The waste unit is then lowered into the hole, to rest on the prepared sand-bentonite bed. The transfer shield is removed and the hole is filled with sand-bentonite fill (85% sand, 15% bentonite). The fill is compacted in layers 10-20mm thick using a special tamper so as to provide a cover of nearly 3m above the top of the waste unit, which provides adequate radiation shielding. The hole is then sealed with an in situ concrete plug and a precast concrete beam is placed and grouted into recesses in the tunnel walls. This arrangement enables the bentonite fill to generate its full swelling pressure on absorption of water, so as to achieve the maximum possible density.

Irrigation of the sand-bentonite fill is maintained throughout the remainder of the commissioning period, so as to prevent the sand-bentonite material from drying out due to the heat generated by the waste units. However, it is understood that KBS are carrying out further studies into the effects of heat generation upon the backfill, in order to determine whether the irrigation system could be simplified or dispensed with. It appears to the author that the introduction of irrigation conduits and deliberate introduction of free water in the emplacement holes is contrary to the basic objectives of waste disposal.

Figure 62 illustrates the proposed backfilling
operations in the repository tunnels. Prior to backfilling, the ventilation dampers in each tunnel are replaced by stop-ends, formed by stacking pre-cast concrete planks, recessed into the tunnel walls. The proposed tunnel fill material is a sand-bentonite mixture similar to that envisaged for the waste emplacement holes. The fill material is delivered by conveyor, and is spread by tractors and compacted by vibrating rollers (concrete lids and beams having first been removed from the waste emplacement holes).

Having placed the fill to the maximum practicable horizontal level, the upper portion is filled with the same material by pneumatic spraying. It is claimed that this method would achieve a high level of compaction due to the swelling properties of the bentonite. Shafts are backfilled using a similar material, although the report does not describe the method of placement. It is proposed that monitoring of temperature, stresses and groundwater movements would take place for a period after sealing of the repository, although no details of the measurements to be made or instruments to be used are provided.

In common with other concepts, the Swedish KBS proposals have placed considerable reliance on the containment properties of bentonite-based backfilling materials. The high swelling capacity of the material is utilised to achieve a dense, low permeability infill, and numerous technical studies have been undertaken for KBS, by Pusch et al., at the University of Lulea to verify this concept (170, 171, 173, 174, 175, 176, 177, 178, 179).

In addition to the use of dry powdered bentonite as proposed in the 'KBS-1' report, research into the
properties of pre-formed compressed bentonite blocks has been carried out (171, 176). Research has also indicated that the swelling capacity of bentonite may result in the self-injection of rock fractures, although electro-phoresis has also been advocated as a means of producing greater penetration of fissures in the vicinity of waste emplacement holes (175).

The physical properties of bentonite in terms of bearing capacity, consolidation characteristics, plasticity and moisture content-density-permeability relationships have also been investigated (179). In relation to heat transfer properties it has been found that bentonite alone is a poor conductor of heat and the addition of quartz sand is required to improve the overall thermal conductivity of the backfill material. In this context, the beneficial effects of the addition of sand have balanced against a corresponding increase in backfill permeability (173, 174).

Major studies have also been implemented to assess the long term geochemical and physical stability of bentonite mixtures. Radionuclide retention properties have been studied in detail, and the high ionic exchange capacity of bentonite is put forward as a significant attribute of the material (173, 174).

Various other backfilling materials have been examined at a subordinate level, including illites and bitumens (68). However, bentonite remains the focus of the Swedish backfilling research programme and is the fundamental constituent of backfills adopted for conceptual repository design purposes.
13.8.4 United Kingdom Proposals

The United Kingdom repository design proposals do not provide any detailed information concerning backfilling materials or methods. However, recognition is made of the practical difficulties of backfilling the annular space around waste units in deep emplacement holes, and a minimum 100mm thick annulus is specified (144). It is also suggested that a measured quantity of pre-mixed backfill comprising bentonite, sand and water, should be placed by a tremie-pipe system prior to waste unit emplacement.

It is claimed that a fully effective filling of the annular void can then be guaranteed by lowering the waste unit into the fluid so as to displace the fill material to the required new level. However, although the immersion method ensures that all void spaces are filled, it is apparent that the necessarily high moisture content and lack of swelling restraint would negate the benefits of utilizing a swelling material. A cementitious fluid fill could also generate interface separation planes, due to shrinkage during the hydration process.

No further proposals have been put forward in the context of a detailed repository scheme. However, a brief review of the backfilling and sealing problem has been carried out by the Building Research Establishment (36) and a detailed backfilling and sealing study has been carried out by the author, under the Commission of the European Communities' second indirect action programme (7).
The American design proposals for a granitic high-level waste repository indicate a general lack of attention to backfilling and sealing aspects. A complicating factor is the incorporation of a five-year long waste retrieval option following initial waste emplacement. As previously noted, steel sleeves are to be inserted within the boreholes and during the specified five-year interval no borehole backfill is provided. Thus, the transfer of heat from the waste units to the surrounding rock must presumably occur primarily by radiation across annular air gaps between the waste units, steel sleeves and borehole walls; see figure 59.

After the five-year period, a special vehicle removes a temporary concrete shielding plugs from the emplacement holes, and backfills the openings with crushed rock. Presumably this material is intended to fill the annulus around the sleeves, although no details of the aggregate size or method of placement are provided. Thus, no consideration is given to the provision of dense borehole backfill to provide good waste-rock thermal contact, and no attempt is made to introduce an engineered barrier in the form of a low-permeability fill (possibly with high ionic exchange capacity).

Similar shortcomings arise with respect to the back-filling of tunnels. It is proposed that tunnels will be backfilled with broken rock obtained from the construction process. Rock-spoil material is simply dumped by trucks and then placed by front-end loaders to within 1.5m of the tunnel crown. No grouting is carried out, and the filling is therefore incomplete, and of high permeability.

It is stated that different materials and techniques are envisaged for the backfilling and sealing of shafts,
although these are yet to be developed. However, although substantial backfilling research studies are known to have taken place in the United States, it is evident that these have not been considered as an integral part of the repository design process.

In fact, fundamental research in America on the properties of potential backfilling materials has been relatively sophisticated. As noted in 13.2 above, the 'Borehole Plugging Programme' arose largely as a result of the discovery of un-sealed boreholes in the vicinity of the proposed experimental saliferous rock repository at Lyons, Kansas. During the period 1973-78, the 'Borehole Plugging Programme' included the following aspects:

- evaluation of cementitious backfill materials, including hydrothermal cements (25B, 85, 189).
- research and development in relation to borehole sealing by rock-melting methods (25A)
- evaluation of materials and methods involving the use of naturally-occurring materials (69)

The relevance of the above has essentially been limited to the sealing of exploratory holes, and the size of the research effort which has been devoted to this topic, largely for political reasons, appears somewhat disproportionate. However, other useful backfilling and sealing studies have been carried out under separate geoscientific research programmes for specific nuclear sites. These are:

- Studies undertaken by Sandia National Laboratories for the WIPP project for disposal of TRU military
wastes in salt (46, 86, 110). The potential repository backfill materials included in this research programme comprise hydraulic (Portland) cements, crushed rock salt and bentonite. Consideration has also been given to the development of composite backfill systems (149).

Studies undertaken by Woodward Clyde Consultants for the Basalt Waste Isolation Project (BWIP) at the Hanford nuclear installation. Materials studies have included crushed basalt, quartz and various cements. The physico-chemical retention properties of smectite clays (bentonite and hectorite) have also been examined (101, 201, 223).

13.8.6 Other Proposals

Repository backfilling and sealing proposals put forward by other countries also vary considerably in terms of the scope of materials research and design. Proposals put forward for the Belgian repository system (98) envisage that the extrados of the steel casings used to line high-level waste emplacement holes will be backfilled with an unspecified material having inherently 'low permeability', 'high plasticity', 'good thermal stability' and 'appropriate radionuclide retention properties'. In addition, the 50mm gap between the waste units and the borehole casings is to be filled with a uniformly graded, fine quartz sand; providing for heat transfer from the waste units and allowing for short-term retrieval during the operational phase.

It is also proposed that backfill will be placed behind all tunnel linings, as construction proceeds, by injection of clay or cementitious grouts. The materials
for bulk filling within the tunnels are unspecified, but mention is made of cement/clay mixtures, with the possible addition of mineral components having sorptive properties. A staged infilling of intermediate-level waste emplacement tunnels is proposed, in parallel with the emplacement process. However, the infilling of high-level waste emplacement tunnels is to be a single-stage operation, to be put into effect when the retrieval option is no longer required.

The repository shafts are to be backfilled with clay which is placed and compacted in a manner which duplicates, as far as is practicable, the characteristics of the undisturbed host material. For this purpose, the shaft linings are to be removed over a length which is greater than the thickness of clay penetrated. In order to achieve this, it is proposed to re-freeze the surrounding ground prior to the back-filling of the lower section of the shafts. All ground-freezing tubes are to be removed, and the holes are to be carefully injected with bentonite at the end of each stage of freezing. The upper portion of the shafts are to be backfilled with sand, and the linings are to remain in place over this section so as to provide lateral restraint during the backfill compaction process.

Whilst recognising the preliminary nature of the Belgian backfilling and sealing studies outlined above, a number of critical comments may be put forward. Aspects which do not appear to have been accounted for in the available literature are summarised below:

- The proposed system for infilling of intermediate-level waste emplacement tunnels involves the delivery of materials via a pipe in the
tunnel crown, without any form of compaction, as shown in figure 63. Such a system would inevitably result in an incomplete filling of the voids around the waste units and the creation of voids in the crown area. These unfilled sections would constitute preferential longitudinal flow paths within the system; see Chapter 10.4.

- The complete infilling of the 21m long, 50mm thick annulus in high-level waste emplacement holes is likely to pose similar problems, especially in view of the presence of 'spacers' which could cause cavities and prevent adequate fill compaction.

- The proposed sand infill is clearly not an impermeable barrier and will also contain air within the void spaces, contributing oxygen to the system. In addition, quartz sand will not contribute towards the geochemical retention of radionuclides.

- At the depths envisaged the host formation will behave as a 'squeezing' ground. Under these circumstances, the concept of placing a continuous backfill behind borehole and tunnel linings, by some form of injection process, appears questionable.

- Whilst the proposals recognise the need to investigate means of eliminating migration paths at the interface with tunnel linings or borehole casings, no mention is made of the possible long-term adverse effects of lining or casing corrosion upon the development of longitudinal flow paths.

- The proposed ground freezing approach to shaft construction will result in a loosening of the
ground behind the lining on termination of the freezing process. Current proposals appear to make no provision for the prevention of associated vertical flow phenomena or the provision of cut-off seals at the clay/overburden interface.

The Danish proposals for high-level waste emplacement in a matrix of deep drillholes in saliferous rock have been described in 13.7 above. Unlike the proposals based on tunnel network systems, backfilling and sealing aspects have been considered as an integral part of the design of the disposal system.

The Netherlands's proposals for a high-activity waste repository in saliferous rock envisage that shafts and tunnels will be backfilled with a mixture of crushed salt spoil and fly ash. Although initial proposals suggested the incorporation of rubble from the demolition of nuclear power plants (92), it appears that this proposal has since been abandoned (90, 91).

The bulk fill within the shafts and tunnels is not regarded as a high-integrity backfill, but reliance is placed upon the long-term ability of the shaft linings to prevent the ingress of water from overburden materials. Impermeable seals are to be placed at intersections between access and emplacement tunnels and at the junctions between access tunnels and shafts. Materials and methods of construction for these seals have not been described in the available literature.

Considerable attention is focused upon the waste emplacement cavities, and it is intended that the backfilling of these should prevent contact with invading groundwater, in the event of a flooding incident during the operational phases.
Figure 64 shows the proposed system for the backfilling of high-level waste emplacement holes. It should be noted that the holes are to be drilled to provide only a 50mm annulus around the waste units. This small gap is intended to close in the long-term, due to the inward creep of the rock salt under the influence of high temperatures. However, a granular infill is to be provided around the waste units to assist in the transfer of heat and to allow an equalisation of air pressure within the void spaces, due to the compression which accompanies borehole closure.

It is not clear whether this small diametral borehole tolerance will be compatible with the anticipated vertical deviations which could occur if drilling is extended to 300m. In addition, it is not apparent whether short-term creep effects within the salt have been taken into account when considering the emplacement of waste units. It is thought that the grading and method of placement of the granular fill within the annular zone will require careful consideration and may impose practical limits on both the minimum diameter and maximum depth of the holes.

The upper 7m of each emplacement hole is to be filled with a composite seal, and the upper 3m section is to be cased with a steel lining, fitted with a flange at tunnel floor level. The casing and flange have a dual function; allowing centralisation of the waste emplacement vehicle and subsequent attachment of a cover plate or cast steel plug after backfilling.

The lower 3m of the borehole plug is filled with a mixture of crushed rock salt, powdered dry clay and fly ash. The proportions of these materials and the type of
clay are unspecified. A further 2m of powdered clay is to be placed above this mixture, and the remaining portion of the hole is to be filled with salt concrete.

A layer of bitumen is to be placed over the finished surface of the concrete. On placement of the steel cover plate or cast steel plug the bitumen should be squeezed out to fill any remaining voids. It is also proposed that two layers of reinforced bitumen will be placed on the tunnel invert in the vicinity of each hole, in order to cover the top of the plug and steel cover plate, and provide further protection in the event of flooding.

The filling of emplacement holes is to take place in parallel with the deposition of waste units, so that no hole remains unfilled for any significant period. This is consistent with the assumption that accidental flooding of the open tunnel excavations during the operational phase is the 'worst case' design scenario to be considered.

In the event of flooding, the borehole seal components are required to function as follows (91):

- The reinforced bitumen layers should provide sufficient flexibility to sustain movements in the tunnel floors, due to thermal creep, whilst maintaining their sealing function.

- The concrete plug and steel casing are assumed to remain intact and to follow any movements of the tunnel floor. The length of the casing is assumed to be an order of magnitude greater than any induced fractures originating at the floor surface.
o The 2m long clay plug which extends partly into the steel casing should swell in the event of any seepage along the inner or outer edge of the casing, providing a 'second stage' barrier.

o The 3m long clay/salt/fly-ash plug should act as a final barrier in the event of groundwater invasion.

The clay component in the plugging system is assumed to possess radionuclide retention properties in addition to its low permeability. The fly-ash is intended to assist in the sealing and filling of interstices within the crushed salt mixture.

Consideration has also been given to various potential adverse effects within the system, namely:

o Air within the granular fill around the waste units could be forced upwards towards the top of the stack. It is assumed that this air will either remain entrapped or diffuse towards the backfill openings without causing any adverse effects. However, under the influence of heating, the rate of closure could be higher than the rate of diffusion of air into the overlying fill. In this event, an air pocket could be forcibly intruded into the fill material.

o Migration of fluid inclusions could result in an accumulation of brine in the granular material which surrounds the waste units. However, it is assumed that closure of the borehole walls will be sufficiently rapid to prevent deleterious effects. The basis for this assumption is the measurement of borehole wall closures of 2.5mm/month during experiments conducted in the Asse salt mine.
The possibility of a direct penetration of the repository by drilling has been considered and the final design may incorporate a conical cast-steel plug to deflect drills away from the waste units, as shown in figure 64.

An emergency hole plugging procedure has been developed for use in the event of flooding prior to the completion of waste emplacement in a particular hole. In this event, a contingency plan would be implemented, involving the placement of batched quantities of salt/clay/fly-ash and pure clay powder within a period of a few hours.

Proposals for the backfilling of intermediate-level waste bunkers are not described in detail in the recent literature, but the original design proposals envisaged backfilling with crushed salt only (92).

The West German conceptual design report for a saliferous rock repository (188) presents a brief description of a backfilling and sealing system, which includes the following components:

- crushed rock salt spoil for the infilling of the bulk excavations
- concrete/clay seals to separate the different waste emplacement areas
- impermeable shaft linings, passing through the overburden materials and 'keyed' into the upper 50m of the salt formation

The crushed rock salt is required to provide mechanical
support, and will not provide any significant contribution in terms of radionuclide retention or geochemical protection of the waste units. The shaft lining and concrete/clay seals are regarded as the principal barriers to the ingress of water; see figure 60.

However, a wider range of potential backfilling materials and methods are being studied within the context of the investigations at the Asse salt mine experimental facility (70). Proposed insitu tests include:

- borehole plugging tests
- full-scale chamber entrance sealing tests

The proposed borehole sealing tests are primarily related to the plugging of site investigation boreholes, but may also be relevant to the sealing of waste emplacement holes. Materials considered include:

- rock salt
- various cement formulations
- bitumen
- melamine resin
- steel and ceramics

These test materials are to be prepared in the form of pre-fabricated plugs, slightly smaller than the borehole section. Closure of the borehole walls under the influence of heat is expected to produce a tight plug/rock seal, and artificial heaters are to be employed during the field tests to produce the desired effect. The various plug materials are then to be subjected to pressure tests involving measurements of gas permeability. Additional laboratory tests are to be
performed to assess thermomechanical properties and resistance to brine corrosion.

The backfilling material to be used in the chamber sealing test is a PFA-based material from Poland, known as 'Katacit'. The material is in the form of a dry powder, requiring the addition of 25% of water for curing. A final compressive strength of 10 MN/m$^2$ is claimed, with a swelling capacity of about 10%.

The following instrumentation is to be installed in the test chamber:

- orientated flat-jacks to measure swelling pressures
- hydraulic pore-pressure probes
- gas permeability probes
- guide tubes for measurement of ultrasonic velocities
- levelling instruments to monitor rock movements

Additional laboratory tests are to be performed to assess various properties of 'Katacit' fill by taking core samples from the cured material within the test chamber.

The above summary indicates the apparent scope of relevant research activities in West Germany. However, no published results of the proposed insitu tests and associated laboratory experiments have been found in the available literature.
13.9 Monitoring and Retrieval

All of the available published repository design proposals make some reference to the need to incorporate monitoring facilities within the underground repository and, as shown in previous sections of this chapter, some also specify that a short-term waste retrieval option is to be incorporated. The latter is generally intended to cover the short-term period prior to the permanent closure of the repository.

However, in general, concepts appear to be poorly developed. In particular, the objectives of the monitoring and retrieval procedures are not clearly specified and, in some cases, would appear to jeopardise the containment system itself.

Both the American and Belgian proposals, for example, specify that permanent steel sleeves or casings are to be used to line high-level waste emplacement holes and uniformly graded sand is to be used to line the annular space around the waste units, in order to facilitate retrieval. In both cases, the specified width of the annulus is small (about 50mm) and it is difficult to envisage how an effective dense infilling of this space could be achieved. It is also evident that the presence of uniformly graded sand would increase the availability of groundwater for corrosion and leaching processes in the near-field and would provide a relatively high-permeability medium for preferential flow. The presence of the borehole linings could also exacerbate problems associated with interface flow phenomena; see figure 34.

In the author's view the incorporation of such measures,
in order to facilitate retrieval in the short-term, is contrary to the more fundamental objectives of long-term waste containment. Thus, provided that integrity of the waste units is thoroughly tested prior to emplacement, and validation trials have proved the efficacy of the planned disposal procedure, the emplacement process should be regarded as final. Whilst it may be prudent to devise a suitable emergency retrieval or remedial procedure to cater for the possibility of an unforeseen accidental failure of a waste unit during emplacement, it is thought that this should not be allowed to influence the design of the containment system itself. Hence, the need to ensure that waste emplacement and backfilling procedures are consistently effective in terms of long-term containment is considered to be of paramount concern.

Similar arguments may be applied in questioning the validity of suggestions for long-term monitoring of an underground repository system. Clearly, in view of the time-scale of the disposal problem, any monitoring programme must be regarded as 'short-term', and will therefore be of limited value. Here it is apparent that some sensible balance needs to be struck in relation to the value of monitoring systems and their potentially adverse effects upon the disposal system. In this context, it must be noted that systems involving 'permanent' physical inclusions and attachments (e.g. wires and cables) may have a deleterious effect on containment.

Some research into the development of appropriate monitoring and instrumentation equipment has been undertaken in America by Westinghouse Corporation (220) and by IRT Corporation (198); the principal objective being to develop equipment capable of assessing the
integrity of borehole seals, without resorting to the use of cable attachments. The most promising outcome of this work appears to have been the development of an insitu electromagnetic wave monitoring device for embedment within borehole plugs. The device measures changes in hydrogen concentration, which it is assumed will indicate entry of either water or gas, or ionisation due to radiation. However, the useful life of the instrument is limited by the relatively short life-span of the batteries.

Whatever the technical merits of such instruments, it must be accepted that they will inevitably present difficulties in interpretation, and their maintenance-free life will cover only a short-term interval during which waste containment is expected to be absolute. Hence, in the author's view, extensive instrumentation and monitoring should be considered only in relation to validation experiments which form part of the initial design process. Only fully-proven techniques should be applied during the final waste emplacement and backfilling stage, and special quality control procedures should be employed to ensure consistent and reliable results without the need for embedded instruments. Further consideration of these aspects is provided in Part 4.
In Part 3, it has been shown that a remarkable international consensus has emerged concerning the design characteristics of high-activity radioactive waste repositories. All countries envisage that high-level wastes will be immobilised by vitrification in borosilicate glass cast into cylindrical casings of stainless steel, so as to provide an initial man-made barrier to radionuclide release. Technologies have also been developed for immobilising intermediate-level wastes in cylindrical or cube-form solid waste units.

However, it has also been shown that the proposed high-level waste characteristics vary widely, even for similar rock types, and no clear consensus has emerged concerning the optimum size, shape, waste concentration, or storage period. The use of overpacks has been advocated by Sweden only, where their function is seen primarily as a physical barrier to corrosion by groundwater. None of the concepts for disposal into salt formations involves the use of overpacks, on the basis that no circulating groundwater will be present. Thus, it appears that none of the proposals recognises that overpacks may also play a useful role in improving the overall heat dissipation characteristics of the emplaced wastes.

All of the proposals place reliance upon the host rock as the primary barrier to radionuclide migration and, apart from waste conditioning aspects, do not foresee any need for sophistication in terms of repository design, as a means of enhancing the level of waste containment. It may be noted from Table 10 that repository design studies have often been carried out by independent groups, based on precepts established by the
nuclear establishment. These have generally assumed a simple generic set of rock properties, including permeability, strength, modulus of deformation, etc (see Table 17) and have required that the design concept should be based on straightforward adaptations of conventional underground design and construction procedures.

As shown, the resulting proposals generally comprise extensive networks of underground openings, with considerable redundancy of underground space to allow for access, ventilation and supply routes for different underground operations; see Table 10. Although the proposals for saliferous rock repositories have recognised that shafts will require special design treatment to avoid groundwater invasion, it also appears that none of the proposals has recognised or sought to mitigate against the problems of longitudinal migration and fissure flow phenomena in brittle rocks, as described in Chapters 9 and 10. For clay host formations, it also seems that the repository layout and waste emplacement configuration have been modelled on those advocated for high competence rocks, without recognising the severe construction difficulties and limitations implied by this approach.

In addition to the above, it has been shown that practical proposals for backfilling and sealing are poorly developed, although the Swedish proposals have clearly placed a greater emphasis on these aspects than the remainder. Generally, the proposals do not provide any assurance that a consistently high-integrity backfill will be provided and, especially for intermediate-level waste emplacement chambers, it seems that the awkward void spaces between units are unlikely to receive satisfactory treatment. Furthermore, none of
the proposals has incorporated geochemical barriers in the form of chemical buffering or retention additives as an integral part of the near-field system; see Chapters 8 and 9.

In many cases, operational aspects are also seen to pose problems in terms of conflict between construction, waste emplacement and backfilling operations; and it is by no means clear how proposed methods will incorporate adequate levels of quality control and radiological safety. Many proposals stipulate a short-term retrieval option, requiring design modifications which may actually reduce effective containment of the wastes. Provisions have also been made for sophisticated monitoring procedures requiring embedded instruments which, in the author's view, could impair overall performance without the benefit of any meaningful results in relation to the assessment of long-term repository performance.

Based on the above, the author considers that there are significant shortcomings in the general approach to repository design, as represented by the current international proposals. Historical perspectives must, of course, be recognised, since conceptual repository design studies were initiated at a relatively early stage in the international research programmes, and have been intended to demonstrate the broad feasibility of underground disposal rather than to provide fully optimised solutions to the problem. Research into the geochemical aspects of near-field containment, in particular, has made significant advances which have not yet been properly translated into coherent design principles.

Nevertheless, it is apparent that the multi-disciplinary aspects of waste disposal research have not yet been
fully co-ordinated into effective repository design strategies. Three broad areas may be identified, namely:

- waste management and conditioning
- geoscientific research
- repository design

This author considers that the compartmentalisation of these aspects has led to an inadequate treatment of the subject as a whole and, in particular, that the importance of repository design aspects in terms of waste containment have not been fully recognised. A change of emphasis is therefore advocated, involving a more integrated engineering approach than hitherto.

It is considered that the repository design process should seek to reduce uncertainties associated with rock mass containment properties, by maximising containment in the near-field. Since near-field parameters are subject to a significant degree of engineering control, whereas host rock parameters are not, opportunities should be sought to introduce near-field barriers which complement the favourable properties of the host environment and mitigate against their more unfavourable or uncertain qualities.

The following specific objectives are identified:

- optimisation of waste parameters in terms of the containment properties of the repository system
- minimisation of the degree of disturbance to the host rock associated with the construction process
minimisation of repository size and maximisation of volumetric efficiency as a means of enhancing near-field containment, reducing costs, and increasing confidence in the reliability of operational procedures

prevention of peripheral disturbance in brittle rocks, and elimination of other potential longitudinal migration paths

introduction of engineered barriers, in the form of specialised backfills, to increase near-field containment to a level comparable with that of the host rock and to complement host rock retention properties

development of operational procedures which produce consistent results, without conflict of underground activities, and which allow high-integrity engineered barriers to be created to laboratory standards.

Various elements of this overall repository design philosophy are developed in subsequent chapters. The principle theme is 'design for containment' and the treatment of relevant aspects is intended to show how the repository design process should be integrated with pre-disposal strategies in order to develop a fully effective disposal system, in which uncertainties are reduced to a minimum.
15. OPTIMISATION PROCESS FOR THE OUTLINE DESIGN OF A HIGH-LEVEL WASTE REPOSITORY IN JOINTED ROCK

15.1 Introduction

This chapter describes a repository design optimisation process which seeks to maximise the level of confidence in the containment properties of jointed host rocks. It is thought that the procedure can and should be beneficially applied in the outline design of high-level waste repositories in crystalline and hard argillaceous rocks, where the influence of rock mass discontinuities creates significant uncertainties concerning the containment properties of the geological system.

Part 2 has provided a comprehensive background to the problems associated with rock mass discontinuities in relation to underground repository systems. Yet, as shown in Part 3, current repository design proposals have not sought to mitigate against the associated uncertainties.

The procedure described in this chapter seeks to manipulate those parameters which are amenable to engineering control in order to reduce the impact of the potentially adverse effects of discontinuities, whose three-dimensional statistical distribution is essentially random in character.

The rock discontinuity distribution is clearly not controllable. However, as already shown, the form and characteristics of the high-level waste units can be precisely controlled by varying the parameters associated with waste conditioning, packaging and temporary storage. It appears that consideration of the
influence of these aspects on the design features and containment properties of repository systems in jointed host rocks has not been taken into account in the repository design proposals put forward to date.

For a repository host formation, the size of the natural containment barrier is measurable on the Km$^3$ scale and, as shown in Chapter 9.1, it must be recognised that realistic fracture flow models cannot be constructed with sufficient accuracy, due to practical limitations in the available data-base. Assessments of the level of containment afforded by the host rock must therefore use a simplified ‘porous medium equivalent’ approach, combined with ‘what if’ calculations, performed to simulate the superimposed influence of hypothetical, but credible, major discontinuities (190). The resulting radionuclide migration models would then indicate upper and lower bound limits for peak dose rates and radionuclide travel times, together with an indication of probability levels to be attached to the results.

However, it is appropriate to examine what measures the repository designer can adopt to influence the results of these predictions. If repository excavations are envisaged as a series of scan-lines of finite length traversing the rock mass, then the quantity of groundwater which could become contaminated by radionuclide release within the system will depend upon the number of transmissive rock discontinuities intersected. Due to the cubic form of the relationship between volumetric flow rate and fissure width, the influence of the larger and less frequent discontinuities will dominate. On a purely non-site specific basis, this implies that the overall size of the repository, or the mean length and total number of scan-lines, should be minimised.
The review of published conceptual repository design proposals, described in Part 3, suggests that the influence of repository size upon the level of waste containment in jointed rocks has not been taken into account. The adoption of relatively compact cuboidal configurations as a basis for the British and French concepts is thought to be based on the known size limitations of the available host plutons rather than the more generalised considerations outlined above. The lack of recognition of the more fundamental impact of repository size on waste containment appears to be confirmed by the relatively extensive (mine-orientated) networks proposed in America, Sweden and Canada; where much larger crystalline rock masses are available.

Some reasonably straightforward changes in the approach to construction planning could bring about significant improvements. The adoption of concurrent phasing for repository construction and waste emplacement is considered contrary to the minimum size philosophy; for reasons described in Chapter 13. For high-competence host rocks, there is no technical reason preventing the completion of all construction activities and validation trials well in advance of the waste emplacement phase. Duplication of shafts and tunnels for access and ventilation would then be avoided, so reducing the overall size of the repository and the number of excavations required.

Other refinements could bring about further reductions in the number and size of excavations. Examples are:

- the use of 'compact' ventilation systems, including ducted compressed air, possibly with refrigeration;
- the use of plant (both for the construction and waste emplacement phases) which does not place significant demands on the ventilation system; e.g. compressed air or electric-hydraulic systems;

- the use of doors (dampers) for diversion of air supply to manned areas of the facility, in order to reduce the total volumetric air flow required. (See Swedish proposals described in Chapter 13).

In the limit, by adopting measures such as these, it should be possible to design repository systems requiring only two shafts. The first could be a relatively small diameter shaft (2 to 3m) for the transfer of waste units. The second could be of moderate diameter (5 to 8m), serving initially as a construction and ventilation shaft, and subsequently as an access shaft during the operational phase; with separate ducting for ventilation supply.

These measures clearly contrast with the majority of proposals advocated for repositories in crystalline rocks. It is notable, however, that the Netherlands' proposals for a saliferous rock repository stipulate only two shafts in order to minimise the risk of groundwater invasion. It is surprising that other countries have not considered it worthwhile to adopt a similar approach for crystalline and argillaceous host rocks, in considering the potential for radionuclide migration from the repository towards the biosphere.

Clearly, the above measures require some changes in emphasis, involving a more sophisticated approach to construction planning and repository operation than hitherto. However, notwithstanding their importance, the thermal aspects of high-level waste repository
design have an even more significant influence on repository size requirements.

For a given maximum allowable rock temperature increase within the centre of an array of buried high-level waste units, the spacing between adjacent units (and hence the total areal extent of the repository) will depend upon the size and shape of the waste units, their cooling history and the geometry of the emplacement system; see Table 13.

The manipulation of these variables offers an opportunity to determine their influence on repository size, and hence to develop an optimum strategy both for waste conditioning and repository design development.

15.2 Variables

As a basis for the present optimisation study, four differently-shaped cylindrical high-level waste canisters are considered. These are based on the Harvest and AVM types described in Chapter 13.3. The 'reference' Harvest and AVM canisters are nominally 0.5m in overall diameter, including a 10mm thick stainless steel outer casing. However, the Harvest canister is 3.0m long whereas the AVM canister is only 1.0m long. Smaller variants are included in the study, by halving the diameters of both standard canisters to produce 0.25m diameter by 3.0m long canisters (half-Harvest) and 0.25m diameter by 1.0m long canisters (half-AVM).

In addition, three levels of metal overpack protection are considered, namely:

- without overpack
o 100mm thick overpack
o 300mm thick overpack

The application of these three overpack conditions to each of the four basic types of waste canister generates twelve differently-shaped waste unit types, with four different values of waste content.

The volume and distribution of vitrified high-level waste inside the units depends upon the dimensions of the canisters and the details of the vitrification/canning procedures. However, the upper portion (approximately one-third of the internal height dimension) normally remains unfilled, to allow the top of the canister to be closed by vacuum welding.

Details of the characteristic dimensions and weights of the waste units used as a basis for the study are provided in Table 19. These are based on information made available to the author by Associated Nuclear Services Limited, during the course of research studies undertaken for the Department of the Environment (7, 9).

In addition to the variations in waste unit size and overpack thickness, four levels of variation are considered for the depths of emplacement boreholes relative to tunnel floor level, namely:

o 6 to 8m depth
o 50m depth
o 100m depth
o 300m depth

This range of borehole depths reflects the variations in international proposals as described in Chapter 13. For each depth, it is assumed that at least 3m is provided
in the upper portion of each hole for backfilling purposes. Thus, the 6m hole depth provides for a planar array of full-sized Harvest canisters without overpacks (one unit per hole). It is also assumed that a stack of three AVM canisters without overpacks may be considered as a planar array; requiring the same hole depth as a single un-clad Harvest canister.

However, as the thickness of overpack increases, so the hole depth required to accommodate three AVM units together with the 3m empty upper section increases. In the limit, a hole depth of 7.95m is required to contain three AVM units with 300mm thick overpacks. The interval 6m to 8m is therefore chosen to represent a 'planar' array of either one Harvest or three AVM units, with or without overpacks.

The larger depths provide cuboidal arrays, requiring multiple stacking, and it is assumed that waste units are placed above one another, in direct contact, in order to provide maximum hole utilisation. The relationship between hole depth and the number of waste units per hole for different overpack thicknesses is shown in figure 65. It is apparent that the number of waste units which can be accommodated depends upon the canister/overpack combination; and the provision of overpacks on the smaller AVM units brings about a greater reduction in hole capacity than for Harvest units.

In order to examine the influence of cooling history, five periods of interim storage are considered for each of the twelve waste unit types identified in Table 19. These are:

- minimum period of storage required to provide a
specified maximum allowable design value of waste unit/rock surface heat flux at the time of disposal (see 15.3 below)

- 30 years interim storage
- 60 years interim storage
- 90 years interim storage
- 120 years interim storage

For each of these interim storage periods, the heat output at the time of disposal is evaluated by reference to figure 10. The pro-rata adjustment of heat output, based on the known weight of vitrified waste in each of the waste units under consideration, produces four separate characteristic heat decay curves as shown in figure 66.

It should be noted that the waste unit characteristics indicated on table 19 and figure 66 are based on an assumed 15% by weight of fission products within a borosilicate glass/waste matrix. This relatively high value is chosen because reduction of waste concentration increases the total number of waste units required. As demonstrated by the Canadian proposals (in which a 1% fission product concentration is adopted), a large number of waste units increases the rate of waste emplacement over a given production period, necessitating a comparatively high level of activity for repository workers and associated plant and equipment during the operational period, and thus increasing the number of delivery routes and the total ventilation capacity required. In addition, a relatively large number of emplacement holes are required. For these
reasons, a single high value of waste concentration is used as a basis for the optimisation study, consistent with the objective of minimising overall repository size.

To summarise the range of variables, twelve different waste units (canister/overpack combinations) are considered in conjunction with four different emplacement hole depths and five different interim storage periods. In combination, this produces up to 240 different sets of repository design parameters.

By establishing a simple non-site specific repository design model, with constant rock properties and a fixed total quantity of high-level waste for disposal, it is proposed to examine the influence of the variables described, and the scope for optimising the overall design strategy by reference to the minimum size philosophy outlined in 15.1 above.

15.3 Thermal Design Theory

Chapter 9.3 has provided a simplified account of the problem of heat transfer associated with high-level waste emplacement, based on radial heat conduction. The basic law of heat conduction states that the linear flow of heat through an elemental volume of material, in thermal equilibrium with a constant heat source, is directly proportional to the thermal gradient. Thus:

\[ fx = - \Gamma \frac{\delta T}{\delta x} \] .......................... (1)

where \( fx \) is the power output per unit area \( W/m^2 \)
\( \Gamma \) is the thermal conductivity \( W/m^\circ C \)
T is the temperature (°C)
X is the distance (m)

However, analogous formulae hold for the y and z directions and thus equation 1 should be written more generally as:

\[ f(x, y, z) = - \int \nabla^2 T(x, y, z) \] .......................... (2)

where \( \nabla^2 = \frac{\partial^2}{\partial x^2} + \frac{\partial^2}{\partial y^2} + \frac{\partial^2}{\partial z^2} \) .......................... (3)

(i.e. \( \nabla^2 \) is the Laplacian operator)

The standard text 'Conduction of Heat in Solids' by Carslaw and Jaeger (37) provides numerous solutions to equation 2 for a variety of different physical conditions and geometries.

For an isolated cylindrical waste unit, conducting heat uniformly from its surface, temperatures at different horizontal distances will be readily symmetric, and may be predicted by considering the waste unit as a finite line source. On this basis, if \( Q(t) \) is the power output of the block as a function of time, it may be shown that (102):  

\[ T(r, z, t) = \frac{1}{8\pi L} \int_0^t dt \frac{Q(t')}{(t-t')} \exp \left( -\frac{r^2}{4\gamma(t-t')} \right) \left[ \text{erf} \left( \frac{L/2 + z}{\sqrt{4\gamma(t-t')}^\frac{k}{2}} \right) + \text{erf} \left( \frac{L/2 - z}{\sqrt{4\gamma(t-t')}^\frac{k}{2}} \right) \right] \] .......................... (4)

where \( r, z \) are polar co-ordinates.

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t is time (sec)
\( \gamma \) is the thermal diffusivity \( \frac{\Gamma}{\rho C} \) (m\(^2\)/sec)
\( \rho \) is the rock unit weight (Kg/m\(^3\))
C is the specific heat (J/Kg °c)
g is the gravitational acceleration (m/sec\(^2\))
\( \text{erf} \) is the error function

The \( Q(t') \) function in equation 4 must incorporate appropriate radioactive decay constants to enable the time-dependence of the waste heat output to be taken into account; and the inclusion of the thermal diffusivity term, \( \gamma \), takes into account the time required for the conduction of heat from the source to the point under consideration.

Since the basic heat conduction equation (equation 2) is linear, the principle of superposition can be applied to the derived temperature expression given in equation 4; provided the rock properties \( \Gamma, \rho, \) and \( C \) are assumed to remain constant with varying temperature. This simplifying expedient enables equation 4 to be used in evaluating the temperature increase at any point within the host rock, due to a geometrical array of several waste units.

For repository design purposes, the position of greatest interest is the rock surface adjacent to the most central waste unit within the array; since this region should experience the greatest induced temperature increase. The significance of the thermal diffusivity term, \( \gamma \), becomes apparent when it is realised that \( \frac{r^2}{4\gamma} \) represents the characteristic time for heat to be conducted over a distance \( r \) (102).

Characteristic heat conduction times are plotted on
figure 67, for radial distances of up to 30m; based on a typical value of thermal diffusivity for granite of \(10^{-6} \text{ m}^2/\text{sec}\). As shown, heat from a single line source would be conducted about 35.5m in 10 years, 61.5m in 30 years and 300m in 700 years. Hence, for the comparatively large distances, conduction times are significant in comparison with the heat-producing life of high-level wastes. As shown in figures 10 and 11(a), the overall half-life of high-level reprocessing wastes is approximately 30 years for the first hundred years or so, due to the influence of the fission products \(^{90}\text{Sr}\) and \(^{137}\text{Cs}\), and increases thereafter as the longer-lived radionuclides such as \(^{241}\text{Am}\) (half-life 430 years) begin to dominate; the overall effective heat-producing life of the wastes being about 1000 years.

It is therefore apparent that for a uniform spacing of waste units, the maximum rock temperature rise at the centre of the array will only be influenced to a significant extent by other waste units within a radius of about 100m to 300m. The addition of waste units beyond this distance will have relatively little influence upon the maximum induced rock temperature.

The integration of equation 4 over time and space, and the superposition of results to predict the maximum rock temperature rise, for a given array of waste units of known size and heat output characteristics, is not amenable to hand-calculation and requires computer evaluation. However, a more useful computer calculation would be to fix the maximum allowable rock temperature rise and to solve equation 4 iteratively, in order to determine the corresponding lateral waste unit spacings.

Such an approach requires a significant amount of computer time, and appears to be the method adopted for
the determination of waste unit spacings in the majority of the conceptual repository design solutions put forward; based on pre-determined waste unit characteristics and emplacement hole depths. However, the use of this rigorous technique, to evaluate waste unit spacings for the full range of variables described in 15.2 above, would be impracticable in terms of computer time. Therefore, a simplified approach is sought which can provide a suitable basis for 'scoping' calculations.

The results of computer-based studies undertaken by the Atomic Energy Research Establishment are thought to provide a satisfactory basis for deriving waste unit spacings, to within a first order accuracy suitable for the present study. Figure 68 is adapted from a report by Bourke and Hodgkinson (28), and shows the variations in maximum rock temperatures with time for a point in the centre of a granitic repository containing approximately 7000 Harvest canisters, each with an initial heat output of 1kW. Three curves are shown, each of which provides a 20m separation between adjacent canisters.

**Curve a** is for a planar array (one canister per hole; 1 x 85 x 85 canisters)

**Curve b** is for a cuboidal array (7 canisters per hole; 7 x 33 x 33 canisters)

**Curve c** is for a cubic arrangement (19 canisters per hole; 19 x 19 x 19 canisters)

It should be noted that the cubic arrangement is a specific example of a cuboidal configuration and, in general, the ensuing discussion will refer to the
generic cuboidal form only.

Significantly, all three configurations result in similar maximum induced temperature increases (approximately 77°C). However, the durations of the temperature increases vary considerably. The curves show that the peak temperature is sustained for one or two years for the planar configuration (curve a) and approximately fifty years for the cubic configuration (curve c). Taking an arbitrary temperature rise of 60°C, the curves show that the periods over which this temperature is exceeded are approximately thirteen years and three hundred years, respectively.

Two significant factors emerge from these results. Firstly, due to the relatively long duration of the induced temperature increase, convective groundwater movements are more likely to develop with cuboidal (or cubic) type configurations than for planar configurations. Secondly, because chemical reaction rates are increased by sustained high temperature, rates of waste canister corrosion and waste leaching may be expected to be higher for cuboidal (or cubic) configurations.

Clearly, a careful assessment of the implications of these observations is necessary to ensure that an appropriate level of waste unit protection is achieved. This could indicate that corrosion protection by means of waste unit overpacks is more desirable for cuboidal configurations, although (as will be shown subsequently) overpacks may also be desirable for other reasons.

Figure 69 illustrates a further design aspect, based on Bourke and Hodgkinson's work (28). In figure 69, waste unit heat outputs and geometrical configurations are
identical to those adopted in figure 68, but the lateral spacings are reduced to 15m. This spacing reduction results in markedly different temperature-time profiles.

For the planar array, the peak temperature rise is increased only marginally, from 77°C to about 80°C; but the peak value for the cubic array almost doubles to around 150°C, whilst the cuboidal array produces an intermediate value. Furthermore, although the periods over which peak temperatures are maintained are shorter for the cubic and cuboidal arrangements, the period over which the temperature rise exceeds the arbitrarily chosen value of 60°C is increased to about 300 and 500 years, respectively.

The results of these studies clearly demonstrate the sensitivity of cuboidal configurations to variations in waste unit spacings and, in conjunction with the likely variations in thermal properties of real rock masses, point to the need to adopt a relatively conservative approach to thermal design aspects for geometrical arrangements of this type.

On the basis of their computer-based studies, Bourke and Hodgkinson concluded that a 20m x 20m x 20m spacing for extensive cuboidal or cubic arrays of Harvest canisters would limit the rock temperature rise to about 80°C. Thus, the equivalent volume of rock per canister for satisfactory heat dissipation was estimated as 8000m³. For planar configurations, they concluded that a 15m x 15m spacing; equivalent to a rock area of 225m² per canister, would also satisfy the 80°C maximum temperature rise constraint.

Clearly, the above results are applicable only to Harvest canisters. For the purpose of the present optimisation
study, it is proposed to apply these results to other waste unit shapes and heat outputs by making a simple pro rata adjustment; relating the heat output per unit of curved surface area of the waste unit under consideration to the rock volume required for satisfactory heat dissipation.

It is also assumed that the 8000m$^3$ rock volume requirement for 1 kW Harvest canisters may be applied to cuboidal configurations without any vertical separation between the units. Hence, the equivalent rock area required per stack becomes $8000 \div 3 = 2700m^3$; equivalent to a mean lateral hole spacing of $\sqrt{2700} = 52m$. The validity of these assumptions, as a good first order approximation, was confirmed during a visit by the author to the Atomic Energy Research Establishment during the course of studies undertaken on behalf of the Department of the Environment (9).

From equation 1, it is apparent that, under steady-state conditions, the temperature rise at any point in a conducting medium is directly proportional to the heat flux (power per unit area) of the source. The heat flux on the surface of a Harvest canister with a heat output of 1 kW is approximately 212 W/m$^2$ (ignoring end-effects). Hence, a waste unit of different size, shape (possibly with an overpack) and heat output, will have a different surface heat flux, $f$, equal to its total heat output divided by its curved surface area; and for close-stacked vertical columns, the value of $f$ will be a constant over the column length.

Hence, the rock area required for satisfactory heat dissipation, such that the maximum induced rock temperature rise does not exceed 80°C, may be estimated for waste units having lower values of surface heat...
flux, \( f \), as follows:

for planar arrays

\[
\text{rock area required} = \frac{f \times 225}{212} \quad (5)
\]

for cuboidal arrays

\[
\text{rock area required} = \frac{f \times 2700}{212} \quad (6)
\]

Equations 5 and 6 form the basis for the assessment of emplacement hole and tunnel spacings in the repository design optimisation model described in 15.4 below.

15.4 Repository Design Optimisation Model

It is proposed to apply the simplified thermal design criteria outlined in 15.3 above, in order to examine the influence of the variables described in 15.2 on overall repository size. This requires the adoption of a reference high-level waste disposal capacity and some simple and consistent repository design rules.

The reference disposal capacity selected is 4000 tonnes of vitrified high-level reprocessing wastes, containing 15% by weight of fission products, as noted in 15.2. This corresponds approximately to the predicted U.K. high-level waste arisings up to the year 2010 (124). The reasons why variations in fission product concentration are not considered have been given
in 15.2 above. However, it may be noted that although reduction in waste concentration would require a larger number of emplacement holes (and hence scan-lines), the overall repository size would theoretically remain constant, due to corresponding reductions in waste unit heat output, and hence emplacement hole spacings.

The geometry of the proposed repository design optimisation model is shown in figure 70; in which the lines (scan-lines) represent the centre-lines of tunnels and waste emplacement holes. The repository model is assumed to be square in plan, in order to simplify calculations, and so that variations in repository size (in the areal sense) can be represented by a single scan-line dimension; namely the length of one side.

Based on the proposed construction planning philosophy outlined in 15.1 above, the repository model comprises a series of parallel emplacement tunnels, linked by a single peripheral access/ventilation tunnel. For simplicity, shafts are not shown, although it may be envisaged that there would be two shafts sited at some substantial lateral distance from the emplacement zone (outside the possible influence of thermal convective groundwater movement generated by the wastes).

All of the repository tunnels are assumed to have a width of 5m and a nominal cross-sectional area of $25m^2$. Waste emplacement boreholes are assumed to have a diameter 200mm greater than the diameter of the waste unit under consideration, so as to provide a 100mm wide annular space for placement of backfill.

For cuboidal emplacement configurations, the centre to centre lateral spacings between adjacent boreholes and adjacent emplacement tunnels are calculated as the
square root of the rock areas given by equation 6 in 15.3 above; so that borehole and tunnel spacings are generally equal. However, for stability reasons, a minimum tunnel centre to centre spacing of 20m is specified in order to provide a minimum 'pillar' thickness of 15m (equivalent to 3 tunnel widths).

Thus, where the square root of the rock area predicted by equation 6 is less than 20, the area must be divided by 20 to give the appropriate borehole spacing. However, for planar configurations, borehole spacings are always less than the tunnel spacing, and are simply obtained by dividing the area given by equation 5, by 20.

The remaining rules associated with the repository design model are concerned with 'end effects' at tunnel intersections, and may be described by reference to figure 70. They are as follows:

- the length of any straight length of tunnel is considered to be the distance between the centre-lines of the transverse tunnels at each end

- a 20m centre to centre spacing is specified between the outermost emplacement tunnels and the adjacent access tunnels, irrespective of specific thermal design features (this again provides a pillar width equivalent to 3 tunnel widths).

- the emplacement boreholes at the extreme ends of the emplacement tunnels are located 12.5m from the centre-line of the peripheral access tunnels, as shown in section A-A; i.e. 10m from the tunnel intersections.
The above 'rules' enable variations in the geometry of the model to be established in terms of the number of emplacement holes, and the borehole and tunnel spacings. The basis of the computations is outlined below:

Let $x =$ number of emplacement holes required
$y =$ number of emplacement tunnels required
$s =$ emplacement hole spacing
$t =$ emplacement tunnel spacing

then, referring to figure 70,

emplacement tunnel length, $l_1 = \frac{xs + 25}{y}$ ............ (7)

access tunnel length, $l_2 = t(y-1) + 40$ .............. (8)

but, $l_1 = l_2$

hence, $\frac{xs + 25}{y} = t(y-1) + 40$

\[\frac{xs}{t} = ty(y-1) + 15y\]

\[ty^2 - y(t-15) - xs = 0\]

\[y = \frac{t-15 \pm \sqrt{(t-15)^2 + 4xst}}{2t} \] ............ (9)

and substituting for $y$ in equation 8 yields the characteristic overall repository dimension.

It is recognised that equation 9 will only give a whole number value for the number of emplacement tunnels, for certain values of $x$. This is due to the assumption of a
square-shaped repository model.

However, in order to bring about an appropriate adjustment, it would be necessary to vary the waste disposal capacity. Since the purpose of the present study is to examine the influence of variables for a fixed quantity of waste, adjustments of this type are considered inappropriate, and it is apparent that the necessary minor mathematical abstraction does not invalidate the results obtained.

15.5 Results

The results of computations based on the repository design model, are presented in tables 20 to 24, inclusive. Each table contains results corresponding to a particular interim waste storage period, for each of the twelve waste unit types and four emplacement borehole depths under consideration.

For each set of waste quantities and heat output characteristics, the tables indicate the required minimum emplacement hole and tunnel spacings, based on the application of the simple thermal design criteria described in 15.3 above. The corresponding repository construction quantities shown in the tables are calculated on the basis of the repository design rules described in 15.4 above.

A set of derived repository size indices is also shown. These comprise:

- The representative areal dimension of the repository; equivalent to the mean length of one
side of its plan area \((l_1 = l_2\) in figure 70)

- **The characteristic rock volume**: equivalent to the plan area of the repository times the depth of the emplacement holes. This quantity therefore represents the total volume of the prismatic-shaped body of rock in which the repository excavations are situated (expressed in \(m^3 \times 10^6\))

- **The total scan-line length**: equivalent to the sum of the length of the tunnels and boreholes distributed within the characteristic rock volume (expressed in km)

These three indices are used as a basis for describing variations in repository size and scan-line length, due to changes in design parameters (waste unit type, storage period and emplacement hole depth).

The maximum number of result sets which can be presented on each of the five tables is 48; suggesting a possible total maximum of 240, as described in 15.2 above. However, the thermal design criteria include a maximum allowable surface heat flux of 212 W/m\(^2\), and it is found in practice that some of the waste units exceed this value for certain of the specified storage periods.

As shown, waste unit number 3 can be buried without any interim storage, and its surface heat flux at the time of vitrification is actually less than the 212 W/m\(^2\) threshold design value. The remaining units require minimum storage periods ranging from 0.5 years to 65 years prior to disposal. Waste units 2, 3, 5, 6, 9 and 12 require minimum storage periods of less than 30 years. For waste units 4 and 10, the minimum period coincides (fortuitously) with the 30-year and 60-year
storage periods, represented in tables 21 and 22 respectively. Units 1, 8 and 11 require interim storage periods of 33 years, 26 years and 40 years respectively; so that for these units (together with waste unit number 10) it is not possible to derive results for the 30-year storage period. Waste unit number 7 requires a minimum storage period of 65 years, and thus both the 30-year and 60-year storage periods must be excluded.

Based on the above, 8 out of the possible 240 result sets cannot be accommodated within the repository design rules, and hence the number of data sets presented in the tables is 232.

The trends which emerge from the study have been represented graphically by plotting storage period against the various repository size indices for different waste unit types and emplacement hole depths, based on the data provided in tables 20 to 24.

Figures 71 to 74 indicate the influence of storage period, canister type, overpack thickness and hole depth on the characteristic areal dimension of the repository. Comparing figure 71 with figure 73 and figure 72 with figure 74 shows that for a given waste storage period, halving the diameter of the waste canister results in a significant increase in characteristic areal repository dimension. This increase is typically of the order of 25% to 35%; the higher values applying to relatively shallow boreholes, shorter storage periods and waste units without overpacks. The half-size diameter waste canisters also create a relatively wide range of repository areas; from less than 250m square to nearly 2km square, whereas the full-size canisters create a narrower range in which the maximum area is always less than 1km square.
Comparing figure 71 with figure 72, and figure 73 with figure 74, shows that the differences in repository area associated with differences in waste unit length are relatively small. As would be expected, due to the effects of stacking waste units with overpacks (see figure 65), a somewhat larger area is required to accommodate the short-length units (based on the AVM canister) due to an increase in the proportion of the hole length occupied by overpacks. The difference is greatest for large overpack thicknesses and relatively shallow emplacement holes, but is generally insignificant in comparison with the effects of changes in the diameter of the canisters, as described above.

For all waste unit types, increasing the period of interim storage is seen to have a significant effect on the repository area. In absolute terms, the effect is greatest for planar arrays and small diameter canisters; although reductions in characteristic areal dimension of over 60% may be brought about by long-term storage for all emplacement configurations and waste unit types.

The repository area reductions brought about by interim storage become proportionately less as the storage period is increased. For planar arrays and cuboidal arrays with relatively modest hole depths, long-term storage (90 years or more) has a significant beneficial effect. However, for large hole depths and overpack thicknesses, storage in excess of 60 years produces relatively small area reductions.

The provision of overpacks is seen to have a significant influence on repository area, and it is interesting to note that the amount of area reduction brought about by increasing overpack thickness from 0 to 100mm is
generally about the same as that brought about by an increase from 100mm to 300mm. The effect of providing an overpack is seen to be similar to the effects of interim storage. In both cases, a reduction in surface heat flux is brought about, causing a reduction in rock area requirements. However, the amount of overall rock area reduction (in absolute terms) due to provision of overpacks is reduced with increasing storage period and emplacement hole depth.

It is apparent from figures 71 to 74 that area requirements are generally greatest for planar arrays, and are reduced with increasing borehole depth for any given interim storage period and waste unit type. The range of repository areas is greatest for waste units without overpacks; and for waste units with overpacks, the reduction in area brought about by increasing the hole depth becomes relatively small. It is interesting to note that a planar array of units with 100m thick overpacks generally occupies a smaller area than a cuboidal array of 50m deep boreholes containing waste units with no overpacks.

Figures 75 to 82, inclusive, indicate the influence of storage period, canister type, overpack thickness and hole depth on the characteristic rock volume and scan-line length of repository systems. Results are found to be markedly different for planar arrays and cuboidal arrays, and for this reason separate suites of curves have been plotted. Figures 75 to 78 present the results for planar arrays, and figures 79 to 82 present the results for cuboidal arrays.

The dotted contour lines, which traverse the solid line curves, represent values of scan-line length per unit of characteristic rock volume occupied by the repository;
expressed in \( \text{km/m}^3 \times 10^6 \). The contours have been obtained by plotting values onto the solid-line curves, using the data presented in tables 20 to 24, and interpolating linearly along the curve lengths to give convenient contour values. Thus, each curve indicates (for a given storage period, canister type and overpack thickness) the corresponding rock volume and scan-line intensity within the prismatic rock mass occupied by the repository system.

Considering first the planar arrays, illustrated in figures 75 to 78, it is evident that the characteristic rock volume varies by more than an order of magnitude, for the range of input variables considered; from approximately 1 million to 20 million cubic metres.

Increasing the period of interim storage (particularly in the range 0 to 90 years) is seen to have considerable effect in reducing the characteristic rock volume. Provision of an overpack also reduces the characteristic rock volume considerably. The greatest reduction (typically 50%) is brought about by the provision of a 100mm thick overpack; the further increase of 300mm thickness having a smaller proportionate influence.

However, the reductions in characteristic rock volume with increasing storage period and overpack thickness are also accompanied by significant increases in scan-line intensity, particularly for waste units with thick overpacks. Thus, for long storage periods (in excess of 90 years), and correspondingly small rock volumes, the provision of a 300mm thick overpack may result in a two to four-fold increase in scan-line intensity, accompanied by a 60% reduction in rock volume.
Significantly, for a given canister type, the scan-line intensity is seen to be more or less constant for a given characteristic rock volume; except for storage periods less than 30 years. In addition, the form of the curves is broadly similar for waste units based on canisters of the same diameter.

Comparing figure 75 with 76 indicates little difference in characteristic rock volumes or scan-line intensities associated with variations in waste unit length. The same applies in comparing figure 77 with figure 78. Hence, it is apparent that increase in canister length has no significant effect, irrespective of the waste unit diameter. However, comparing figure 75 with figure 77 and figure 76 with figure 78 shows that halving the canister diameter increases the characteristic rock volume significantly (for a given storage period).

The volume is increased by a factor of 2 for canisters without overpacks, by about 60% for canisters with 100mm thick overpacks and about 30% for canisters with 300mm thick overpacks.

However, for a given storage period, the increase in rock volume brought about by reduction in canister diameter is also accompanied by a significant increase in scan-line intensity for all overpack thicknesses. Thus, it is clear that in jointed rocks, the selection of the smaller diameter 'half-size' canisters creates a higher probability that the repository excavations will intersect transmissive rock mass discontinuities.

Variations in characteristic rock mass volume and scan-line intensities for cuboidal arrays are shown in figures 79 to 82. For a given waste unit type, it is found that variations in hole depth have little
influence on rock volume requirements. Hence, results for the 50m, 100m and 300m deep emplacement holes all fall within the bands shown.

This is because the thermal design model, based on the extrapolation of Bourke and Hodgkinson's computer studies, effectively treats the emplaced waste columns in cuboidal arrays as infinitely long line sources. Since the hole lengths selected all have suitably high length to diameter ratios, the results shown here are considered reasonably accurate. It should be recognised that, in practice, there would be a gradation in characteristic rock volume requirements with increasing hole depth, between the values shown for planar arrays and those for the 50m hole depth configurations. However, the nature of this transition (for hole depths in the range 8m - 50m) cannot be explored due to the limitations of the present thermal design model.

In general, the trends which have been described for planar arrays are seen to apply also to cuboidal arrays. However, the values of characteristic rock volume and scan-line intensity are very different. For a given waste unit type and storage period, the characteristic rock volume is typically 6 to 7 times greater for the cuboidal arrays compared with planar arrays, whilst the scan-line intensity is typically an order of magnitude less.

Based on the foregoing, it is apparent that an opportunity exists to obtain optimum values of characteristic rock volume and scan-line intensity to suit the anticipated characteristics of the rock mass, and to reduce the probability of encountering transmissive features. The ability to examine the influence of hole depth for intermediate values, in the
range 8 - 50m, would clearly form an important aspect of such more detailed studies. Nevertheless, the results and trends indicated by the present method of analysis, should enable the range of variables to be reduced considerably.

Further insight into the effects of the design variables considered may be obtained by reference to the volumetric efficiency of the repository systems generated by the design model; where volumetric efficiency is defined as the ratio between the total volume of emplaced waste and the total volume of the repository excavations. Values of the latter are shown in tables 20 to 24, whilst volumes of waste for each waste unit type may be obtained from table 19.

As a background, it is instructive to refer to the values of volumetric efficiency shown in table 18, based on the international high-level waste repository concepts reviewed in Chapter 13. The Danish matrix of drillholes concept achieves the greatest volumetric efficiency value of 22%, which is attributable to the absence of underground shafts and tunnels. Volumetric efficiencies for concepts which are based on the in-floor system vary considerably from 0.095% for the French concept to 2.3% for the Canadian concept, with apparently no influence due to rock type.

Based on these observations, it would appear that no attempt has been made to maximise volumetric efficiency. However, in considering the values shown in Table 18, it must be noted that they represent gross volumetric efficiencies; i.e. the volume of waste considered includes the volume of the overall waste unit (the Swedish concept includes provision for a 100m thick overpack). Furthermore, the fission product
concentrations vary (see table 12). Hence, the 1% fission product concentration adopted by Canada results in a relatively high apparent volumetric efficiency; but for comparison with the results of the present study (based on a 15% fission product concentration) the value should be reduced by at least an order of magnitude.

The range of volumetric efficiencies generated by the present repository design model are shown in figures 83 to 90 inclusive. Gross volumetric efficiencies based on the volume defined by the overall dimensions of the waste units (canister and overpack) are indicated in figures 83 to 86. Net volumetric efficiencies, which include only the volume of the borosilicate glass/waste matrix, are shown in figures 87 to 90.

The total excavation volumes indicated in tables 20 to 24 are based on the tunnel and borehole excavations which make up the repository design model configuration shown in figure 70. Therefore, a fixed additional excavation volume has been added to allow for access shafts and distribution tunnels. For this purpose, it has been assumed that the repository is located at 500m depth, and is connected to the surface by two shafts; comprising a 2.5m diameter waste delivery shaft and an 8.0m diameter access and ventilation shaft. In addition, allowance has been made for twin 200m long 25m$^2$ cross-section tunnels to provide access and distribution routes between the shaft bottom and the waste emplacement zone. These additional excavations are thought to be reasonably consistent with the minimum size philosophy previously described, and result in the addition of a further 37,600m$^3$ to the total excavation volumes shown in tables 20 to 24.

Variations in gross volumetric efficiency are shown in
figures 83 to 86 inclusive. The striking feature of the results shown is that the values of volumetric efficiency are significantly greater than those obtained in the international repository design proposals, as shown in Table 18; the actual difference depending upon the combination of variables. The latter is thought to demonstrate the usefulness of optimisation studies of the type described here, and to underline the relevance of the minimum repository size philosophy which has been adopted.

Comparing figure 83 with figure 84 and figure 85 with figure 86 shows that an increase in canister length brings about a modest, but significant, increase in gross volumetric efficiency for waste units with thick (300mm) overpacks; especially for deep hole configurations. This is due to the relatively large contribution to the gross waste volume which is made by overpacks when used on the shorter (AVM-type) canisters.

Comparing figure 83 with figure 85 and figure 84 with figure 86, shows that the half-diameter variants of the standard Harvest and AVM canisters result in greater gross volumetric efficiencies than for the full diameter standard canisters. This is again most pronounced for waste units with thick overpacks, and for deep emplacement hole configurations; and is attributable to the significant reduction in surface heat flux brought about by reducing the volume of waste per unit, whilst maintaining a large outer surface area due to the provision of an overpack. The overall effect is to bring about a relatively close emplacement hole spacing, so reducing the overall length of tunnelling required.

Significantly, several of the curves shown in figures 83 to 86 develop a sinusoidal shape, which indicates that
prolonged storage has a more beneficial impact on gross volumetric efficiency for some waste unit/hole depth combinations than for others. It is also apparent that for certain storage period/waste unit combinations, relatively modest emplacement hole depths can produce higher gross volumetric efficiencies than for deep emplacement hole configurations.

For modest emplacement hole depths, in the range 0-50m, and with 100mm thick overpacks, gross volumetric efficiencies of 2-2.5% are readily attainable for all waste unit types, provided that interim storage in excess of 90 years is provided. In all cases, gross volumetric efficiencies of 10% or more are obtainable only by adopting relatively deep emplacement hole depths, providing thick (300mm) overpacks, and/or storing the waste units for periods in excess of 90 years.

Variations in net volumetric efficiency are shown in figures 87 to 90 inclusive, in which only the total volume of borosilicate glass has been included in the evaluation of waste volumes; i.e. the volume contribution made by overpacks and the evacuated space within the canisters has been ignored. The waste volume is therefore constant, irrespective of the waste unit type (1536m$^3$ is the total volume of glass; see Table 19). Hence, variations in net volumetric efficiencies are a direct indication of the variations in total excavated volume, as shown in Tables 20 to 24.

Comparing figure 87 with figure 88, and figure 89 with figure 90, shows that an increase in canister length brings about a modest increase in net volumetric efficiencies for waste units with overpacks. This is the reverse of the trend noted for gross volumetric
efficiencies, and is due to the decrease in the efficiency of hole filling due to the provision of overpacks on AVM-type canisters compared with Harvest canisters; see figure 65. However, the effect is small, and is considered negligible in the case of the full-size Harvest and AVM-type units; see figures 89 and 90).

Comparing figure 87 with figure 89, and figure 88 with figure 90 shows that the full-size Harvest and AVM-type waste units produce significantly greater net volumetric efficiencies than is the case for the half-diameter variants. This is attributable to the relatively large numbers of waste units required, where the half-diameter versions are used, necessitating a proportionately large number of emplacement holes to accommodate the same net volume of waste material. Furthermore, where thick overpacks are used, these occupy a greater proportion of the emplacement hole volume than is the case for the full-size Harvest and AVM waste units.

For several of the curves, a sinusoidal shape is developed (as noted for variations in gross volumetric efficiency), and it is apparent that an optimum storage period may be selected to suit each particular waste unit/emplacement hole combination in order to maximise the net volumetric efficiency. The curves suggest that, in general, prolonged storage in excess of 90 years has a beneficial effect in terms of net volumetric efficiencies. This is most apparent for the full-size Harvest and AVM-type units; see figures 89 and 90.

The most significant aspect which emerges from the examination of figures 89 to 90, is that the provision of overpacks can increase the net volumetric efficiency of repository systems. This is attributable to their
beneficial influence in reducing the surface heat flux of the waste units, and hence the emplacement hole spacings. However, detailed examination of the curves shows that the thickest overpack (300mm) does not produce the greatest net volumetric efficiency in all cases. For some waste unit/storage period hole depth combinations, a 100mm thick overpack is seen to provide the best result.

15.6 Costs

The author has previously stated that cost comparisons should be introduced only where a number of repository design options are seen to provide comparable levels of waste containment; see Chapter 5. However, in order to provide greater insight into the trends which have emerged from the results described in 15.5 above, a cost analysis has been performed in which a notional cost has been assigned to each of the 232 repository schemes represented in Tables 20 to 24.

The analysis is based on U.K. construction prices at January 1982, and includes for the provision of interim storage and waste unit overpacks, as appropriate. In order to evaluate the cost of constructing and maintaining interim storage facilities, reference has been made to estimates made by Mott, Hay & Anderson for an underground interim storage facility to be located above an underground repository, at a nominal depth of 50m (144). The cost of overpacks has been assumed to be nominally £1,100 per tonne, based on advice given to the author by Associated Nuclear Services Limited, during the course of studies undertaken on behalf of the Department of the Environment (9, 124).
The calculated costs are based on the following 'fixed cost' and 'variable cost' components:

Fixed Costs

- construction of adits and shafts for the interim waste storage facility (144)
- construction of twin access shafts and shaft bottom connection/access tunnels, as described in 15.5 above
- service and ventilation tunnels in the interim waste storage facility, comprising tunnels $16m^2$ in section and totalling 300m in length
- capital costs for mechanical and electrical installations for ventilation and hoisting in the interim storage facility
- a notional provision of £1M for surveillance and monitoring

Variable Costs

- construction costs for the tunnels within the interim storage facility (dependant upon the number and type of waste units)
- construction costs for the repository tunnels (based on data in Tables 20 to 24)
- construction costs for emplacement holes (based on data in Tables 20 to 24)
- provision of overpacks (based on sizes in Table 19)
operational costs associated with sub-surface handling and waste emplacement

power supply and spares

The following additional notes are relevant:

construction costs are based on drill and blast excavation for shafts and tunnels

drilling costs are based on a conservative treatment of manufacturer's estimates of both unit costs and progress rates

no provision has been made for the projection of capital investment over the period of construction and operation of the facilities

no provision has been made for backfilling and sealing costs (see Chapter 17.3)

underground construction costs are based on normal rates of progress, based on the assumption that construction of the repository will be completed in advance of the waste emplacement phase

The derivation of appropriate unit rates to be applied to the above cost components is attributable to Stephens, who undertook a similar cost analysis, in collaboration with the author, during studies undertaken for the Department of the Environment (9). The author has drawn freely from relevant basic data for application to the present analysis; see Acknowledgements.
The results of the cost analysis, applied to each of the repository schemes derived from the optimisation model, are presented in Table 25. The figures shown represent the total cost of disposal, including:

- provision of overpacks
- interim storage (including construction of storage facility)
- construction of the repository
- repository operation and waste emplacement

Items excluded are:

- cost of vitrification
- cost of surface transport of waste units
- cost of repository backfilling

The figures shown in brackets represent the costs exclusive of overpacks, for waste units 2, 3, 5, 6, 8, 9, 11 and 12. The difference between the un-bracketed and bracketed figures therefore corresponds to the cost of overpacks, as entered at the bottom of the table.

It should be noted that costs have not been included for waste units 2, 3 and 5 for the minimum storage periods specified in Table 20 (4 years, zero and 5 years respectively). For these units, interim storage costs cannot be assessed because their relatively high heat output of the units at the start of the interim storage period could not be accommodated by the storage facility assumed for costing purposes (144). It should
be noted that waste unit 3 would, in practice, require the provision of storage facilities since some short-time buffer storage capacity would be required in order to control waste delivery rates.

The economic effects of interim storage are seen to vary according to the waste unit type and the emplacement hole depth. The general trends may be summarised as follows:

**Planar Configuration**

For the half-diameter canisters (types a and b), it is found that increasing the storage period reduces overall costs for units without overpacks and with 100mm thick overpacks, throughout the time-scale considered. However, for waste units with 300mm thick overpacks, the minimum cost occurs at a storage period of about 60 years. For the full-diameter canisters (types c and d) it appears that cost minima occur for all overpack conditions at around 90 years.

**50m Hole Depth**

For the half-diameter canisters (types a and b), the minimum overall cost for waste units without overpacks occurs with 120 years interim storage. Where 100mm overpacks are provided, it appears that an 'optimum' storage period occurs at approximately 90-120 years; and for units with 300mm overpacks, the optimum is reached at 60-90 years. For the full-diameter canisters (types c and d), the minimum overall costs are achieved for relatively short storage periods of up to 60 years, irrespective of the overpack thickness.
100m Hole Depth

For the half-diameter canisters (types a and b) the minimum overall cost for all overpack conditions occurs with relatively short storage periods of up to 60 years. For the full diameter canisters (types c and d), it appears that the shortest possible storage period provides the minimum cost solution.

300m Hole Depth

For all waste unit types it appears that the shortest possible storage period provides the minimum cost solution. The only exception is waste unit 12, where the minimum overall cost occurs with a 30-year storage period.

Overall, it is significant that the full-size Harvest and AVM canisters provide significantly lower-cost solutions than for waste units based on the half-diameter variants. It should also be noted that waste units based on canisters of the same diameter produce a very similar overall costs for any given storage period/overpack thickness/hole depth combination.

It is also apparent that the greatest cost variations occur for relatively short periods of interim storage. With extended periods of storage (greater than 60 years) overall costs tend to converge, and for waste units based on full-size canisters are found to lie within the range £70M to £384M.

For repository systems based on Harvest or AVM-type waste units with 100mm thick overpacks and with
moderate emplacement hole depths (up to 50m), overall costs are found to lie in the range £133M to £196M.

15.7 Other Aspects

Thus far, no consideration has been given to the practical engineering aspects of waste handling and emplacement. Detailed consideration of these aspects will be provided in subsequent chapters. However, in the context of the present optimisation study, it is appropriate to consider the influence of waste unit type upon the broad logistics of sub-surface handling operations, consistent with the minimum repository size philosophy which has been adopted.

Table 26 indicates the waste unit delivery and emplacement rates which would apply for each waste unit type, based on an average delivery rate of 200 tonnes of vitrified reprocessing waste per year, over a 20-year period. A seven-day working week has been assumed, with 50 working weeks per year, and it has also been assumed that peak delivery rates could be up to twice the average delivery rates.

It is evident that waste units based upon canisters (b), (c) and (d) require the minimum of equipment and personnel to facilitate routine emplacement. A single canister transporter and hole backfilling unit would probably suffice to cope with normal and peak periods of activity, and therefore the requirements for ventilation and other supporting services would be relatively small. In contrast, waste units based upon canister (a) are likely to require two operational groups during normal activity, and four during peak periods, with consequent heavier demands upon supporting services. The need may
also arise for passing points, or a traffic control system within the distribution tunnels. A requirement for additional stand-by/emergency equipment or facilities is a separate issue, but would also have to be considered.

Thus, adoption of a waste unit based on canister (a) appears less attractive from an operational viewpoint. However, for the deepest holes (300m), the time taken to fill a single hole is relatively short (about 6 weeks for waste unit 1, compared with 6 months for waste unit 7, based on average delivery/emplacement rates). This may be an added operational consideration for cuboidal repository configurations.

On examination of the different physical characteristics of the waste units shown in Table 14, types 6 and 12 are seen to be relatively large and heavy (17 tonnes and 26 tonnes respectively). This may not be significant in relation to the lowering of units to the repository level, since it would be possible to provide low-speed hoisting equipment of the required capacity, with provision for additional weight associated with extra radiation shielding if required; however, canister size and weight could become significant factors within the repository itself.

Due to the long length of the waste units, the sub-surface transporter vehicles would probably incorporate a transfer flask with tilting facilities, to allow transportation of units in the horizontal mode (in order to lower the centre of gravity of the system); see Chapter 13. For waste units based on canisters (b) and (d) these transporters may exceed 5m in length, and would therefore require provision of relatively large radii at the ends of the emplacement tunnels. This
would tend to increase the overall size of the repository.

During emplacement, the transfer flask would assume a vertical position above the hole. With due allowance for ground clearance, top and bottom shielding, and provision of winching apparatus, the overall headroom required would probably exceed 5.5m, and would therefore require a larger emplacement tunnel size than has been assumed, or local enlargements at the emplacement hole positions. In addition, the winching mechanism and fail-safe devices could be cumbersome, in view of the heavy loads involved.

In contrast, the smaller units based on canister types (a) and (c) would be relatively easy to manage. The lower weight and reduced height would not pose any special problems in relation to waste transfer or emplacement. These units could probably be transported in the vertical mode, without the need for tilting facilities.

Thus from an overall operational viewpoint, it appears that waste units based upon canister (c) are the most attractive, whilst units 1, 2, 3, 6 and 12 pose various problems in relation to mechanical design and logistical aspects, and would require an increase in the length and volume of tunnel excavation.

Another significant practical aspect emerges from consideration of the emplacement hole spacings indicated in Tables 20 to 24. For certain waste unit type/storage period/hole depth combinations, the derived emplacement hole spacings become very small. For several of the waste unit types, storage periods in excess of 60 years allows emplacement hole spacings of
less than three hole diameters, for planar arrays. These situations are depicted by shading in Tables 22, 23 and 24.

Where this occurs, excessive stresses could be generated, associated with the superposition of induced stress fields around adjacent holes. The author therefore advocates that, for situations where predicted hole spacings are suitably small, consideration should be given to the adoption of an in-room disposal system, in which boreholes are dispensed with altogether.

This would enable waste units to be emplaced end-to-end in the centre of the emplacement tunnels. Tunnel dimensions could then be considerably reduced, and scan-line interpretations would be simplified since all the excavations would lie in the same plane. This proposal would require detailed consideration of the practical engineering aspects of emplacing the wastes and backfilling the tunnels in a satisfactory manner, without causing any significant radiological hazard to the operatives. Suggested techniques based on 'push-fit' or 'screw-fit' principles are described in Chapter 19, and it is thought that these concepts could provide a satisfactory basis for practical development.

15.8 Summary of Findings

The results of the optimisation study have shown that waste characteristics, including shape, size, overpack thickness and storage period, have considerable influence on the available repository design options. By reference to a thermal design model, it is possible to determine a range of possible repository configurations with a view to minimising the areal
extent, characteristic rock volume and scan-line intensity of repositories in jointed rock, so that the risk of intersecting transmissive discontinuities is minimised. The opportunity also exists to improve volumetric efficiencies by several orders of magnitude, by comparison with those achieved in the international repository design proposals put forward to date.

It has been shown that waste units based on full-size Harvest or AVM-type canisters are to be preferred rather than half-diameter variants, since the 'minimum size' repository systems which result are relatively compact, with smaller characteristic rock-volumes and scan-line intensities. The selection of AVM or Harvest canisters also places lower demands on the level of sub-surface activity, and thus reduces overall ventilation demands and obviates the need for enlargements for passing points, etc. The AVM-type canister (1.0m long) is to be preferred rather than the Harvest-type canister (3.0m long), since radius bends would not have to be introduced at tunnel intersections, and relatively small tunnel dimensions could be accommodated.

It has been shown that the provision of overpacks is beneficial in terms of reducing overall repository size, and thereby enhancing confidence in the level of containment afforded by the host rock (in addition to the provision of corrosion protection). A 100mm overpack thickness is found to provide a significant level of improvement, but the provision of a 300mm thick overpack does not bring about a significantly better result for most emplacement configurations.

It has been shown that extended interim storage for periods of 90 years or more can produce significant benefits in relation to the design of repository systems.
in jointed host rocks, especially for moderate hole depths. The provision of deep emplacement hole configurations can provide very compact repository systems, but a similar result may be obtained by increasing the storage period and providing overpacks.

Based on the above, the author advocates the adoption of AVM-type waste canisters with 100mm thick overpacks, and an interim storage period of about 90 years. Moderate emplacement hole depths are recommended, in view of the greater technical difficulties foreseen in constructing deep emplacement holes and emplacing the waste units and backfill materials. The associated costs of disposal, including provision of overpacks, temporary storage, repository construction and waste emplacement would be of the order of £180M, at 1982 U.K. prices, for 4000 tonnes of vitrified waste at 15% by weight of fission products, delivered to the repository over a twenty year period.

Consideration should also be given to the adoption of an alternative system, involving the in-room emplacement of high-level waste units, such that units are placed end-to-end within small diameter emplacement tunnels. This would require further detailed examination of waste characteristics and storage period requirements, but could produce further improvement in terms of reducing the risk of encountering transmissive discontinuities within the host rock.
16. REPOSITORY DESIGN CONSIDERATIONS

16.1 Introduction

Previous chapters provided a basis for a re-examination of the philosophy of repository design and construction for each of the host rock types under consideration with a view to increasing overall confidence in the level of waste containment, whilst maintaining economic and engineering viability. The matrix of drillholes concept is not considered further, since it has been shown to apply only to special circumstances involving the disposal of relatively limited quantities of high-level wastes. Attention is therefore focused on tunnel network concepts, based on in-room and in-floor emplacement systems for high-level and intermediate-level wastes.

The results of the optimisation studies described in Chapter 15 have clearly shown that the design of an underground disposal system for high-level wastes should include careful consideration of waste unit parameters and conditioning requirements, in addition to the relationship between construction planning and host-rock performance. Results obtained have been described in terms of construction planning for crystalline rock high-level waste repositories. However, due to the generality of the model, the broad trends which have emerged can also be applied in considering alternative pre-disposal strategies and repository layouts for high-level waste disposal in other rock types. Some of the trends may also have implications for intermediate-level waste repository planning.

Subsequent sections of this chapter discuss some of the essential aspects of repository planning and design,
according to rock type, and indicate where (in the author's view) changes in design emphasis could bring about effective improvements in comparison with the proposals put forward to date.

16.2 Crystalline Rocks

The minimum-repository size philosophy described in Chapter 15 has three essentially non-site specific elements which are relevant to the outline design of high-level waste repositories in crystalline rock, namely:

- the adoption of separate construction and waste emplacement phases, so that the need for duplication or enlargement of excavations is avoided

- the use of specially-designed plant and equipment to reduce ventilation requirements (and hence the amount of redundant space occupied by ventilation systems) to a minimum

- the adoption of a pre-disposal strategy (including waste form characteristics and storage periods) which minimise characteristic rock volumes, scan-line intensities and tunnel dimensions; and which result in suitably low rates of waste unit delivery and emplacement

The optimisation process described in Chapter 15 provides a basis for determining the preferred waste unit size, shape, overpack thickness and interim storage period, together with a range of possible repository configurations which will enable the designer to minimise the probability that randomly-spaced...
transmissive discontinuities will intersect the excavations.

However, the separate phasing of construction and waste emplacement provides further opportunity to improve the conceptual design layout in the light of insitu ground conditions. Provided that the directional trend of major discontinuity sets can be determined insitu, advantage may be gained by orientating the longitudinal axes of the principal repository excavations parallel with the lines of minimum fracture frequency. Thus, the final orientation and spacing of emplacement tunnels and boreholes (consistent with thermal design constraints) should be regarded as highly site-specific, and should not be rigidly defined at the outline design stage.

Thus, decisions concerning the final layout and orientation of the network should be based on exploratory excavations undertaken as part of the preliminary repository development process (34). Figure 91 illustrates how such a system could work for a real repository scheme, on the basis of fracture data obtained during shaft construction and shaft-bottom development. The construction of twin-shafts (along the axes of preliminary boreholes), joined by connecting dog-legged tunnels or working chambers, should provide sufficient rock exposure to allow a reasonable estimation of the three-dimensional discontinuity pattern, (108). The leading dimensions and orientation of the emplacement networks may then be tailored to suit the rock fracture system, so as to ensure that the containment properties of the intact rock are exploited to maximum advantage.

Figure 91 provides only a 2-dimensional indication of the relationship between repository scan-lines and the
rock discontinuity system. However, the vertical siting of the emplacement tunnel horizon and the corresponding optimum emplacement hole depth should also be determined on the basis of insitu fracture frequency data, from a range of possibilities, determined from studies similar to those described in Chapter 15.

Overall, the need to provide scope for detailed design decisions, on the basis of insitu data obtained during repository development, reinforces the need for conservatism in devising an appropriate pre-disposal strategy for high-level waste disposal in crystalline rocks. The provision of long-storage periods and AVM-type waste units with 100mm thick overpacks would appear to provide a range of suitably compact repository design options. The necessary insitu adjustments in the light of real ground conditions would then result in a small increase in characteristic rock volume and corresponding decrease in scan-line intensity which, coupled with a systematic planning of scan-line positions, should provide a tailor-made design layout, providing enhanced levels of confidence in the containment properties of the overall system.

However, the general assumption that crystalline host-rocks will require little or no support at the depths envisaged must be viewed with caution. As shown in Chapter 13, several of the proposals which have been put forward involve construction depths of 1000m; yet assessments based on competence factors suggest that nominal support may be necessary for the rock strengths envisaged; see Table 17 and figure 28.

It should be noted that the conventional form of support in such situations comprises rock bolt reinforcement systems, which generally allow significant movement
Although they may provide an efficient and effective method of underground support in conventional rock tunnelling situations, they could be unacceptable in relation to repository construction, since increases in peripheral fracture porosity must be avoided; see Chapter 10.

Hence, it would be preferable to remove any associated risks of longitudinal migration by siting the repository at a horizon where rock competence factors are relatively high (implying fully elastic ground deformation response). Under these circumstances, no support should be necessary and smooth-blasting techniques should ensure minimal peripheral damage. However, deformation monitoring should be carried out to verify that no dilatant shear movements have occurred.

Where ground movements indicate peripheral disturbance, special measures are required to isolate these areas from the excavations in which wastes are to be emplaced. The use of 'cut-off collars' to intercept the zone of peripheral disturbance could provide a suitable technique, which is discussed more fully in 16.5 below.

16.3 Argillaceous Rocks

In terms of their containment properties and structural performance, the stronger argillaceous rocks may be regarded as generically similar to the crystalline types. The discussion in 16.2 above is therefore broadly relevant. However, since sedimentary strata tend to be relatively limited in vertical extent and homogeneity there is an incentive to minimise emplacement hole depths for high-level wastes. In addition, due to the deleterious effects of prolonged
exposure on certain types of argillaceous materials, as described in Chapter 8, it is considered that all unsupported excavations in argillaceous rocks should receive a membrane seal coating at the construction stage, to avoid oxidation and/or mechanical degradation due to exposure to the atmosphere.

Argillaceous rocks of intermediate strength could be partly self-supporting or could become highly plastic and unstable. The potential variation is enormous, and no single repository design concept can be fully representative (none are found in the available literature). As a broad principle, however, it is considered that rigid structural linings should be installed close behind the working face, to avoid post-peak brittle fracture development; see Chapter 10.4.

From the repository engineering viewpoint, plastic clay formations must be treated as a distinct group; since various special repository construction problems arise. It is emphasised that tunnelling at great depths in clay strata is outside conventional underground construction experience. There are no guiding precedents other than civil engineering experience in relatively shallow tunnel construction in soft clays, at relatively low stress levels; see Chapter 10.2.

In consequence, although clays undoubtedly offer considerable advantages as a confining medium, questions concerning the economic and engineering viability of constructing a deep repository system assume major significance. These are most conveniently illustrated by reference to the proposed Belgian repository system which has been described in Chapter 13; see figure 53.
The general arrangement adopts a combination of in-room and in-floor disposal systems for intermediate and high-level wastes, respectively, and utilises a combination of in-room and in-floor disposal systems; with a rectangular tunnel network configuration broadly similar to those which have been advocated for other host-rock types. The main distinction is the great technical difficulties associated with construction, and the need to provide immediate ground support, using thick tunnel linings.

The Boom clay formation is extensive in the lateral sense and, at the Mol site, lies at approximately 160–270m below ground level. Repository construction is proposed at approximately 200m depth; and at this level, the undrained shear strength of the material, $C_u$, and the total overburden pressure, $\sigma_v$, are about 850 kN/m² and 4500 kN/m² respectively. Hence the over-load factor, $\sigma_v/C_u$, is greater than 5 (see table 17), and corresponding ground loadings are an order of magnitude greater than those hitherto experienced in clay materials. Broadly similar instability problems could arise in other potential clay host formations in Europe; such as the Oxford clay in Britain.

As described in Chapter 13, outline proposals for the construction of the Belgian repository involve the extensive use of ground-freezing to stabilise the ground. This applies particularly at tunnel junctions, which are notoriously difficult to construct in unstable clay strata.

Support is to be provided by massive, segmental, bolted cast iron linings (98). Based on the assumed ground loadings, outline calculations for a nodular cast iron lining, suggest that a minimum skin thickness of about
50mm and a minimum flange thickness of about 200mm would be required. These would be very heavy linings by normal standards, and would represent a considerable proportion of the total construction costs. Yet, assuming an average corrosion rate of 15μ per year, on the outside face only, they could be expected to survive no longer than 3500 years. Galvanising the linings, or applying bitumen coatings would increase their design life. Nevertheless, it must be expected that an uncontrolled transfer of stress onto the backfill would occur at some unknown time after commissioning the system, due to lining corrosion.

In view of the above, and in the light of the discussion in Chapter 10.4, it is considered appropriate to examine the feasibility of a radical design alternative which avoids the use of structural linings in strata of this type. It is proposed to consider the use of shield-tunnelling methods rather than ground freezing; since the former should be more economic, and could also provide new opportunities for the on-site conditioning, processing and placement of backfills; see Chapter 17. It is also considered preferable to adopt an in-room disposal system throughout, so as to avoid problems associated with large diameter borehole construction in squeezing ground, and to exploit the limited vertical thickness of the host material (as a containment barrier) to maximum advantage.

Configurations involving multiple junctions are least suited to shield-driven tunnel construction; and a system involving circular or spiralling tunnels is considered more appropriate. Figure 92 illustrates a conceptual layout in which twin spiralling tunnels, totalling some 25km in length would provide adequate capacity for 9000 high-level waste units, each 1.5m
long, emplaced longitudinally at 2.5m centres. This design capacity is equivalent to the Belgian requirements (assuming waste units 0.3m in diameter and 1.5m in length; see table 15). Assuming 4m internal diameter emplacement tunnels, the space surrounding the high-level waste units could theoretically also provide adequate capacity for the disposal of up to 168,750 m$^3$ of intermediate-level wastes.

The three shafts and the main shield chambers/marshalling areas would have to be constructed by ground-freezing methods; although full-face excavation of the shafts may be possible, using pressurised bentonite as a stabilising fluid. The 1400m long, 6m diameter access tunnel which links all three shafts would be constructed using a slurry-pressure tunnelling machine, and a cast iron lining would be installed.

The remainder of the repository construction would then be carried out concurrently with waste emplacement and backfilling, as a staged process. This would follow a sequence which allows progressive development and commissioning, whilst providing for the re-use of tunnel linings for temporary support.

The junction chambers in the access tunnels (probably constructed using ground-freezing methods) would provide 'resting points' for the tunnelling shields. This would enable each loop to be traversed at a fairly rapid, steady rate of progress. This is an essential factor, since the clay would behave as a 'squeezing ground' and continued advance of the shield would be required to prevent 'lock-in', due to the development of high ground interaction pressures. However, the overall rate of repository development could be matched to the rate of
waste production.

A possible sequence for the development of one half of the spiral system is shown in figure 93. Two tunnelling shields are required: a slurry-pressure shield for tunnel construction, and a second, highly specialised shield, capable of removing linings and emplacing the waste units and the backfill in a single pass. As shown, the construction shield would traverse each loop in an anti-clockwise direction. The waste emplacement shield would subsequently traverse the same loop in the opposite direction.

The re-utilisation of tunnel linings in successive loops means that a maximum of only 6.5km of linings would be required. However, the principal objective of removing linings would be to minimise potential longitudinal migration problems and enhance the level of waste containment. Any savings in lining costs would be regarded as a secondary benefit.

In the author's view, very significant advantages may be gained by the removal of tunnel linings for repositories in clay host strata. Figure 94 represents the stress changes and pore pressure response generated in the vicinity of a circular tunnel in clay where the overload factor $\sigma_v/Cu = 5$; the latter value being chosen in view of the similarity with the conditions likely to be encountered at Mol. For convenience, it is assumed that $K_o$ is unity, so that stresses in the plane of the tunnel are isotropic.

On opening the tunnel, if the ground were capable of remaining in an elastic state, the tangential stress at the tunnel walls $\sigma_\theta$ would increase to $2\sigma_v$, and the radial stress $\sigma_r$ would reduce to zero; see figure 27. With increasing distance from the tunnel boundary,
both stresses would approach the virgin insitu stress $\sigma_v$.

However, due to the high overload factor, the elastic limit of the material will be exceeded, so that the ground undergoes undrained plastic shear within a zone of radius $R$, where $R$ is defined by (48):

$$ R = a \exp \left( \frac{\sigma_v - P_i - Cu}{2Cu} \right) \quad (1) $$

where $a$ is the tunnel radius

$P_i$ is the internal support pressure

Within this zone, the radial stress increases with distance, according to the relationship:

$$ \sigma_r = P_i + Cu \ln \frac{r}{a} \quad (2) $$

and the tangential stress increases according to the equation:

$$ \sigma_\theta = \sigma_r + 2 Cu \quad (3) $$

Beyond the plastic zone defined by equation 1, the stresses are 'elastic' and are given by:

$$ \sigma_r = \sigma_v - Cu \left[ \frac{a}{r} \right]^2 \exp \left( \frac{\sigma_v - P_i - 1}{Cu} \right) \quad (4) $$

$$ \sigma_\theta = \sigma_v + Cu \left[ \frac{a}{r} \right]^2 \exp \left( \frac{\sigma_v - P_i - 1}{Cu} \right) \quad (5) $$

Using the overload factor $\sigma_v/Cu = 5$ and $P_i = 0$ for the immediate undrained loading condition, equations 1 to 5
may be used to plot the resulting stress-field, as shown in figure 94. Based on the chosen overload conditions, the ground is shown to be in a state of plastic disturbance within a region approximately 7.4 times the tunnel radius.

It may also be shown (48) that the theoretical pore-pressure change $\Delta u$ within the plastic zone is determined by:

$$\Delta u = p_i - \sigma_v + Cu \left(1 + 2\ln \frac{r}{a} + \alpha\sqrt{6}\right) \quad \ldots \ldots \quad (6)$$

and within the elastic zone:

$$\Delta u = \alpha\sqrt{6} Cu \left[\alpha\right]^2 \exp \left(\frac{\sigma_v - p_i}{Cu}\right) - 1 \quad \ldots \ldots \quad (7)$$

where $\alpha$ is Henkel's pore pressure coefficient.

The possible ranges of $\Delta u$ are shown in figure 94, for values of $\alpha$ in the range -0.6 to +0.12. These correspond to the likely range for heavily over-consolidated to normally-consolidated clays. In general, negative values of $\alpha$ are considered most likely for the clay host materials likely to be encountered in repository construction.

Figure 94 shows that a high negative pore-pressure response may be induced by mobilising the full strength of the ground around the tunnel openings. Even in normally consolidated or lightly over-consolidated clays, the effect may extend over a considerable distance from the tunnel boundary. and the hydraulic gradients induced are likely to be higher than the original ambient values by several orders of magnitude.
The process of pore-pressure dissipation is highly complex, and involves the suction of water into the zone of negative $\Delta u$. Thus, where the water demand of the negative zone is greater than the surplus of the positive zone (or where no positive zone occurs), water must travel from great distances. The time required for ambient pore pressures to be re-established throughout the domain is potentially enormous, and thus the through-flow conditions necessary for waste leaching and radionuclide migration would be significantly delayed; see Chapter 18.2.

It is apparent that the swelling process induced in the surrounding clay could produce substantial loading at the tunnel boundary. Therefore, the removal of repository linings during the waste emplacement and backfilling process would also result in very high compaction pressures being applied to the fill, eliminating any imperfect interface contacts within the system.

Thus, the removal of linings, concurrent with waste emplacement and backfilling, is seen as a highly desirable objective, because it produces a predictable and significant improvement in the waste containment properties of the system. The confining qualities of clays, especially those with high sorption capacities, are excellent; and the additional confinement which would result from an 'engineered' delay to the re-establishment of through-flow conditions in the region of the repository offer clear advantages. In the author's view, therefore, the concept of removing linings merits serious engineering research.

However, the engineering difficulties associated with this proposal should not be underestimated. The spiral
tunnel concept which is advocated requires an ability to steer the construction shield to within a few metres of the intended horizontal and vertical alignments; despite exceptionally high ground pressures. The provision of adequate face support would also require large stabilizing pressures. Based on the conditions assumed for the Mol site, for example, a stabilizing slurry pressure of about 1.1 MN/m² would be required to reduce the overload factor at the working face to a value of 4. This value of face support pressure is thought to be at least three times greater than those utilized to date.

In view of the above, it is apparent that considerable development work would be required to produce a satisfactory tunnelling shield for deep-level repository construction in plastic clay host formations. The following are among the factors to be considered:

- The need for particularly high diameter/length ratios, or an effective form of articulation to allow the negotiation of curves

- Cutter heads, slurry plenum chambers and ancillary equipment capable of providing substantial support pressures at the tunnel face

- Peripheral jacking systems capable of allowing diametral contraction of the shield to prevent 'lock-in' due to excessive ground squeeze

- Peripheral lubrication to reduce the jacking pressures required to advance the shield

The development of the emplacement shield would pose further difficulties. Figure 45 shows an outline design
which could form a basis for further study. Without more detailed research and development, ultimately involving full-scale tests, it is speculative to consider whether such a system could be made viable in practice. However, the straight access tunnel shown in figure 92 would provide an ideal opportunity to conduct machine performance trials.

The foregoing account has outlined a radical conceptual repository design alternative for plastic clay host formations. It involves the removal of linings to eliminate uncertainties associated with their corrosion in the long term and to improve the overall containment afforded by the host environment, due to associated inward ground movements accompanied by widespread pore pressure changes. The system also adopts an in-room system for the combined disposal of intermediate and high-level wastes which should result in an increase in overall volumetric efficiency of at least an order of magnitude, and would maximise the vertical thickness of the natural clay barrier around the wastes.

Due to the difficulties foreseen in shield-driven tunnel construction at great depth in plastic clays, it has been shown that the concept is dependent on considerable research and development. Nevertheless, it is thought that development costs could be offset by economies gained in the re-use of linings and increases in volumetric efficiency. Very high costs would clearly be associated with the extensive use of ground-freezing, as advocated in the original Belgian concept, and the practical feasibility of this form of construction over the long tunnel lengths envisaged has not been proven.

However, by an optimisation process similar to that described in Chapter 15, it is thought that the
manipulation of high-level waste unit parameters could bring about further improvements in the proposed spiral tunnel system. As shown on table 17, the proposed Belgian pre-disposal strategy involves storing the units for 50 years prior to disposal. Extending this period by 15-20 years would considerably reduce the waste heat output per unit of waste volume (see figure 10) and would therefore allow a greater volume of waste to be incorporated in each unit. This in turn would allow a reduction in the number of units required. Adjusting the storage period to allow an increase in waste unit diameter to 0.5m, for example, could reduce the total number of high-level waste units from 9000 to 3240. The total length of tunnelling required would then be reduced from 25km to about 11km, and the corresponding areal extent of the repository system would be more than halved.

Unlike the case of a crystalline rock repository, the primary benefit of size reductions would not be to improve the level of confidence in the containment properties of the host formation, but to offset the relatively high costs associated with construction and commissioning. Due to the need to avoid multiple junctions, haul lengths around the spiral system would be relatively long. Hence, any size reductions which could be economically achieved by modifying the pre-disposal strategy could also prove highly beneficial in operational terms.

16.4 Saliferous Rocks

As described in preceding chapters, saliferous rocks are fundamentally different from crystalline and argillaceous rocks, in terms of their waste containment.
properties. The absence of circulating groundwater is their principal attribute, and the lack of any significant sorption capacity may be regarded as immaterial, provided that a groundwater circulation route is not established as a result of repository construction.

The need to ensure that water from surrounding country rocks is excluded from the repository assumes over-riding importance, in view of the solubility of saliferous rock minerals and the highly corrosive properties of brines; see table 7. Inundation by non-saline waters would result in progressive enlargement of cavities by dissolution, and the provision of backfill barriers or waste unit overpacks could not be expected to provide adequate long-term protection of the wastes in the event of such a scenario. Hence, from the waste containment viewpoint, all the 'eggs are in one basket' and it is essential that all possible measures are employed to prevent the development of longitudinal groundwater flow paths into the host-rock, via the repository excavations.

Fortunately, as described in Chapter 10.2, the time-dependent visco-elastic 'flow' behaviour of saliferous rocks tends to prevent transmissive features from developing around backfilled openings within the host-rock, in the long-term. However, four important repository design requirements can be identified, as follows:

- the limitation of the number and size of repository shafts to an absolute minimum
- the adoption of special methods of shaft construction through water-bearing overburden strata
and cap-rock materials to prevent the development of a ground-water invasion route

- the siting of the emplacement area as far as possible from the margins of the formation (i.e. the interface with surrounding country rocks)

- the development of a design procedure to cater for potential emergencies, in the event of a flooding of the repository during the short-term period prior to full commissioning and closure of the underground system

It is clear from the above that the minimum repository size philosophy should be invoked in order to allow the design of a repository system well away from the margins of the formation, and to reduce the number of shafts required to supply the facility. Thus, it would appear prudent to adopt relatively long storage periods for high-level waste units and/or to consider the use of overpacks as a means of reducing minimum waste unit spacings (as opposed to an anti-corrosion measure). In view of the relatively high thermal conductivity of saliferous rocks in comparison with crystalline types, it may be anticipated that characteristic rock volume and area requirements for a given waste unit type and storage period should be significantly lower than those indicated in Chapter 15.

For bedded salt deposits, planar configurations or moderate-depth cuboidal configurations are to be preferred. However, over-excavation of shafts (or boreholes) in order to determine the optimum emplacement horizon runs the risk of penetrating water-bearing horizons at lower depth. In general, it is considered that salt domes would provide a greater measure of
confidence in the optimum siting of a saliferous rock repository, based on a compact cuboidal configuration, using relatively deep emplacement holes for high-level waste disposal.

However, unlike crystalline rocks, the separate phasing of construction and waste emplacement is considered undesirable. At best it is considered that all access and waste delivery routes should be constructed separately, as a first stage operation, recognising that these 'semi-permanent' excavations must be constructed over-size or regularly trimmed to account for ground closures. Machine excavation, using electric-hydraulic road-header type plant, would minimise risks associated with peripheral disturbance which could (in the short-term operational period) provide a potential groundwater invasion path.

To cater for the possibility of flooding during subsequent stages of development, it is considered that 'cut-off collars' or enlargements should be constructed at strategic points within the principal excavations; including the shaft bottom area. The general concept of 'cut-off collars' is described in 16.5 below. However, in the present context, they would comprise localised tunnel enlargements in which a massive rigid lining would be installed after excavation, so that subsequent ground closure would be expected to transmit large interaction pressures. Within the opening, water-tight doors should be fitted so that, in the event of a flooding emergency, the affected area could be sealed off to allow subsequent recovery. Their siting should therefore ensure that more than one escape route is available so that, in the event of a flooding incident, it should be possible to recover the situation without abandoning the waste emplacement and backfilling
activities already in progress.

It is considered that the subsequent development of waste emplacement areas and the emplacement of wastes should proceed concurrently, as a second stage operation; with simultaneous backfilling and sealing of the whole facility, on a retreating basis. In this way, the minimum number of excavations would be exposed to the immediate effects of a potential flooding incident at any one time. In addition, the stand-up time of excavations within the waste emplacement zone would be minimised, so that the risk of occurrence of tertiary creep phenomena would be avoided; see Chapters 10.2 and 10.4 above.

It is significant that the international proposals for saliferous rock repositories do not appear to have attempted to minimise the size of the facility, the size of the excavations, or the stand-up time requirements for excavations in waste emplacement areas. The concepts reviewed in Chapter 13 have all adopted in-floor methods for high-level waste emplacement and in-room methods for intermediate-level waste emplacement. The latter have included a variety of proposals, involving the use of large caverns, with mechanical waste stacking arrangements or large vertical 'bunkers' with random placement of waste units and backfill. Ground closure due to creep has generally been relied upon to ensure containment and densification of fills, both for high-level and intermediate-level emplacement areas.

However, none of the proposals attempts to exercise direct control over rates of ground closure or peripheral ground disturbance; although considerable reliance is placed upon predictions of ground response. In the author's view, it is essential that the
uncertainties inherent in such predictions should be minimised in the repository design context, wherever adverse consequences may result. Thus, a greater measure of engineering control should be incorporated. This implies that excavation tolerances in all waste emplacement areas should be small, and that stand-up time requirements should be minimal. Ground closure without peripheral disturbance may then be predicted with greater confidence, due to the avoidance of large strains; see figure 31. In addition, backfill requirements may be minimised and advantages in terms of volumetric efficiency may be obtained.

The author advocates a system involving the overlapping of high-level and intermediate-level waste emplacement areas; in which the intermediate-level waste emplacement system is based on adaptation of long-wall mining techniques, as illustrated in figure 96.

The essential design requirement for this proposal is the selection of an adequate panel width to ensure rapid ground closure onto the waste units (the 'goaf' side). A special fill would be required at the sides of the emplacement panel, where strains are at a minimum. This would require 'end-on' filling of the outer access tunnels ('main gate' and 'tail gate'), and the incorporation of fill blocks of the same size and shape as the waste units at each end of the long-wall emplacement area. A temporary filling of high-level waste emplacement tunnels would also be required to provide long-wall face stability and to minimise the amount of uncontrolled ground deformation.

It is considered that the concept could provide a greater degree of engineering control over repository construction and waste emplacement in saliferous host
rocks than concepts proposed hitherto. The system avoids many technical uncertainties concerning ground closures and the waste stacking and backfilling and sealing arrangements for intermediate-level waste disposal areas. The following advantages are also apparent:

- the amount of bulk fill is reduced to an absolute minimum, and maximum possible overall volume utilisation is achieved
- only four tunnel access-ways are required, reducing the number of potential routes for groundwater ingress
- ground closure is controlled during the commissioning process, and hence minimal reliance is placed on predictions of long-term ground deformation response

Since excavation and backfill volumes are minimised, there may also be significant cost advantages; see Chapter 17 below. However, as in the case of the alternative clay repository concept described in 16.3, these cost reductions must be offset against the development costs and capital investment costs associated with the specialised equipment envisaged.

Research and development work would be required in devising a satisfactory automated system for the long-wall face; both for excavation and waste emplacement. Difficulties to be overcome include:

- the need for dust suppression during excavation
- provision for radiation shielding during maintenance work
of the design of conveyor systems and ram assemblies for correct alignment and high-density emplacement of the waste units.

However, long-wall mining technology has advanced considerably over recent years, and it is thought that the proposed concept could be readily developed on the basis of existing engineering technology.

The method of shaft construction for saliferous rock repositories is clearly an area which requires special attention, in view of the problems associated with groundwater invasion. In common with argillaceous host rocks, it may be anticipated that unstable, water-bearing overburden strata must be penetrated in order to gain access to the host rock. In the author's view, the procedures advocated to date, based on the use of ground-freezing techniques and special 'impermeable' linings, fail to take adequate account of possible adverse effects which could lead to the establishment of anomalous vertical flow paths. Detailed consideration of an alternative engineering approach to this problem is provided in 16.5 below.

16.5 Special Construction Measures

It has been shown that 'conventional' construction processes can lead to the development of anomalous longitudinal flow paths under various conditions, for each of the host rock types under consideration. Some suggested methods for reducing or eliminating problems of this type have been described in preceding sections.
of this chapter. However, two types of situation merit special consideration. These are:

- problems associated with shaft construction through sequences of unstable, water-bearing overburden strata; typified by concepts for repositories in argillaceous and saliferous host rocks

- problems associated with peripheral disturbance, due to post-peak brittle fracture in rock materials which cannot be regarded as completely self-supporting

International repository design proposals for argillaceous and saliferous rock repositories have generally adopted ground-freezing as the preferred method of ground stabilization for shaft construction. In addition, composite shaft linings have been specified which are designed to be impermeable in the transverse sense; see figure 60. In all cases, it is proposed that the freezing process should be extended well into the host material, in order to obtain an effective shaft 'seal' at the base.

However, the proposals which have been put forward generally fail to address the following significant problems introduced by ground-freezing:

- the freeze-thaw process generates a disturbance in the surrounding ground due to the expansion/contraction of water at the ice-point. In non-cohesive strata, this is likely to result in a zone of increased porosity in the immediate vicinity of the shaft. However, the 'impermeable' linings, as currently envisaged, will only inhibit transverse flow. A permanent vertical flow path anomaly may
therefore be created as a result of ground-freezing during the construction stage.

- in unstable cohesive strata, the freeze-thaw process in the surrounding material may disrupt the fabric of the ground; resulting in loss of strength, unpredictable lining loads, and changes in permeability.

- the freeze-tubes outside the area of the completed shaft are themselves potentially transmissive features, unless fully withdrawn and grouted-up; a factor which is likely to be highly problematic in unstable ground at large depths.

In general, it appears that current repository design proposals have failed to recognise and counter these potentially detrimental effects associated with ground freezing. However, in the author's view, it should be possible to solve the problem by constructing the shafts in two stages, as shown in figure 97. Twin small-diameter shafts could be constructed using conventional ground freezing techniques, involving the use of a peripheral ring of freezing tubes. These 'pilot' shafts would be lined with a temporary lining and would be connected at the emplacement horizon by a tunnel, as a preliminary stage of repository development. The second stage of shaft construction would then involve reaming-out the shafts to the full diameter, using raise-boring techniques, as shown in figure 97a.

During the reaming-out process, the ground freeze would be maintained over the upper length of the shaft. However, the lower section of the freeze-tubes would be grouted to a pre-determined level over the length to be
reamed-out, as shown in figure 97b. The procedure would involve placement of a batched quantity of grout in each hole while coolant circulation is temporarily halted. Once grout set is achieved, the flow of coolant could be resumed by blowing out a pressure gland in the walls of the inflow pipe, using a packer assembly. Repetition of this sequence, hole by hole, would enable the freeze to be terminated over a given length of the shaft, allowing sufficient time for the reaming-out process.

The procedure would involve the consumption of the grouted lengths of the freeze-tubes by the reaming cutter-head, with spoil removal from the second shaft. The permanent lining installation would then be carried out in stages, progressing from the shaft bottom upwards. The need to incorporate swelling materials on the outside of the permanent shaft lining has already been described. However, as an additional precaution, it may be desirable to construct cut-off collars at various points along the shaft length. In order to avoid further problems with peripheral freeze-tubes, special temporary shaft lining segments could be used over the sections concerned. These segments could incorporate freeze-tubes within their structure, allowing isolated sections of the ground to be re-frozen at a later stage, for the removal of linings and construction of cut-off collars during the final shaft sealing process.

This proposed method of shaft construction could avoid many of the backfilling and sealing difficulties created by conventional ground-freezing methods. It may be noted that minimal reliance is placed on remedial grouting. In view of the many uncertainties involved, it is considered that ground treatment methods, in general, should be regarded as a precautionary measure.
rather than as a long-term first line of defence; see Chapter 17 below.

The provision of 'cut-off collars' is another technique which could be applied in the prevention or reduction of peripheral flow path anomalies. However, the method would be limited to indurated materials which are prone to the development of fissures due to brittle failure associated with localised over-stressing. It is therefore applicable to crystalline or hard argillaceous host rocks which are not wholly self-supporting, or in areas of 'bad' ground.

The 'cut-off' collar concept is illustrated in figure 98, and involves the construction of an enlargement (in a tunnel or shaft) over a short length of the main excavation. It is envisaged that extreme care in the construction of the enlargement should minimise any peripheral disturbance at its outer boundary associated with the excavation process itself. The zone of peripheral disturbance around the main excavation would therefore be truncated at each end of the enlargement, so that the continuity of the associated longitudinal flow path anomaly would be interrupted.

Chabannes, Stephenson, et al have suggested that an opportunity exists to optimise the length and diameter of cut-off collars in terms of the effect of these variables on longitudinal flow (38A). This assumes that groundwater flowing along the periphery of the main excavation would be forced into the body of the rock mass, and would subsequently find a route back to the periphery on the opposite side of the cut-off collar, via a small zone of disturbance around the enlargement.

However, their discussion fails to recognise the
influence of stress re-distribution as a limiting factor. The predicted re-distribution of stress around tunnel and shaft openings, as depicted in figures 27 and 29, assumes that plane strain conditions apply; i.e. that the longitudinal dimension of the excavation is very long in comparison with its diameter. In practice, the limitation of peripheral disturbance around an enlargement will depend upon the limitation of its length, such that end-restraint contributes to its overall stability.

Thus, if the diameter of the cut-off collar is determined on the basis of the predicted extent of the zone of disturbance around the main excavation, the maximum length which can be excavated without localised over-stressing must be determined on the basis of a 3-dimensional stress analysis. The evaluation of the effect of a single cut-off collar, in terms of a reduction in peripheral migration potential, would then provide an indication of the number of such enlargements required at discrete positions along the main excavation.

The evaluation of relevant requirements cannot be included within the scope of this thesis. However, the 'cut-off' collar principle is seen to be a potentially valuable design technique in mitigating against the adverse effects of peripheral flow phenomena. The author therefore considers that the topic merits detailed analytical research in order to ascertain relevant design principles; especially for brittle host rocks.
17 BACKFILLING AND SEALING ASPECTS

17.1 Introduction

This chapter provides a broad assessment of the backfilling and sealing aspects of repository design and construction. A considerable volume of geoscientific research has been carried out into the physical and geochemical retention properties of various backfill materials (22, 25, 154, 172, 201). However, it has been shown in Chapter 13 that, in the majority of cases, the engineering design proposals for repository backfilling and sealing are poorly developed.

Geoscientific research appears to have been mainly concentrated on materials such as hydraulic cements, siliceous aggregates, bentonite clays and crushed salt; materials which are traditionally employed in conventional bulk filling and underground stowing operations. In this chapter, it is proposed to present the results of a wider review of potential backfill material combinations; to indicate the likely costs of repository backfilling, based on the use of representative fill material assemblages (according to rock type); and to provide an outline indication of the preferred backfill systems for each of the principal host rock types under consideration.

Elsewhere, the author has provided a comprehensive review of the repository backfilling and sealing problem; including a full account of relevant properties for a wide range of candidate backfill materials (7). Here, it is proposed to highlight only those materials whose long-term properties can be predicted with reasonable confidence, and to concentrate on the practical engineering aspects of their use as a part of the
overall repository containment system.

Preceding chapters have discussed various repository design and construction measures which can increase confidence in the long-term containment of radioactive wastes, by removing or mitigating against aspects which tend to have detrimental effects on the integrity of the host formation as a natural barrier. In the author's view, an equal emphasis should be placed on the design of the backfiling and sealing system; since opportunities exist for the creation of very effective, and relatively predictable, engineered barriers.

If appropriate backfill selection, design and emplacement methods can be devised, it should be possible to develop an engineered barrier system which greatly enhances the containment properties of the repository system as a whole. This may be achieved in two ways:

- by providing a backfill barrier whose containment properties largely duplicate those of the host rock;

- by providing a backfill barrier whose properties are complementary to those of the host rock; for example, where a particular host rock tends to retain certain radionuclides more effectively than others, it should be possible to include backfill materials which will compensate for its sorption deficiencies.

Much reference has been made, in the international literature, to the concept of developing 'engineered barriers' (including high-integrity waste units and high-containment backfills). However, to date, it
appears that no serious attempts have been made to develop an integrated approach to the design of such systems, or to predict their performance in the repository engineering context; see Chapter 18.

Requirements for repository backfilling are far more onerous than for conventional backfilling or mine stowing operations. Material property requirements are extremely wide-ranging, and methods of waste and backfill emplacement will have an important impact on overall effectiveness and reliability. Backfilling and sealing therefore cannot be considered in isolation from other aspects of repository design, construction and commissioning. Neither can conventional experience be expected to provide a fully adequate basis for development.

The author has found it convenient to draw a distinction between the independant and interactive functions of an engineered repository backfilling and sealing system (7). These may be broadly defined as follows:

**Independent Functions**

- to restrict or prevent the flow of groundwater;
- to retain radionuclides by physico-chemical reactions.

**Interactive Functions**

- to provide physical support to waste units;
- to act as an efficient heat transfer medium, by conduction of heat from high-level waste units into the host rock;
to inhibit the corrosion of waste unit claddings and the leaching of radionuclides by means of chemical buffering reactions;

- to provide structural support, where necessary, to ensure the long-term stability of the surrounding host rock and to prevent the formation of anomalous flow paths at backfill-rock, backfill-waste interfaces;

- to selectively retain those radionuclides which tend to remain mobile in the host rock/groundwater system.

The two independent backfill functions above duplicate the principal containment mechanisms which generally operate within the host rocks under review. However, the remainder depend on interactions between the backfill and the waste units and/or the natural underground environment.

It is clear that no single backfill material can fulfil all the listed requirements to best advantage. Designed backfills are therefore expected to comprise a variety of material assemblages, conditioned to produce the optimum physical form for emplacement. Thus, the possible variations in form and consistency of a particular backfill material may have considerable bearing on its range of applications. Designed fill assemblages may also be expected to vary from one part of the repository system to another, according to host rock type.

The author has chosen to define the terms 'active' and 'redundant' to distinguish between the backfilling
requirements for different part of repository systems (7). 'Active' areas comprise those repository excavations in which waste is to be emplaced; namely, the emplacement tunnels and drillholes within the repository network. The tunnels overlying high-level waste emplacement holes are included in this category, recognising that they should logically be used for the emplacement of intermediate-level wastes. 'Redundant' areas comprise the access shafts and tunnels of a repository system, which serve a useful purpose only during the relatively short-term operational period. They contain no waste, and yet they constitute potential direct migration pathways between the emplaced wastes and the biosphere.

Objectives for backfilling and sealing in active and redundant areas differ according to host rock type. In crystalline and argillaceous rocks, there is an incentive to create a high-integrity engineered backfill system within active areas, so as to improve the level of waste containment beyond that attributable to the host rock alone. However, in redundant areas, the over-riding objective is to prevent the development of preferential radionuclide migration paths leading from the emplacement zone. In redundant areas, the ideal situation would be to return the rock to its original condition prior to excavation. There is no incentive to provide a backfill whose containment properties are significantly better than those of the surrounding rock; since the preferred migration routes will be developed on the 'weakest link' principal.

In saliferous rocks, the situation is more or less reversed. Here, the objective of design is to prevent the risk of groundwater invasion into active areas of the system. As previously described, the effects of
dissolution are such that the design of an effective backfilling system, capable of resisting the prolonged adverse effects of inundation and through-flow conditions, is a practical impossibility. Hence, the minimisation of backfill requirements in active regions, as advocated in Chapter 15.4 above, is the only logical design objective.

However, it is apparent that the redundant areas of saliferous rock repositories must receive special treatment. This applies particularly to shafts and shaft-bottom connection tunnels, which represent the most vulnerable parts of the system. It may be noted that the saliferous rock repository system proposed by the author in Chapter 15.4 minimises the extent of redundant excavations within the repository.

17.2 Materials

Considerable backfill materials research has been carried out within the international development programmes. Unfortunately, however, standardised testing procedures have not been universally applied. The majority of the available data is therefore based on current material usage, supplemented by the results of a limited number of special tests. The latter reflect the specific repository design requirements envisaged by researchers in each of the countries concerned. In all cases, difficulties have been experienced in attempting to reproduce the wide range of physical and chemical conditions which are likely to occur in the real underground repository environment. Problems in attempting to predict long-term performance, outside the experimental time-scale, have been particularly difficult.
As previously noted, the majority of the available data relates to the more traditional materials, in widespread use within the construction industry. Specialised materials, such as hydrothermal cements and bentonite/quartz mixtures, have received special attention by some research organisations and (170, 189); and other candidate materials, including a variety of sorbants, chemical buffers and other additives, have also been studied. However, for most materials, current assessments are based on an extrapolation of existing knowledge from applications largely un-related to underground construction and radioactive waste containment.

The material properties of principal interest for repository backfilling and sealing purposes comprise the following:

- heat transfer properties
- hydraulic properties
- chemical buffering properties
- radionuclide retention properties
- mechanical properties

Table 27 presents a list of candidate backfill material groups and their principal material sub-types, based on a comprehensive review of the available geoscientific literature carried out by the author(7). The table also identifies the principal attributes of each material group in terms of the emphasis placed on the properties listed above.
In order to provide an indication of the potential application of these various materials, some form of screening and classification procedure is required to indicate the status of current knowledge concerning the probable performance of each material in the repository environment. The longevity of backfill materials is of special concern; and since the extent to which the properties of backfills may be allowed to deteriorate as a function of the declining toxicity of the wastes cannot be evaluated in practice, it is considered appropriate to assume that all materials must exhibit a longevity measurable in thousands of years.

In the present context, longevity may be considered as synonymous with long-term geochemical stability, and therefore depends upon the compatibility of the backfill material with its host environment. A material which remains stable under normal ambient conditions may undergo considerable deterioration in another situation, where the environment is subject to transient or permanent change. In a repository situation, such changes may be attributed to the following:

- variations in physical conditions, such as temperature, stress or displacement, which may cause mechanical change or disruption of the material fabric.

- changes of phase or chemical composition, due to slow but significant geochemical reactions between the backfill and the host rock or the waste units.

Physical changes affecting the backfill may be predicted or controlled with reasonable confidence. However, geochemical changes are less easy to predict; since materials which exhibit no measurable short-term
reactivity may undergo major chemical change over an extended period.

Geological evidence, when available, provides the most reliable guide to long-term material stability. Materials which have remained stable over geological periods, under conditions similar to those arising in a deep-level repository, are clearly able to satisfy the longevity criterion. Geological evidence of this kind may be appropriate in the evaluation of the long-term stability of naturally-occurring materials, and may also be applied, by analogy, to certain synthetic materials of similar composition. It should be noted that geochemical stability is most apparent in geological materials which are the end-products of long weathering chains (e.g. clays and quartz minerals). However, these may not possess all of the many other desirable properties required in a repository backfilling and sealing system.

Archeological records provide another source of information concerning the longevity of both naturally-occurring and man-made materials; notably cements, mortars and pozzolanic mixtures. However, the archeological record extends only for a period of about 2000-3000 years, and artifacts which have survived often tend to be those which have experienced atmospheric or shallow sub-surface conditions; usually in areas which have remained dry. Statements which claim that the preservation of remains, such as those found in the sand deserts of Egypt, effectively demonstrate the feasibility of long-term radioactive waste containment are therefore inherently mis-leading when applied to deep underground repositories in most of the western nuclear power-producing countries.
Numerous experimental laboratory studies have been carried out to monitor the geochemical stability of potential backfilling materials. However, these inevitably present difficulties, due to the restricted time-scale of laboratory-based methods. Attempts have been made to overcome these by accelerating chemical reaction rates, utilising increased temperatures and pressures, in specially designed autoclaves (201). However, the author considers that such procedures are of dubious value, since the kinetics of the reactions are different from those occurring in practice. Results may suggest the potential for reactions which cannot occur, even over extended periods at the lower temperatures and pressures which prevail in the repository environment.

Other researchers have adopted a thermo-dynamics approach in order to assess the long-term potential for geochemical change (128). This involves the identification of all chemical species present in the system, and postulating all possible reactions amongst them. Thermodynamic data are then obtained for all the chemical components and their reaction products, and theoretical principles are invoked to predict reaction rates. However, there is a shortage of thermodynamic data (reaction rate laws and constants) for the various mixtures, solid solutions, etc. to be considered. This approach cannot, therefore, provide a satisfactory indication of backfill longevity, based upon present scientific knowledge.

A combined approach appears to be the only practical method of evaluation. Since geological evidence is the most convincing proof of material longevity, naturally-occurring materials which appear to possess suitable properties are clearly the most promising
candidates, based on existing technical knowledge. Considerable research is required to provide convincing data regarding the longevity of other materials; particularly the more complex mixtures. However, experiments which fail to take due account of the actual physical and geochemical conditions which govern the potential for long-term deleterious change are clearly of little value.

Hence, for present purposes, a simple screening system is proposed, which incorporates priority rankings for longevity, based on current knowledge concerning the long-term geochemical stability of potential backfill materials. The basis of these rankings is as follows:

<table>
<thead>
<tr>
<th>RANKING</th>
<th>DESCRIPTION</th>
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</thead>
<tbody>
<tr>
<td>1</td>
<td>documented evidence of geochemical stability over geological periods of time</td>
</tr>
<tr>
<td>2</td>
<td>documented evidence of stability over significant time-intervals, suggesting a high probability of satisfactory long-term performance</td>
</tr>
<tr>
<td>3</td>
<td>some doubt as to long-term stability under certain physical-chemical conditions</td>
</tr>
<tr>
<td>R</td>
<td>fundamental uncertainties concerning longevity; basic research required.</td>
</tr>
</tbody>
</table>

Based on current knowledge, it is considered that only materials assigned a priority ranking of 1 or 2 can be considered as potentially suitable for incorporation.
within a repository backfilling and sealing system. Those with rankings 3 or R may be tentatively considered; recognising that they may subsequently be discounted on the basis of new geochemical evidence.

Table 28 indicates the rankings assigned to the various groups of backfill materials contained in Table 27. As shown, it is considered that spoil materials, clays, pulverised fuel ash, hydraulic cements, minerals/aggregates, bitumens, graphite, certain silicate grouts, and carbons may be considered to be potentially suitable on the basis of current knowledge.

Materials whose stability may be adversely affected by conditions within a deep underground repository include zeolites, natural pozzolanas and metals/metalllic compounds. These are promising materials, those geochemical stability must first be carefully assessed in relation to their working environment within the repository system.

The longevity of organic chemical grouts poses some fundamental uncertainties. It is considered that several of these materials must be discounted in terms of any long-term sealing function; whilst others must be considered with caution. Unfortunately, the organic chemical grouts tend to have the greatest range of potential applications for the treatment of low permeability materials; see figure 33. Their formulations are very complex and are normally guarded by commercial secrecy. However, their long-term stability is generally questionable, and special construction methods which avoid the necessity for ground treatment by grouting are therefore to be preferred; see Chapter 16.5.
The longevity criterion clearly provides some basis for establishing materials research priorities. However, it is also necessary to make an outline comparison of materials in terms of their known design properties. It should then be possible to make an outline assessment of the relative merits of alternative materials when used to fulfil specific functions within a given repository situation.

A ranking system is adopted as a basis for comparing material properties, which is broadly similar to that previously described in the assessment of material longevities. The proposed property ranking system is as follows:

<table>
<thead>
<tr>
<th>RANKING</th>
<th>DESCRIPTION</th>
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<tbody>
<tr>
<td>1</td>
<td>favourable properties</td>
</tr>
<tr>
<td>2</td>
<td>high probability of favourable or neutral properties</td>
</tr>
<tr>
<td>3</td>
<td>possibility of adverse properties</td>
</tr>
<tr>
<td>R</td>
<td>properties unknown; fundamental research required</td>
</tr>
</tbody>
</table>

The resulting distribution of rankings is shown in Table 29. However, it is emphasised that the classification cannot take account of the full range of material properties. For example, clays are known to sorb several species of radionuclides, whilst some metallic elements or compounds can retain other varieties (notably anions) not accounted for by the clays. Likewise, the type of structural support provided by clays is different from that provided by concrete. These
factors reflect the need to utilise materials whose properties complement those of the host rock. Multiple rankings (for each of three host rocks under consideration) have not been employed, and the rankings shown are those which correspond to the use of the materials in an appropriate host rock environment, see 17.3.

Table 29 provides a preliminary indication of potential design applications for each material group, and may be used to identify the following:

- single material groups/sub-types likely to be capable of performing one or more specific design functions within a repository
- material groups/sub-types which can only fulfil a given range of design functions when mixed with other constituents
- the range of mixes which should be examined in order to identify the optimum material formulation for a given design function.

An example of all three applications may be described by reference to clay materials. Based on the assigned property rankings shown in Table 29, these may be suitable for use both in redundant areas of the repository and in active areas occupied by intermediate-level (non heat-generating) wastes. However, due to their relatively poor thermal properties they are unlikely to fulfil design requirements in the vicinity of high-level wastes, unless other materials are added to improve their thermal conductivity. The property rankings suggest that aggregates, metallic compounds or graphite may be suitable additives, and
research into the combined properties of relevant mixes may be warranted. Other materials and mixtures may also be examined in a similar manner to assess design priorities and materials research requirements.

A broader picture may be obtained by a combined examination of Tables 28 and 29. On this basis, material types having combined longevity and other property rankings at level 1 or 2 are considered suitable for incorporation in backfill design proposals. Consideration of the lower order rankings is likely to suggest a suitable system of priorities for fundamental materials research.

Based on the foregoing account, it is considered appropriate to define three backfill material categories as follows:

Category 1 Materials

Materials whose properties are currently known to an extent which allows their design performance to be assessed on a quantitative basis.

Category 2 Materials

Materials whose properties are known to an extent which enables them to be tentatively incorporated in repository backfill designs, pending the outcome of more fundamental research concerning specific properties.

Category 3 Materials

Materials which possess desirable attributes, but whose properties are poorly understood in the repository engineering context.
A detailed assessment of the attributes of all the materials listed in Table 27, based on knowledge and experience within the civil engineering and mining industries and a review of relevant radioactive waste management and disposal literature, has been presented elsewhere by the author (7). The results in Table 30 indicate the groupings which emerge as a result of the screening and classification process previously described. The summary below indicates the principal attributes of the materials in each category, and summarises their potential applications in terms of repository design.

**Category 1 Materials**

As would be expected, the majority of the Category 1 materials indicated in Table 30 are traditional engineering materials widely used in the construction industry.

Under appropriate circumstances, spoil materials from each of the three host rock types considered may be suitable for use as backfills, and the majority of the international repository design proposals envisage the re-utilisation of excavated material within the backfilling and sealing system (29, 35, 98, 121, 188). However, in practice, the suitability of spoil materials will be governed by the method of construction and requirements for stockpiling and conditioning.

Crystalline rock spoil will not be adversely affected by stockpiling over long periods, and should require relatively little processing; although crushing and screening will be required to produce a specified particle size distribution. However, spoil material
from hard argillaceous rock repositories is likely to deteriorate on stockpiling, and would probably have to be dried, ground and graded (and carefully stored) to produce the required material formulation. Depending on the mineralogical composition of the material, it may therefore be considered preferable (on both technical and economic grounds) to utilise an off-site source of bulk filling material for hard argillaceous rock repository systems.

For clay host formations, the re-use of repository spoil is likely to be an attractive proposition. The engineering properties of clays (particularly remoulded clays) are generally well-understood and those containing a predominance of illite, kandite and smectite minerals may be regarded as category 1 materials.

Where shield tunnelling methods are employed for repository construction, utilising a slurry plenum chamber for face-support as advocated in Chapter 16.3, slurrified clay spoil would be removed to the surface by pumping. The treatment of this spoil, in order to produce an engineered backfill formulation, could then form an integral part of the construction process. The addition of other materials, to improve geochemical retention, such as smectite clays or zeolites, or the addition of sand to increase thermal conductivity, could be readily accomplished under controlled conditions at a surface processing facility.

Saliferous rock spoil could also be re-utilised effectively as a bulk fill, although special care is required in stockpiling, in view of the solubility of the material. Most of the international proposals envisage sub-surface storage and processing to eliminate
this problem. However, the author considers that this approach is undesirable, in view of the associated increase in excavation requirements (and hence backfilling problems) within the redundant regions of the repository.

Crushing and screening plant would be required to achieve the desired fill grading, as in the case of crystalline rocks; and where it is intended that insitu re-crystallisation will ultimately occur under the influence of ground loading, it is apparent that the initial grading of the fill material is likely to have an important influence on performance. Pulverised fuel ash (PFA) and bitumen have been considered as possible binders for crushed saliferous spoil, and it seems likely that these materials could play a significant role. However, in the author's view, the concept of insitu re-crystallisation of saliferous rock fill is one which must be treated with caution in the absence of adequate research data. A more positive approach, with more predictable results, would be to manufacture preformed fill blocks by compressing the crushed spoil under controlled conditions, and utilising these fill blocks in conjunction with a bitumen 'mortar'; see Chapter 18.2.

Apart from repository spoil, several other materials are considered suitable as 'category 1' backfills. Portland cements and siliceous aggregates, when combined to produce concrete, form a strong and versatile material whose properties can be closely controlled and are well understood. Concrete therefore has many potential applications in all three host rock types. However, in view of its shrinkage properties, problems associated with the development of separation planes at interface areas must receive special attention.
PFA is a synthetic pozzolana which has a beneficial influence on the properties of concrete and cement grouts, in terms of reduced shrinkage and bleeding and increasing chemical durability (130, 152, 162). It may also be included as an inert filler, in combination with a variety of other backfilling materials.

Siliceous aggregates are strong, durable materials which may be deemed suitable for use in a variety of bulk fills. Their relatively high thermal conductivity generally favours their use in fills used to surround high-level waste units.

Bitumens possess hydrophobic qualities which suggest a wide range of applications in the construction of impermeable seals (68, 114). Their thermo-setting properties could also be used to assist placement. Although their thermal conductivity is poor, certain varieties also have surprisingly high temperature stabilities. The rheological properties of bitumens may be used to particular advantage in the sealing of inter-face areas, and in other situations where viscous or plastic 'flow' properties are desirable.

Category 2 Materials

Category 2 materials are those which are known to possess desirable properties; although a lack of sufficient data prevents a realistic assessment of their performance in the repository environment. The majority of the materials considered to fall within Category 2 are specialised materials having potential applications as sorptives, impermeable seals and grouts.

Palygorskite clays (including palygórskite and
attapulgite) are thought to possess anion sorption properties which could be highly beneficial in the design of engineered barriers (83). However, their sorption and hydraulic properties are generally less well understood than those of the kandites and smectites (177). These clay minerals therefore require further study, under conditions representative of the repository environment.

The zeolites and natural pozzolanas possess important cation sorption properties (113, 130, 141, 162, 211). Zeolites, in particular, have high cation exchange capacities and molecular sieving properties which could outweigh their relatively poor thermal, hydraulic and mechanical performance. Incorporation of these sorptives into an engineered backfill could therefore be highly beneficial in terms of waste containment. However, research is required to establish the mix specifications which will achieve the optimum retention capacities and selectivities in the repository situation. It is also necessary to establish the functional life of these materials (retention limits) in order to assess the quantities which should be incorporated in backfill designs; see Chapter 19.

Polymer cements offer considerable advantages over Portland cements in terms of strength and permeability (152, 162). However, they have not yet been used in underground applications, and their placement properties and geochemical longevities require careful evaluation. Hydrothermal cements may exhibit similar qualities (189). However, at present, it appears that difficulties in placement may outweigh their apparent advantages, and practical research is required to assess the feasibility of incorporating hydrothermal cement seals within repository systems.
Anhydrite has been suggested for use in high-level waste emplacement holes in saliferous rocks, due to its relatively high thermal conductivity (25). Another potential application is as a constituent of crushed saliferous rock spoil. On contact with infiltrating groundwater, it should hydrate to form relatively insoluble gypsum, accompanied by an increase in volume. Although a considerable amount of data is available concerning the properties of anhydrite, further research is required to establish rates of dissolution and hydration under an appropriate range of groundwater flow conditions. The mechanical properties of crushed anhydrite also require careful evaluation.

Metals and metallic compounds could form an important component of engineered backfills, due to their favourable chemical buffering properties and their ability to undergo irreversible reactions which result in the precipitation of the relatively mobile anionic radionuclides (25, 113, 204). However, at present, there appears to be insufficient empirical data to allow a quantitative assessment of their performance in the repository situation.

The chemical grouts appear to offer considerable potential in ground treatment applications; notably in fissured host rock formations (21, 30, 209, 210). Unfortunately, however, much of the detailed data concerning the chemical composition of these materials are trade secrets. Research is therefore required, in collaboration with specialist manufacturers, to establish the geochemical longevity of the materials concerned and, in the absence of adequate data, their use for long-term sealing functions in high-activity radioactive waste repositories is considered.
Graphite is a virtually inert material of proven longevity, which may be produced in a variety of forms. Certain types of manufactured graphite exhibit very high thermal conductivities and may also be completely impermeable to water (82, 113). These factors suggest a variety of sealing applications; including use as a near-field hydraulic barrier in high-level waste emplacement holes. The lubricating qualities of graphite also suggest that it could offer a means of facilitating an emplacement procedure (or retrieval option) for high-level waste units; see Chapter 19. However, the effects of anisotropy have not yet been fully examined, and it is thought that the material could exhibit excessive brittleness. These factors require further evaluation before the performance of the material can be assessed with confidence.

Category 3 Materials

Category 3 materials are those which are thought to possess desirable attributes, but whose properties or performance in the repository environment cannot be assessed without more fundamental research than hitherto.

Vermicullite exhibits the highest cation exchange capacity of all the clays (83). However, its permeability, temperature-stability and geochemical longevity within the repository environment are poorly understood.

Synthetic zeolites are produced on an industrial scale for a wide variety of applications; as sorptives, molecular sieves, etc. (96). However, their retention
capacities and selectivities for the longer-lived radionuclides do not appear to have been studied in detail. It may be that synthetic zeolites can be produced with retention properties which are superior to those of the natural varieties. However, their thermal and geochemical stabilities require detailed evaluation.

Magnesium oxide/quartz mixtures have been suggested as potentially suitable materials for creating impermeable barriers; due to the ability of magnesium oxide to swell on hydration to form brucite (25). The material could therefore offer an alternative to the current proposals involving swelling mixtures of Na-bentonite clay/quartz. However, little empirical data is available concerning the swelling properties of magnesium oxide, and a programme of more fundamental research is warranted.

Charcoal is used as an industrial filter and molecular sieve, and may therefore form a useful fill additive (113). However, little data is available concerning molecular sieving properties, with radionuclides in solution, and its potentially adverse effect upon the hydraulic properties of fills.

17.3 Backfilling Costs

An outline indication of typical construction costs for a representative high-level waste repository in crystalline rock has been provided in Chapter 15. Similar assessments could be made for repository concepts in other host strata, although the likely construction costs of the concepts which have been advocated, by the author, for repositories in clay and saliferous host rocks cannot be assessed with the same degree of confidence, due to uncertainties concerning
Repository design proposals presented in the available literature generally indicate that the overall cost of underground disposal should represent a small fraction of the corresponding cost of electricity production. However, in most cases, backfilling and sealing aspects have not been developed as an integral part of the design proposals and it appears that the economic aspects of repository backfilling, in particular, have not been examined in detail.

Therefore, an analysis is presented which provides a basis for comparing likely backfilling costs in each of the host rock types under consideration. As a basis for the analysis, a representative repository model is chosen having a constant void volume. The repository model is based on a review of published design concepts, and is used as a basis for assessing generic backfilling costs for an appropriate range of fill material assemblages, according to host rock type.

As shown in Table 18, current international repository design proposals indicate that the proportion of the total volume of a repository which will be occupied by wastes is likely to be small; typically less than 1% for high-level wastes, and 5-40% for intermediate-level wastes. Examination of the current European proposals indicates that typical excavation volumes for combined high-level and intermediate-level waste repositories are likely to be in the range 1-32M m$^3$, and the percentage of the total volume occupied by both types of waste is likely to be about 10-13%.

Based on civil engineering and mining experience, it would appear reasonable to assume that the mean unit
The overall cost of repository backfilling varies according to the backfill material specification for each host rock type, and includes the following components:

(a) the cost of the specified backfill materials at source

(b) the cost of transport to the site

(c) the cost of on-site storage

(d) the cost of processing and batching the materials

(e) the cost of handling and placement within the excavations

The range of backfill assemblages appropriate to each rock type has been identified by the author, based on combinations of materials referred to in 17.2 above, and broadly in line with current materials research.
programmes. However, the more site-specific or design-dependent cost elements (b), (c) and (e) above cannot be fully accounted for in any generalised sense. Fortunately, the remaining elements (a) and (d) are more amenable to generic cost assessments. The present analysis is therefore based on these factors, and the results obtained exclude the site-specific transport and labour elements; although their likely impact, in terms of overall costs, is considered subsequently.

In order to obtain basic material cost data, standard enquiries have been made to a large number of commercial suppliers, specialist contractors and other organisations, as listed in table 32. These enquiries covered all the candidate backfilling materials described in 17.2 above, with the exception of the following:

- chemical grouts
- hydrothermal cements
- polymer cements

For these materials, the overall cost of backfilling is likely to be highly dependent upon placement costs and other site-specific operating constraints. Any assessment which excludes these factors is therefore likely to be misleading. For chemical grouts, for example, costs could vary over a wide range according to the ground conditions, type of grout and the nature and extent of the treatment envisaged. In the case of hydrothermal and polymer cements, basic materials costs are unlikely to be significantly greater than those for ordinary concrete. However, little or no experience is available in their use in underground backfilling.
applications, and their placement costs are likely to be disproportionately high.

For the remaining materials listed in table 32, the prices obtained are based on 1982 rates. Processing costs have been accounted for by specifying that the materials should be in dry powdered or granular form; either bagged or in bulk containers, as appropriate. Prices have been obtained ex-works (i.e. at the source of production) for materials produced within the United Kingdom; or at the most convenient British port for materials supplied from other countries. The resulting basic material cost data are shown in table 33.

By implication, the basic cost data applies to a repository located somewhere in Britain. However, in view of the widespread availability of most of the materials within Europe as a whole, the costs obtained are also considered reasonably representative for all other West European countries. In particular, despite international variations in material prices when expressed in absolute terms, the comparative costs are likely to be broadly similar.

Although the majority of the candidate backfill materials are readily available in Britain and other European countries, there are a limited number of materials whose principal sources of supply are outside Europe. For these materials, the prices obtained include a substantial cost component attributable to sea transport. However, the cost of transport from the source of production (or port of origin) to the repository site is not included in the data shown in table 33. It is thought that the addition of an additional cost element to account for local transport

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(for example, on the basis of a nominal number of route-km by road and rail) would distort the final cost comparisons.

The data shown in table 33 are therefore essentially non-site-specific, and include:

- the cost of the backfill materials at source
- the cost of sea transport, where appropriate
- the cost of processing and batching the materials

These comprise items (a), (b)* and (d) as previously listed.

The methodology employed in the cost analysis involves the comparison of material costs for a representative range of backfill material assemblages (or backfilling concepts), within the reference repository model. The procedure has been applied to four representative host rock environments, namely:

- crystalline rock
- unindurated plastic clay
- strong indurated argillaceous rock
- saliferous rock

* cost of sea transport for materials produced outside Europe only.
In each case, the cost evaluation procedure involves the calculation of insitu material costs, by reference to representative volumes and probable insitu compacted dry densities of the materials involved. Since a wide range of concepts has been applied in each host rock type, the final results are expressed in terms of cost indices. These represent the overall backfill material costs, normalised with respect to the reference volumes indicated in table 31.

The sequence of analysis is presented in standard tabular form in tables 34 to 37. The procedure is most conveniently described by reference to the labelled columns in the first of these tables (table 34) as follows:

**Column (a)** lists a wide range of backfill material assemblages appropriate to the host rock under consideration. As shown, these are grouped as 'bulk fills', 'special fills' or 'additives', according to their primary design role. The material assemblages shown in each category correspond to the outline backfill specifications identified in the available geoscientific literature, together with other concepts considered by the author to be worthy of inclusion.

**Column (b)** indicates the proposed percentage volume utilisation and location of each material assemblage within the reference repository model, broadly in accordance with current repository backfill design proposals.
Columns (c) and (d) indicate the types of material included within each assemblage and their proportions by weight, respectively.

Columns (e) and (h) indicate the corresponding volume of each material assemblage within each of the four repository zones identified in the model (see table 31). The values shown are obtained by multiplying the volume given in column (b) by the overall volume of each zone within the model, as shown on the top of columns (e) to (h).

Column (j) indicates the probable value of the insitu dry density, \( \rho_d \), for each material assemblage, assuming complete infilling and proper insitu compaction.

Columns (k) to (n) indicate the total insitu tonnage of each material type, based on the various assemblages shown. Values are calculated as the product of:

(i) the appropriate volumes shown in columns (e) to (h);
(ii) the percentage composition by weight of the material concerned, as shown in column (d);
(iii) the dry density of the
Column (p) indicates the total tonnage of each material type within each material assemblage. Values are obtained by summing the appropriate tonnages shown in columns (k), (l), (m) and (n).

Column (q) indicates the basic cost per tonne of each material at the source of origin. For the majority of materials, values are obtained directly from table 33. However, unit costs for spoil materials have been assessed on the basis of the probable costs of on-site processing. For crystalline, hard argillaceous and saliferous rock spoil, the cost of crushing and screening is estimated as £1/tonne. For unindurated plastic clays, the cost of processing is estimated as £10/tonne.

Columns (r) and (s) indicate the total cost of each material type, in situ, and the overall cost of the corresponding material assemblage. Values are shown in £1M (to the nearest £0.1M), and are obtained by multiplying the total tonnages shown in column (p) by the appropriate unit cost; as shown in column (q). The total cost of each material assemblage (column (s)), is the sum of the individual material
Thus far, the analysis is seen to evaluate basic insitu material costs, based on a wide range of possible material assemblages/backfilling concepts. However, in order to make meaningful cost assessments and comparisons, the data must be normalised with respect to the void volumes actually available within the reference repository model. The results so obtained are represented by the 'cost indices' shown in the last three columns of the cost analysis tables.

The unit cost indices shown in column (t) represent a numerical ranking of the costs associated with the various assemblages/backfill concepts. These are calculated by dividing the insitu costs for each material assemblage, as shown in column (s), by the corresponding volume, as represented by the sum of the figures shown in columns (e), (f), (g) and (h). The unit cost indices therefore represent the cost per cubic metre of the various material assemblages, as placed within the repository.

The last two columns (total cost indices) provide a numerical ranking of the relative overall costs of the individual material types and assemblages; normalised with respect to the average total backfill material cost (TBMC). TBMC values, shown separately on table 38, are calculated from the cost analysis data as follows:

1. The mean cost per tonne for each backfill category (bulk fills, special fills, sorbants, chemical buffers) is calculated by dividing the sum of the total material costs in column (r) by the sum of the total tonnages in column (p).
2. The percentage of the total backfill weight is calculated for each backfill category, assuming that the total dry weight of backfill materials within the reference repository model is 1.8 Mt (i.e., assuming an overall mean in-situ fill dry density of 1.8 t/m³).

3. The total category cost is estimated as the product of the mean cost per tonne, the percentage weight and the assumed total backfill weight (1.8 Mt.).

4. The average total material backfill cost (TBMC) is then calculated as the sum of the various category costs.

Returning to the cost indices shown in Table 34; the total cost index for each material type (column (u)) is obtained by dividing the cost of the material as shown in column (r), by the TBMC. The cost index for each of the material assemblages is similarly obtained by dividing the cost of the material assemblage shown in column (s) by the TBMC.

The foregoing account described the mechanics of the cost analysis adopted for each of the host rock types under consideration. The results are presented according to rock type, and comprise the cost comparison data shown in Tables 34 to 37 and the corresponding total backfill cost summaries shown in Table 38.

The first and most readily anticipated conclusion is that materials which are relatively expensive at the source of origin (see Table 33) do not necessarily impose a high cost penalty in relation to the overall cost of backfilling.
The cost contribution of each material is seen to depend upon its percentage volume utilisation within the backfill system as a whole, and the insitu density attained. In the present analysis, both factors have been assessed by combining the results of a comprehensive literature review with a liberal use of engineering judgment. Despite the generic nature of these assumptions, the accuracy of the results is considered to be the maximum attainable, based on the present state of development in the field of repository backfilling research.

Overall results are best described according to rock type; firstly by reference to the detailed cost comparison data shown in tables 34 to 37; secondly, by reference to the total backfill material cost summaries shown in table 38.

**Crystalline Rocks (table 34)**

As expected, the cheapest backfill material assemblages are those incorporating crushed rock spoil. Based on the unit cost indices shown in column (t), the use of a bulk fill comprising crushed rock spoil/bentonite clay is approximately one-third of the cost of concrete and one-fifth of the cost of crushed quartz/bentonite clay.

Both bentonite and cement are relatively expensive in terms of their unit cost, but a surprisingly high cost penalty is incurred when a high-grade of quartz sand (from off-site sources) is specified. This is illustrated by the high total cost indices for the crushed quartz/clay bentonite assemblage as advocated in the Swedish KBS proposals (122), to fill the whole of the repository. Two-thirds of the cost of this assemblage is attributable to the crushed quartz sand.
This illustrates the need for research to assess the sensitivity of backfill performance to variations in materials specification. For example, it appears that a 50% saving in material assemblage cost could be made if natural sand (basic price £6/tonne) were substituted for the crushed quartz sand as specified. Alternatively, a ten-fold saving could be made by substituting crushed rock spoil (assumed basic price £1/tonne).

The cost of special fills can vary greatly, as illustrated by the two concepts utilised in the analysis; namely, high-quality concrete cut-off collars, and expansive blocks of pre-formed bentonite (placed as circumferential seals). The ratio of the corresponding total material costs shown in the column(s) is 1:16. This is due to differences in both basic unit material costs and their comparative volume utilisation. Unit cost indices (representing costs per unit volume insitu) are at a relatively low ratio of 1:1.8; since these are primarily dependent on basic unit costs and insitu densities. However, the total cost indices for the special fill assemblages (which indicate their relative contributions to the average total backfill material costs) are in the ratio of 1:10; i.e. the introduction of expansive bentonite blocks is ten times as expensive as the construction of the concrete cut-off seals, in terms of basic material costs.

The mean unit cost indices for additives (sorbants and chemical buffers) are shown to be five to thirty times greater than for the bulk fill assemblages to which they are introduced. Nevertheless, the total cost indices show that additives do not make a disproportionate contribution to the overall cost of backfilling.
By reference to the total cost indices, most of the material assemblages are seen to make broadly similar contributions to the total backfill material cost. The greatest range occurs within the bulk fills and, as previously noted, the extensive use of crushed quartz sand obtained from off-site sources can impose a relatively high cost penalty.

Unindurated Plastic Clays (table 35)

Fewer backfill concept variations have been considered for unindurated plastic clays than for other host rocks. This is partly because only one conceptual repository design proposal has been put forward, namely the proposed disposal facility at Mol, in Belgium (98A). However, it is also considered that less emphasis is likely to be placed on engineered barriers than for the indurated rocks; in view of the high sorptive capacities of plastic clay host formations in general.

Considerable emphasis is placed upon the use of clay spoil within the bulk fill; although a bentonite/quartz sand assemblage has been assumed for infilling the voids in high-level waste emplacement holes. The only 'special' fill considered is a cement/clay grout, for backfilling the voids behind tunnel linings, etc. In view of the high sorption capacity of plastic clays, a relatively low proportion of 'additives' (sorptives and chemical buffers) has been included by comparison with the crystalline and strong argillaceous rocks.

The unit cost indices for clay spoil (used as a constituent of the bulk fill) are seen to be high in comparison with those for the other host materials. This is because the unit costs associated with clay
spoil processing are relatively high (assumed to be £10/tonne for clay processing by filter pressing, drying, etc; as opposed to £1/tonne for rock crushing and screening). The addition of cement as 15% by weight is seen to increase the unit cost index by a factor of 1.75, whilst the unit cost index for the bentonite/quartz assemblage is approximately 3 times greater than for clay spoil.

The unit cost indices for special fills (clay/cement grouts and additives) are comparatively high, as anticipated. However, volume requirements are relatively low and their overall cost contribution is not excessive (as shown by the total cost indices).

The total cost indices shows that the cost of processing the clay spoil makes a significant contribution to the total backfill material cost. This aspect therefore merits careful engineering and economic appraisal. The use of cement for stabilising the bulk fill (i.e. preventing consolidation), and the use of metallic compounds as additives for anion sorption/chemical buffering both have significant impacts on total material costs.

It should be noted that the cost of tunnel linings has been excluded from the analysis. For the purpose of this analysis, it is assumed that the cost of linings will be separately accounted for in the repository construction cost estimates. However, as described in Chapter 16.3, it is considered that the development of a concept involving the removal of linings in clay host formations would be more beneficial in terms of long-term waste containment. Although the associated construction costs are not likely to be cheaper, backfilling costs could be lower than those estimated
for the present repository model.

**Strong, Indurated Argillaceous Rocks (table 36)**

No backfilling concepts for strong, indurated argillaceous rock repositories are found within the available literature. Therefore, the range of material assemblages shown in table 35 has been formulated largely by engineering judgment.

It is considered that backfilling in the stronger argillaceous rocks will require an approach similar to that adopted for crystalline host formations. Both may be regarded as relatively rigid, self-supporting materials, whose containment properties are strongly influenced by the presence of discontinuities, and whose sorptive properties are poor by comparison with the plastic clays. Thus, although differences occur in the bulk fill assemblages, the range of special fills and additives used in the cost analysis are identical to those shown in table 34.

As shown, the cheapest bulk fill assemblage identified is a mixture of 20% dry pulverised rock spoil and 80% natural sand. For this mixture, it is assumed that the pulverised spoil material would possess some swelling properties to ensure a dense infilling after the saturation phase. By comparison, a mixture of 80% pulverised rock spoil combined with 15% cement and 5% PFA is found to be almost twice as expensive (in terms of unit cost indices). Alternative bulk fills, comprising concrete and bentonite/quartz sand, are respectively 3 and 5 times more expensive in terms of unit cost indices by comparison with the cheapest bulk fill assemblage.
As in the case of crystalline rocks, the use of bentonite/crushed quartz sand as a bulk fill is found to be relatively expensive. The substitution of an appropriate naturally-occurring sand within the mixture would reduce the unit cost index by approximately one-third.

The cost indices derived for special fills and additives are identical to those shown for crystalline rocks; see table 34. The influence of these components on overall backfilling costs is therefore as previously described.

**Saliferous Rocks (table 37)**

The bulk fill material assemblages envisaged for saliferous rock repositories comprise two distinct types; firstly, those which are based on crushed saliferous spoil, for infilling the majority of the tunnel areas; secondly, the various composite fill mixtures specified for the backfilling of shafts and high-level waste emplacement holes. The former category consists of fills whose primary role is to provide mechanical support within the repository openings. The second category is required to fulfil more specialised functions (in terms of ground water exclusion and heat transfer) in the more vulnerable zones of the system.

Of these bulk fill assemblages, the lowest unit cost index (£7/m³) is obtained for a mixture of 85% crushed spoil and 15% PFA, used as a tunnel infill. The cost index is seen to be doubled (15/m³) by substituting bentonite in place of the PFA.

The unit cost indices for the composite bulk fills used in shafts and boreholes are broadly comparable with one another, varying by a maximum of only 16%. Again, it is
apparent that a high content of crushed quartz sand makes a substantial contribution to the cost of these fills, suggesting that suitable naturally-occurring sands could provide a more economic alternative. The unit cost indices for the various composite bulk fills are some 3 to 7 times greater than for the more basic tunnel fills, and the total cost indices suggest that they are likely to represent over 20% of the total backfill material cost.

Impermeable seals, constructed as engineered barriers to prevent the ingress of groundwaters, are seen to make a significant cost contribution in terms of total backfill material costs. The lowest unit cost index (£38/m$^3$) is obtained for salt-concrete plugs. Bitumen or bitumen/concrete plugs are relatively expensive; with unit cost indices of £136/m$^3$ and £119/m$^3$ respectively. In overall terms, seals are likely to make a cost contribution which is comparable or somewhat greater than the composite bulk fills.

**Total Backfill Material Cost Comparisons**

The average total backfill material costs shown in table 38 provide a basis for overall comparative assessments. It is recognised that the summary is biased towards concepts which have yielded the highest total cost indices in tables 34 to 37. Nevertheless, the backfill assemblages specified in the cost analysis tables are thought to provide a broad coverage of international repository design proposals and backfill research to date. It is therefore considered reasonable to summarise the overall trends by normalising the data with respect to the repository model previously described.
The resulting hierarchy of total backfill material costs (TBMC) is as follows:

- crystalline rocks £51M
- hard, indurated argillaceous rocks £51M
- unindurated plastic clays £39M
- saliferous rocks £21M

Since the reference model includes a total backfill volume of £1Mm$^3$, the results shown are also equivalent to the estimated mean unit costs of backfilling materials, expressed in £/m$^3$. Results may therefore be directly applied to repositories of different sizes on a pro-rata basis; provided that the percentage distribution of the total void volume in each case is broadly similar to that shown in table 31.

The variation in TBMC, according to host rock type, is attributable to differences in the categories of backfill materials, their corresponding unit cost indices and their volume utilisation. In combination, these factors reflect the emphasis placed on the use of special assemblages to create engineered barriers within each of the host rock formations concerned.

In each case, 80-95% of the total backfill weight is attributable to bulk filling materials. The TBMC is therefore particularly sensitive to the mean unit cost of the materials involved. As shown, the mean unit costs for bulk fills in crystalline and hard argillaceous host rocks are similar, and are approximately twice as high as those for saliferous rocks. The proportion of the total backfill material costs represented by bulk fills in each host rock type is as follows:
crystalline rocks 44%
unindurated plastic clays 69%
strong indurated argillaceous rocks 44%
saliferous rocks 67%

The remaining material costs are attributable to fill materials having one or more specialised functions within the repository. These include special fills (in the form of expansive blocks, cut-off seals, sorbants and chemical buffers). As shown in table 38, crystalline and hard argillaceous rock repositories are likely to require the highest mean percentage by weight of these specialised fills (20%). In the unindurated plastic clays, the proportion is estimated as 7%, comprising 5% for the backfilling of cavities behind linings, and 2% for additives within the bulk fill. In the case of saliferous host rocks, only 5% of the total backfill weight has been attributed to specialised fills. These comprise the impermeable seals which represent the primary engineered barrier within the system. It should be recognised, however, that composite fills within the shafts and high-level waste emplacement holes are likely to make a significant contribution to the overall cost of the bulk filling materials; see table 37.

As previously noted, the data used in the analysis has excluded the cost of preparatory works within the repository, and the cost of local transport, on-site storage, handling and placement. The total backfill material costs (TBMC values) are therefore gross underestimates of the overall costs associated with repository backfilling and sealing. The cost elements excluded from the analysis are clearly dependent on site location, design details and labour costs and are therefore highly variable. However, it is instructive
to examine the potential significance of these factors in a general way, recognising that a rigorous approach can only be adopted in the light of specific circumstances.

It may be assumed, for purely illustrative purposes, that the full cost of placing bulk fills, sorbants and chemical buffers could be a factor of 2 times the basic material costs shown in table 38. By comparison it may be reasonable to assume that the full cost of placing 'special fills' and 'seals' is ten times the basic material costs (in view of the more labour-intensive and highly skilled nature of the operations). Applying these factors to the basic cost data in table 38 would result in overall backfilling costs approximately 4 times greater than the total backfill material costs indicated.

The general hierarchy of backfilling costs according to rock type would remain unaltered. However, the proportion of the total costs attributable to bulk filling operations would be approximately halved. Therefore, despite the somewhat arbitrary nature of the illustrative examples given, it appears likely that a major portion of the overall cost of backfilling will be attributable to the placement of the specialised fills which are designed to act as engineered barriers.

It is also apparent that the cost of repository backfilling is likely to represent a significant proportion of overall repository development costs. The outline cost analysis for crystalline (or jointed) rock repositories presented in Chapter 15 suggests that a typical overall cost, excluding backfilling, could be about £180M for AVM units with 100mm thick overpacks emplaced in planar or moderately deep emplacement holes.
after 90 years interim storage. Based on table 23, the total excavated volume is 0.38Mm$^3$ and, from figure 85, the gross volumetric efficiency is 8%. Hence the volume to be backfilled is some 0.35Mm$^3$. Assuming an overall unit backfilling cost of 4 x £51 = £204/m$^3$, the overall backfilling cost would be about £71.5M. Thus, the overall backfilling cost would represent some 30% of the total cost of providing the disposal system. Clearly, for systems in which no repository size optimisation procedure has been attempted, the proportion would be expected to be significantly higher.

Overall, therefore, the analysis of backfilling costs confirms that the minimum repository size philosophy should be applied for economic as well as technical reasons, and suggests that international design feasibility studies, in general, have tended to underestimate the likely impact of backfilling costs on the overall economics of underground waste disposal strategies.
18. ENGINEERED BARRIER SYSTEMS

18.1 Introduction

Reference to engineered barriers has been made in Chapter 17 with specific emphasis on the repository backfilling and sealing system. It has been suggested that backfills can perform both independent and interactive functions within a repository in order to provide the most effective overall containment system. However, the waste units and the host rock are participants in these interactive processes. In order to devise the most appropriate engineered barrier system it is therefore necessary to examine how these interactive processes operate and thence to determine how engineering design can influence overall performance.

The processes involved are both physical and chemical and must be regarded as dynamic over the time-scales involved. Serious shortcomings in the available analytical models and the empirical data-base preclude accurate performance assessments at present. Nevertheless, an outline assessment of various design aspects is presented in this Chapter in order to demonstrate how control could be exercised over the level of waste containment in the near field, to demonstrate how effective engineered barriers could be created, and to indicate the broad requirements for further research.

The generalised radionuclide release and migration processes illustrated in figure 12 involve the following four consecutive stages:

(a) ground water inflow/saturation, leading to the
establishment of equilibrium through-flow conditions

(b) corrosion of waste unit cladings

(c) leaching of radionuclides

(d) radionuclide migration

In the final analysis, it would be desirable to evaluate the performance of engineered barriers in terms of containment models based on a combination of the above; their relative significance depending upon the host rock under consideration.

The time required for groundwater to permeate the near-field repository system and to corrode the waste unit claddings determines the total delay to the onset of the radionuclide release process, and represents a time-span during which waste containment may be considered absolute (stages a and b above).

The radionuclide release rate, after the breach of the waste unit claddings, is dependent upon a leaching process (stage c above). This defines the source-term for the subsequent migration of radionuclides, and is expected to be time-dependent.

The subsequent migration of radionuclides through the repository backfilling and sealing system is the final stage (stage d above) in which engineered barriers can operate; and the rate of migration from the backfill to the host rock may be regarded as a component of a more sophisticated transport model for the repository system as a whole.

Saliferous rocks need to be regarded as a special case,
since the solubility of the rocks and the chemically aggressive nature of brines implies that, in the event of groundwater invasion, the release and migration processes would occur very rapidly. Thus, the delay term implied by stage (a) assumes over-riding significance, and the corrosion, leaching and migration models would be significant only in evaluating the consequences of a catastrophic breach of the repository system.

However, for crystalline and argillaceous formations, the presence of circulating groundwater means that all four stages are significant in terms of engineered barrier performance. Most repository performance studies to date have largely ignored the influence of waste unit design, and backfilling and sealing aspects, in terms of overall containment. Some studies have examined the effects of backfill sorption properties on radionuclide transport (76, 149, 150, 151, 176). However, it appears that no systematic attempt has been made to assess which parameters exert the greatest influence on the sequence leading to loss of containment from an engineered barrier system, based on the four stages outlined above.

Some generalised aspects of the design of containment by engineered barriers, with specific reference to the four stages described, are discussed in 18.2 to 18.4 below. The development of detailed predictive models is beyond the scope of this thesis. However, the results of some outline assessments are described, and these are used to indicate the general priorities for further research.

18.2 Saturation Phase

It is an over-simplification to assume a sharp
transition between the completion of the saturation phase and the onset of waste cladding corrosion; since, in practice, some localised corrosion could occur before complete saturation of the backfill is achieved. However, significant rates of uniform corrosion cannot be established until a through-flow of groundwater occurs. Prior to this stage, any pore waters in contact with the waste units will remain essentially static. Chemical saturation within the near-field pore water system is therefore expected to inhibit the corrosion of the waste cladding materials.

The time required for saturation of the repository will depend upon the permeability of the host rock and the backfill, and the effects of ground/backfill movements on the pore-pressure regime. In highly competent, brittle rocks, the ground will behave essentially as an unyielding mass. Groundwater inflow will occur primarily in a transverse sense, normal to the line of the excavations, and the rate of inflow will be governed by the permeability of the fill and the presence of discrete fissures which intersect the excavations.

Most conceptual design studies have assumed that the time required to saturate the backfill and create through-flow conditions will be measurable in hundreds of years at most. For this reason, many studies have considered that the provision of high-integrity overpacks is the most viable method of creating a significant delay to the onset of the waste leaching process; see 18.3 below.

However, a delay model based on the time required for saturation of the backfill may be equally significant, and may provide a basis for designing a more economic form of barrier. In any event, a redundancy of design
barriers is considered desirable in fissured rocks, due
to various uncertainties in assessing their containment
properties, as described in earlier chapters.

A fill which is prone to shrinkage or consolidation
would not be considered appropriate in an unyielding
host rock material. A hydraulic fill, for example,
should be regarded as completely unacceptable. The use
of swelling materials is preferred; since a much longer
period would be required for pore-pressure equalisation,
and the swelling process could ensure a complete
infilling of the voids within the excavations.

In swelling materials, the rate of pore-pressure
equalisation is governed by the coefficient of swelling,
$c_{vs}$, which may be defined by analogy with the
coefficient of consolidation, $c_v$ (127). The value of
$c_{vs}$ is given by:

$$c_{vs} = \frac{K}{\gamma M_{vs}} \hspace{1cm} (1)$$

where $K$ is the permeability of the material

$\gamma$ is the bulk density

$M_{vs}$ is the coefficient of volumetric swelling
potential

A low value of $c_{vs}$ implies a slow rate of pore-pressure
equalisation and a correspondingly large delay period.
Thus, equation 1 suggests that the following fill
characteristics are desirable:

- The material should have a low permeability and
  should be compacted at a moisture content which is
dry of optimum; since the presence of inter-connected air voids in partly saturated soils decreases their permeability (127).

- A high density should be achieved by imparting a high compaction energy to the material.

- The material should have a high swelling potential as reflected by the coefficient $M_{VS}$. This is dependent on mineralogy, and increases with the activity of the material; see figure 14C.

The optimisation of these parameters could possibly be achieved by the production of dry, highly compressed blocks of high activity clay. This is illustrated by the research undertaken by Pusch, et al, in relation to statically compacted samples of dry, compressed bentonite (170, 171, 172, 177).

Seepage phenomena in partly saturated soils, and the mechanics of the swelling process, are poorly understood in general. However, it is known that the rate of pore-pressure equalisation is slower than for saturated materials, since the pore air must be compressed in advance of the saturation front and must diffuse through the system. Theoretical models which describe rates of pore-pressure equalisation must be obtained by combining equations of:

- continuity of the pore fluid
- continuity of the solid
- motion of the pore fluid
- compressibility of the pore fluid
- compressibility of the solid

Although sophisticated solutions have been developed for
three-phase soil systems (i.e. soils containing pore air and pore water), there is a general shortage of supporting experimental data. For present purposes, therefore, it is expedient to utilise equation 1 for illustrative purposes. However, it must be recognised that the coefficient $C_{\text{vs}}$ strictly applies to saturated soils, and more complex relationships are required to describe the theoretical pore-pressure response of partly saturated materials.

No published data have been found concerning the range of $C_{\text{vs}}$ values for a suitably wide range of potential fill materials. However, Pusch has presented some laboratory data concerning the time required for the maximum swelling pressure to be attained in small samples of partly saturated, compressed bentonite (170). His test results, based on simple oedometer experiments, may be interpreted to predict an equivalent value of $C_{\text{vs}}$ of $2.9 \times 10^{-10} \text{ m}^2/\text{sec}$.

As an illustrative example, therefore, the saturation of a circular tunnel opening, filled with dry compressed bentonite will be considered by analogy with the consolidation of a saturated triaxial sample, draining only to its radial boundaries (127). Under these conditions, the formula for pore pressure equalisation is:

$$T = \frac{\pi R^2}{16 C_{\text{vs}}} \hfill (2)$$

where $T$ is the time required for full pore-pressure equalisation.

$R$ is the tunnel radius
Utilising the value \( C_{VS} = 2.9 \times 10^{-10} \, m^2/sec \), equation 2 provides an outline estimate of fill saturation times for openings of various sizes, as shown in figure 99a. Direct extrapolation of Pusch's laboratory test data, for the case of uni-directional inflow/saturation of a peripheral lining of bentonite blocks, provides an estimate of saturation times as shown in figure 99b.

Both these curves suggest that the delay periods are not likely to be very significant for the excavation sizes and/or fill thicknesses envisaged in practice. However, they must be regarded as lower-bound estimates, for the following reasons:

- The laboratory test conditions, on which the assumed fill properties are based, ensured complete confinement of the bentonite; whereas, in practice, the bentonite would undergo some volumetric expansion to fill voids remaining in the system before generating its full swelling pressure.

- Equation 2 implies a perfect hydraulic radial boundary, at which water is freely available; whereas, in practice, the host rock (fissured type) is a low permeability medium and provides an imperfect hydraulic boundary, in which flow is concentrated at a few discrete points.

It is therefore apparent that in order to derive a realistic model for the saturation of fills of this type, more representative experimentation and theoretical analysis is required. Provided the effects of partial saturation and imperfect hydraulic boundary conditions can be taken into account, it may be possible to demonstrate that saturation times could be significant in comparison with the nominal
corrosion-free life of the waste units.

However, further delay may be introduced by:

- the use of cut-off collars to seal-off any fissures which are recognised as a source of groundwater inflow; see Chapter 16.5.
- the use of hydrophobic (truly impermeable) membranes at the periphery of the backfill system.

The use of the latter could greatly increase the backfill saturation delay period. Figure 100a illustrates a system in which a flexible outer impermeable barrier is placed against the tunnel boundary (e.g. bitumen blocks), and is lined with an inner layer of swelling material. Supposing that ingress of water occurs primarily through fissures which intersect the tunnel, water under pressure could find a path around the backfill-rock interface and enter the swelling material through imperfections in the outer hydraulic barrier. However, if swelling pressures generated within the inner layer are sufficiently high, in relation to the hydrostatic pressure, the impermeable membrane could be forced back against the fissure intersections to prevent further inflow.

It is evident from the above example that considerable scope exists for studying the parametric influences involved in providing an engineered fill barrier to prevent, or greatly impede, the development of through-flow conditions. As illustrated, the characteristics of the host rock must be taken into account in developing the optimum solution.

For clays and indurated argillaceous rocks having low
competence factors, the situation is markedly different, in view of the influence of ground movements. In brittle argillaceous rocks, with low competence factors, it is considered desirable to install rigid linings close behind the working face in order to limit the extent of peripheral plastic disturbance; see Chapter 10.4. In these circumstances, the linings should remain undisturbed and should be incorporated within the backfilling and sealing system; since their de-stressing or removal would inevitably induce further fractures in the vicinity of the openings.

However, the structural life of conventional linings is limited, due to the effects of corrosion in the long term. The internal support pressures must therefore ultimately be transferred to the host rock and/or the backfill; see figure 30. Unless this transfer of stress is brought about in a controlled manner during the backfilling process, it will occur in an uncontrolled manner at some stage after commissioning. The latter is undesirable, since its consequences are unpredictable. For example, it would be impossible to assess whether stress transfer would occur prior to or after the backfill saturation phase.

The backfill should therefore be relatively incompressible and should be placed so that the lining and backfill form a structural monolith. Such a system would minimise the likelihood of any loss of ground support pressure in the long term; preventing further localised rock fracture and associated increases in ground permeability.

Based on the above, the most satisfactory bulk fill is likely to comprise a rigid cementitious-type material. The permeability of such materials is likely to depend
on the properties of the cementing matrix, rather than the aggregate constituents. It is desirable that the matrix should be partly saturated in order to increase the pore pressure equalisation time. Assessments must therefore be based on experiments to determine the influence of mix design, curing conditions, etc. on the free water content and pore-pressure response of the material.

However, it may also be possible to increase the saturation delay-time by introducing impermeable hydraulic membranes and swelling materials around the bulk fill, as previously described. The lining itself may incorporate special caulking systems, bituminous coatings, etc to provide hydraulic barriers to groundwater inflow (and increased corrosion resistance). Swelling materials could be incorporated on the internal and/or external face of the linings, to provide internal support pressure and additional delay time.

As previously shown, the assessment of saturation times for these materials would require experimental work to determine optimum values of coefficients of swelling $C_{wq}$, and the influence of hydraulic boundary conditions. Suitable swelling materials may include compressed dry sodium bentonite/quartz or MgO/quartz mixtures; see Chapter 17. By careful design it may be possible to ensure that these mixtures are suitably stiff in compression, but are also capable of generating high swelling pressures.

A possible arrangement of backfill assemblages is illustrated in figure 100b. In the example shown, the lining segments are bitumen-coated; but the extrados of the lining also incorporates a layer of suitably engineered, dry swelling material, possibly coated with
a retardant film. On erecting the lining, the outer annulus could be grouted with a bentonite grout, so that a uniform swelling pressure would eventually be induced outside the lining; possibly sealing-off a number of fissures by self-injection (175). The joints of the lining would be tightly caulked and bolts, if used, would be grommeted using special materials.

During the final backfilling stage, a secondary layer of bitumen could be applied to the inner surface, followed by an inner lining of dry bentonite blocks, somewhat thicker than the outer layer. The central void could then be filled with an incompressible cementitious material, placed under injection pressure. Any water inflow through the lining would initially be expected to occur at joints and grout holes. However, the inner layer of swelling material could generate sufficient pressure to force bitumen into the entry-points; overcoming the hydrostatic pressure, and so sealing-off the inflow. The incompressibility of the fill material, should ensure that no significant reduction in internal support pressure occurs due to lining deterioration in the longer term.

The example is analogous to that shown in figure 100a. Both involve a rigid boundary within the backfilling and sealing system; although, in the present example, the geometrical design of the lining, and its coating and internal lining systems, is likely to be problematic. Further study of these aspects would be required before a realistic saturation delay model could be developed.

However, the incorporation of a permanent rigid lining within the backfilling and sealing system may not provide the optimum delay period in all situations. For deep-level construction in clays and certain types of
uncemented, weak argillaceous rocks, plastic deformation may occur as a 'flow' process, without the development of transmissive discontinuities. In these conditions, it may be desirable to encourage large-scale plastic deformation of the surrounding host material. The resulting pore-pressure response of the ground could bring about a significant increase in the delay period required for the establishment of through-flow conditions; as shown in Chapter 16.3.

In a repository design concept of this type, the optimum bulk fill assemblage could comprise dry compacted clay blocks, or some other form of prefabricated fill. Figure 100c illustrates a possible backfill assemblage which also incorporates an outer permeable barrier of bitumen or other suitable material. Considerable fill densification could be produced by the transfer of ground loads, and it is anticipated that the time required to establish through-flow conditions could be measurable in several hundreds of years, in appropriate conditions.

As indicated in 18.1 above, the prevention of ground water invasion assumes an overriding importance in saliferous rocks. In order to ensure containment of the wastes, an infinite delay period is required. Since this cannot be achieved by providing corrosion-resistant claddings around the waste units, the objective must be to prevent the ingress of groundwaters into the active areas of the repository.

Unlike the other host rocks under consideration, groundwater inflow can only occur in the longitudinal sense, via the access-ways which penetrate water-bearing strata. It may be possible to provide a totally impermeable bulk infill in strategic locations;
hydrophobic materials such as bitumen or graphite. Alternatively, crushed saliferous spoil could be utilised provided that the material can be made to re-crystallise insitu, to form a homogeneous mass. The latter is crucial; since, if the spoil remains in a particulate (finely divided) state, water could percolate through the fill. A large surface area would then be available for dissolution, possible leading to progressive increase in the size of flow channels.

Re-crystallisation, however, would result in a reduction both in pore volume and in the surface area available for dissolution. Despite the occurrence of residual porosity (resulting in a small but finite permeability), it is likely that a chemical equilibrium would be reached between the backfill and the groundwater. This could lead to the chemical saturation of percolating waters and the re-precipitation of relatively insoluble salts, which could effectively seal-off any remaining flow channels. The result could be the formation of an effective hydraulic seal; similar to the 'cap-rock' which develops at the upper margin of a salt dome.

A systematic evaluation of candidate fill materials is required in order to develop appropriate material formulations and placement techniques. For bituminous materials, the rheological properties of the fill must be designed according to whether the fill is to be placed in a shaft or a tunnel. In shafts, the ability of the material to 'flow' under the vertical dead weight of the fill is likely to assist in achieving a 'lateral spread' to ensure complete infilling. In tunnels, plastic as opposed to viscous deformation properties may be preferable.

For saliferous spoil infill, it is necessary to evaluate
the factors which govern the re-crystallisation process, and the relationships between the mechanisms of flow, dissolution and re-precipitation of salts within the material. It is likely that particle-grading exerts a considerable influence. The use of additives such as PFA may therefore assist in achieving insitu re-crystallisation of the crushed spoil. The introduction of anhydrite may also be beneficial. Although re-crystallisation may occur less readily under dry conditions, the process may be encouraged by slow water percolation. This could cause an expansive hydration process, in which the anhydrite converts to gypsum, effectively sealing-off further inflow.

However, in general, it is considered unwise to rely excessively on self-induced sealing phenomena such as insitu re-crystallisation; since the processes involved are essentially uncontrolled, and are therefore subject to a measure of uncertainty in extrapolating the results of experimentation.

The fill-rock interface and the zone of peripheral disturbance are also areas of great concern; see figure 34. In 18.5 below, the influence of separation at the interface, or localised increase in rock permeability, are shown to be highly significant in the analysis of longitudinal flow. However, within saliferous host formations, the creep properties of the ground may be used to advantage. Provided that the backfill is placed at an appropriate stage, the secondary creep process could eliminate interface flow problems, due to the pressure imposed by the ground. Creep deformation and long-term stress relaxation phenomena will also determine the load to be sustained by the fill and the densification or re-crystallisation effects which could result. As shown in figure 32, a zone of brittle
fracture (plastic yield) can develop around openings in saliferous rocks, due to localised over-stress. Interactive stress analyses, based on rheological models, predict that this zone would be eliminated in the long-term as the deviatoric stress is slowly reduced. This process may be accompanied by the annealing of fractures due to re-crystallisation.

However, once again, it is considered unwise to rely excessively upon re-crystallisation to eliminate potential flow paths in the long term. A better solution could be to install stiff linings in the areas where high-integrity hydraulic barriers are required. Construction of the hydraulic seals would then involve the staged removal of the linings, in short lengths, and the placement of fills of the type previously described. The low volumetric compressibility of the materials (especially bitumens containing fillers) could allow a more controlled transfer of stress to the backfill, with closing of interface contacts and comparatively little peripheral ground disturbance.

However, where localised brittle fracture cannot be prevented, a series of carefully constructed cut-off collars may be beneficial (see figure 98). These could, in any event, provide a valuable additional security measure. A rigid concrete fill (e.g. salt-concrete or polymer impregnated concrete) would probably provide the most satisfactory infill material in these areas.

It is apparent from the above discussion, that the parameters which influence the delay model in saliferous rock repositories require special consideration. A range of potential backfill material assemblages is illustrated in figure 101a. The prevention of groundwater invasion is attributable to a series of
engineered hydraulic barriers to ensure maximum confidence in the containment achieved; see figure 101b.

Longitudinal flow paths must also be eliminated in the shafts above the salt formation. Most construction problems envisage the use of ground-freezing techniques. With this process, it is important to minimise the zone of peripheral disturbance outside the shaft linings as a result of the freeze-thaw process. Linings of the type illustrated in figure 60 may be ineffective in this respect. Shaft linings in frozen water-bearing overburden strata should therefore incorporate swelling materials on the extrados of the structural ring, so that the potential effects of ground loosen ing during the thawing-out stage are offset by the volumetric expansion of the outer fill layer; see figure 101c. An alternative approach to ground freezing in shaft construction should also be considered, as discussed in Chapter 16.5.

18.3 Corrosion Phase

The use of corrosion-resistant claddings, or overpacks, in waste unit construction has been referred to in previous Chapters. In Chapter 15 it has been shown that the provision of metallic (high thermal conductivity) overpacks around high-level waste units can have a beneficial impact in terms of a reduction in repository size requirements; a factor which does not appear to have been considered in the international repository design proposals put forward to date. Overpacks may also perform a useful function in contributing to the radiation shielding of waste units during the short-term operational period prior to waste emplacement.
However, in the engineered barrier context, the primary function of an overpack or waste cladding material is to prevent physical contact between the immobilised wastes and the groundwater. High corrosion resistance is therefore the basic requirement.

For high-level waste disposal in saliferous rocks, repository design optimisation studies may indicate that overpacks are desirable in order to limit repository size requirements for a given waste storage period. However, there would be no significant benefit in specifying a high corrosion resistance for the overpack material, since reliance is placed on prevention of groundwater invasion. The provision of high integrity barriers against corrosion therefore applies only to crystalline and argillaceous rock repository systems; and a distinction is drawn between the incorporation of overpacks for other engineering reasons.

As described in Chapter 4, the 1000-year interval after burial is particularly significant in terms of high-level waste containment, since a large proportion of the fission products decay to negligible concentration levels during this period; see Table 3. Furthermore, heat output rates are reduced, bringing about a corresponding decrease in leach rates; see figure 22. Thus, the 1000-year interval is considered a sensible minimum design goal in terms of corrosion resistance for crystalline and argillaceous high-level waste repository systems.

Although similar transient thermal conditions do not generally apply to intermediate-level wastes, there would appear to be no justification for accepting an overall delay factor shorter than the 1000-year minimum
figure established for high-level wastes. However, in the absence of high heat outputs, corrosion rates will be lower and protection criteria less demanding.

A significant delay interval may be brought about by maximising the sum of the periods required for backfill saturation and for waste unit cladding corrosion. Methods for maximising the backfill saturation period have been outlined in 18.2 above, and it has been shown that the times involved could be very significant. The additional time required for groundwater to corrode through the waste unit claddings may be maximised by the use of corrosion resistant materials, in combination with chemical buffering materials within the backfill system, in active zones of the repository.

Because the groundwater chemistry will be unique to the selected repository site, the life-limiting factors which govern the choice of container materials must be based on a thorough knowledge of groundwater composition, the range of ambient temperatures and pressures, and the corrosion reactions and reaction products of the cladding materials. Metals have been the preferred medium for the fabrication of high-level waste containers, largely due to their high thermal conductivity, strength, ductility, ability to form welds and ability to sustain the high temperatures involved in the waste immobilisation process. Although ceramics have been considered by some researchers, it is not yet known whether the industrial manufacture of ceramic waste containers is viable. Their use is therefore not considered further.

The corrosion of metals is controlled by oxidation and reduction processes at the metal surfaces. Oxidation
reactions, promoted by the presence of oxidising agents (electron acceptors) within groundwater may be expressed:

\[ M \rightarrow M^{n+} + ne \]  \hspace{1cm} (1)

where \( M \) denotes the metal \\
\( n \) denotes the number of electrons produced and is equal to the valency of the metal ion.

Reduction reactions are promoted by the presence of reducing agents (electron donors) thus:

\[ M^{n+} + e \rightarrow M^{(n-1)+} \]  \hspace{1cm} (2)

As the processes of oxidation and reduction are mutually dependent, rates of corrosion can theoretically be controlled by limiting the rate of either type of reaction. In broad terms, decreasing the acidity and lowering the concentration of oxidising species within the groundwater will tend to lower the rate of oxidation reactions, resulting in less severe corrosion; especially for 'active' metals; see figure 3.

However, certain metals and metal alloys can exhibit a phenomenon called 'passivity', under appropriate chemical conditions, due to the formation of passive films. These metals include iron-based alloys, aluminium, nickel, titanium, copper and lead; and it is this group which has been most widely considered for the construction of high-level waste unit claddings (88, 136, 137, 157).

Nutall and Urbanic (157) have demonstrated the phenomenon of passivity for some of these metals in
aqueous solutions by electrolytic experiments. Their generalised corrosion behaviour is illustrated in figure 102, which relates corrosion rate, expressed in terms of the current density $I$, to the oxidising power of the solution, expressed in terms of the electrode potential $E$, at a given temperature and pH. The diagram is divided into three zones, namely:

- the active zone
- the passive zone
- the transpassive zone

In the active zone, the initial increase in corrosion rate with increasing oxidising power is similar to that observed in normal (active) metals. However, for certain metals, a maximum is reached at a potential defined as the passivation potential ($E_p$). Further increase in oxidising power then produces a rapid reduction in corrosion rate, until a minimum occurs at a potential defined as the activation potential ($E^*$). With further increase in potential, the corrosion rate remains effectively at a minimum over a given range, which defines a zone of passivity for the ambient temperature/pH conditions.

It is found that this 'passive' behaviour is attributable to the formation of surface films of chemical reactants at the metal/solution interface, which may be stable over a wide range of oxidising power (157). However, these films may break down locally to produce pitting corrosion if certain aggressive species are present at a threshold potential ($E_{pit}$). At higher values of potential, the zone of transpassive potential is characterised by a rapid increase in corrosion rate, similar to that which occurs in the active zone, and denotes the presence of very powerful oxidisers at a
potential in excess of the threshold value ($E_{\text{trans}}$).

The use of experimental $E/I$ relationships for various metals, as shown in figure 102, is seen to be a potentially powerful tool in predicting metal corrosion characteristics under appropriate conditions. This requires the introduction of appropriate chemical species into the solution (see table 7), and the incorporation of pH and temperature effects. The extension of the concept, by utilising $Eh/pH$ diagrams to predict zones of immunity or enhanced corrosion appears particularly attractive; see Chapter 8.2.

Nuttal and Urbanic (157), Verick and Pourbaix (168, 213) and Garrels and Christ (74), have utilised experimental $E/I$ data for different pH's to predict $Eh/pH$ corrosion stability fields for metals in various aqueous solutions. Examples of these are shown in figures 103 to 105, for the metals which have been most widely considered for use in constructing high-level waste unit claddings.

In all cases, it would be desirable to introduce chemical buffering materials, within the backfilling materials placed adjacent to the waste units, so as to control oxidation/reduction reactions and to remove aggressive species from infiltrating groundwaters before contact with the waste units.

Paradoxically, it appears that the most suitable chemical buffering materials are likely to be metallic compounds. Bird (25) has evaluated a number of possible buffer materials for use in conjunction with high integrity waste unit claddings, including compounds of copper, lead and iron.
He suggests that finely divided copper, copper oxides and sulphides may be useful as reducing agents since they are capable of reacting with aggressive anions in solution (e.g. Cl, $O^{2-}$), to form compounds of very low solubility. Relatively low pH values would favour redox buffering reactions involving copper, but high pH values are required to maximise the stability of copper as a canister material. High chloride and/or carbonate content in the invading groundwater would increase the solubility of copper and copper compounds. A careful evaluation of stability fields for redox reactions under site specific groundwater conditions would therefore be required in each case. From a backfill design viewpoint, experimental data would be required to evaluate the optimum proportion of copper compounds and the specific surface (grading) of the particles required to achieve a given measure of chemical protection.

Lead, introduced into the backfill in finely divided particulate form, could buffer the ambient pH to alkaline conditions (pH approximately 10), and PbO (massicot and litharge) could react with a number of anions other than oxygen over a wide range of concentrations. Values as high as 95% of anions removed from solutions, with concentrations as low as $10^{-12}$ molar, have been observed under controlled laboratory conditions (25). Lead in inorganic compounds is not very mobile in natural groundwaters, and the products of buffering reactions may be expected to remain fairly close to the place of formation. Many compounds of lead may be suitable buffering materials, and further quantitative experimental data would be valuable. However, PbS (galena) would be unlikely to provide significant protection in view of its narrow Eh/pH stability field; see figure 105.

The iron phosphate mineral, vivianite ($Fe_3(PO_4)_2 \cdot 8H_2O$)
has also been considered as an effective oxygen buffer (25), and has been considered as a possible additive to bentonite by Swedish workers. The investigation of methods of oxygen removal to compensate for the introduction of oxygen within the backfill itself has been identified as an important area of research (150).

Notwithstanding the desirability of providing high corrosion resistance in high-level waste unit claddings, and the introduction of chemical buffers within the backfill, the influence of temperature should not be forgotten. Metals which produce passive conditions at low or moderate temperatures, may experience corrosion at higher temperatures, as the oxidising power of the solution is increased. The effects of adjusting the storage period, overpack thickness and emplacement configuration on the value and duration of maximum induced temperatures should also be considered, and may play an equally important role in increasing the corrosion resistance of high-level waste units; see Chapter 15.

A much wider range of packaging materials has been considered for intermediate-level wastes. The principal materials envisaged are:

- steel; as a cladding or container material for immobilising materials such as concrete or bitumen
- cement and concrete; as an immobilising or containment medium
- bitumen; as an immobilising medium within an external cladding
- plastics; as an immobilising medium with or without
Except where steel claddings are used to provide fully-shielded waste units, only a nominal steel thickness is likely to be used in waste cladding. Most current proposals envisage the use of mild carbon steels as opposed to stainless steels.

General corrosion rates in mild carbon steels are high in corrosive groundwaters, and, in the absence of chemical buffers, maximum rates of corrosion after 12 years exposure may be as high as 0.2 to 0.4 mm per year (157); although lower rates may be anticipated in prerepresentative repository environments. Requirements for chemical buffering within surrounding backfills are likely to be similar to those outlined for high-level waste units. However, the design of the fill system must be modified to take account of the stability fields associated with differences in steel composition and ambient temperatures.

Low chloride and sulphate contents are necessary for the preservation of cement paste and concrete in the aqueous environment. Granitic groundwaters may contain concentrations as low as 40 ppm for either of these species; see table 7. However, all groundwaters are site-specific, and chemical buffering is likely to be desirable in all repository situations. Metals and metallic compounds again appear to offer the greatest potential for use as chemical buffers in near-field backfill mixtures.

Groundwaters which contain low concentrations of dissolved silica (SiO₂) and potassium (K⁺) may leach these species from the cement matrix. The presence of zeolites, in particulate form, or clays containing a
A proportion of unweathered potassium feldspar could allow sufficient increase in groundwater concentrations of $\text{SiO}_2$ and $\text{K}^+$ to eliminate the leaching problem.

Bitumen is generally stable in most groundwater systems, although properties vary according to the precise formulation of the material. The main compound which dissolves bitumen is carbon disulphide ($\text{CS}_2$), but this does not occur in natural groundwaters. However, certain microbes can attack bitumen, if present within or on the surface of the material. For this reason bitumens used for waste immobilisation should be sterile and should be clad in air-tight containers.

Radiolysis can also cause deterioration, since gaseous products generated within the material can cause expansion, accompanied by pore formation. Some bitumens are more prone than others in this respect, and the waste unit design must incorporate the optimum bitumen formulation and waste concentration to avoid deleterious effects.

The wide range of plastics which have been considered for use in the immobilisation of intermediate-level wastes are difficult to assess. Plastics are considered primarily due to their impermeability to water (hydrophobic qualities). All comprise complex organic molecules which are found to be extremely resistant to chemical attack in laboratory-controlled experiments. However, their long-term geochemical stability must be questioned and the range of potential reaction fields have not been identified. The geochemical properties of all plastics must be researched in greater detail before backfill buffering requirements can be properly evaluated. Susceptibility to radiation damage is an area of considerable concern. It is well known that
long-term exposure to ultra-violet light can cause degradation of many plastics. Similar effects may be produced due to radiation.

Overall, the problem of assessing the optimum waste unit cladding material and chemical buffering system is seen to be a highly complex one, meriting careful and systematic research. At present, much of the available experimental data appears to be based on aqueous solutions which are not representative of typical host rock groundwaters. Much uncertainty in prediction therefore remains and, in the author's view, further research on the basis of Eh/pH stability fields would be justified in order to develop corrosion models which can provide an effective contribution towards the design of an engineered barrier system.

18.4 Leaching Processes

The onset of the waste leaching process follows immediately from the breach of waste unit claddings due to corrosion and, for crystalline and argillaceous formations, the rate of removal of radionuclides from the waste immobilisation matrix into the groundwater may be regarded as the source term for subsequent migration through the backfill system into the host rock itself.

However, the development of realistic radionuclide release models represents a formidable problem, since leach rates are likely to be influenced by:

- chemical composition of the waste immobilising medium
- temperature variations
- the chemical composition of the backfill and the
The leach rate of borosilicate glass is known to increase considerably under extreme conditions of pH, and in the presence of certain aggressive chemical species. In addition, assuming congruent leaching of the glass and the radionuclides, the leach rate may be controlled by the concentration of dissolved silica.
within the groundwater. Thus, the addition of silica-rich additives in the near-field backfill (i.e. quartz sand or rock flour), together with suitable chemical buffers, could have significant effect in reducing leach rates.

The rate of groundwater flow may also have a significant effect upon leach rates. A rapid rate of groundwater replenishment would reduce near-field concentration levels within the leachate, increasing the rate of waste dissolution. However, slow-moving, semi-stagnant groundwater conditions could inhibit the leaching process considerably (40).

The assumption of congruent leaching implies that the process is essentially a surface dissolution phenomenon, in which the leach rate may be expressed in units of mass removed per unit area per unit time. This suggests that the time required to dissolve a given mass of material is dependent both on its size and its characteristic dimensions. Thus, the release model for the backfilling and sealing system must include terms which account for the geometrical characteristics of the waste units.

The assumptions made about the leaching process may also have a fundamental impact on backfill property requirements. A process involving congruent dissolution of the immobilising medium and the radionuclides implies that the volume of each unit of immobilised waste will diminish as a function of time. Voids would therefore be created within the fill during the leaching process, and any collapse of the surrounding fill would obviously alter the flow system. The corresponding increase in the ratio of groundwater : backfill (i.e. backfill moisture content) would also modify the chemical
buffering capacity and sorption capacity of the system. Thus, if the postulated leaching mechanisms are accepted, it would appear that backfill materials must be designed so that they can swell to fill the voids which develop as the wastes dissolve. In this case, the ratio of the volumes occupied by the waste units and the backfill, at the time of placement, must be pre-determined in relation to the available volumetric swelling capacity of the fill. Alternatively, the fill must be designed as a rigid, self-supporting medium which does not rely upon the interactive support provided by the waste units. The strength of the fill then becomes an important design parameter, which would become particularly significant in designing backfill systems for the in-room placement of intermediate-level wastes. It is therefore fundamentally important to establish the relevance of the surface dissolution models currently described in the available literature.

In practice, it is thought unrealistic to extend the concept of congruent leaching of the waste and the immobilising matrix to predict the formation of a void, since reaction products would undoubtedly be substituted. However, it is apparent that the more basic scientific research is required to establish the true nature of the leaching process for the wastes under consideration. Studies must also investigate the effects of groundwater chemistry and flow conditions upon measured leach rates. Present assessments are normally based on a combination of the following laboratory test methods; using borosilicate glass as the reference material (40):

- 'Soxhlet' leaching experiments, in which the sample is in contact with continuously replenished
distilled water.

- Leaching experiments involving air-lift re-circulation of distilled water at 250 ml/min, with periodic complete changing of the water.

- Long term static leach tests.

In combination, the data from these experiments provide a wide range of leach rates, which are found to vary with glass composition and temperature conditions. However, as previously noted, little or no data is available for intermediate-level waste immobilising materials.

In view of potential influence of several physical and chemical factors, as described, it is likely that the assumption of a constant leach rate, based on congruent dissolution, is a gross over-simplification. The combination of a theoretical approach with systematic empirical tests, could provide a more satisfactory means of assessment, including provision for time-dependent factors in the analysis of the leaching process.

In view of these difficulties, and to simplify the present discussion, only two parametric influences will be considered in further detail. These are:

- Leach rate variations
- Waste unit size and shape.

Laboratory leach tests on borosilicate glass samples in distilled water have generally provided leach rate values in the range $10^{-7}$ to $10^{-3}$ gm/cm²/day (109). The upper limit corresponds to tests in which the temperature of the water approached $100^\circ$. However, at any given temperature, leach rates are found to vary
according to the actual composition of the glass and the temperature of its formation (during the vitrification process).

As shown in figure 22a, anticipated geothermal temperatures at the repository construction depths under consideration lie in the range 17 to 50°C. The corresponding range of leach rates, predicted on the basis of laboratory tests, is approximately $5 \times 10^{-6}$ to $2 \times 10^{-4}$ gm/cm²/day, as shown in figure 22b.

The only available long-term insitu experimental data have been obtained from nepheline syenite glass blocks, buried just beneath the water table at Chalk River Nuclear Laboratories in Canada (84). These tests indicated a mean leach rate of $5 \times 10^{-11}$ gm/cm²/day, based on a 15 year test period. Assuming a mean ambient temperature of 50°C, and correcting for the effects of temperature increase at depth, this represents a leach rate of nearly 5 orders of magnitude less than the laboratory-based data.

Current methods of assessing release rates generally assume a simple congruent dissolution model in which the leach rate is constant (100). The release of rate $R$ is then defined by:

$$ R = C \times A(t) \times l $$ \hspace{1cm} (1)

where

- $A$ is the exposed surface area of the waste at a particular time $t$.
- $l$ is the leach rate (assumed constant).
- $C$ is the concentration of a particular radionuclide within the waste matrix.

The release rate $R$ varies with time, because the area
available for dissolution, \( A \), decreases as the leaching process continues. Applying this principle to a spherical waste fragment, equation 1 becomes:

\[
R = \frac{C\pi l(d - 2t_1)^2}{\rho} \tag{2}
\]

where \( d \) is the initial diameter
\( t \) is the time of exposure
\( l \) is the leach rate
\( \rho \) is the density of the waste matrix.

Thus, the dissolution model predicts that the release rate is a quadratic function of time and the diameter of the waste fragment; and a cubic function of the leach rate. However, the time required for complete dissolution of the waste is directly proportional to the diameter and inversely proportional to the leach rate, thus:

\[
t = \frac{d \rho}{2l} \tag{3}
\]

Thus, assuming a spherical waste fragment of 10cm diameter and a waste matrix density of 2.6 g/cm\(^2\), the laboratory-based data predicts dissolution times in the range 180 - 7120 years. However, the insitu test data from Chalk River suggests a dissolution time of the order of 3500M years allowing for an ambient repository temperature of about 30°C.

These simple assessments again illustrate the need for more detailed investigation of leach rates. Experiments which allow for varying flow rates and representative groundwater compositions could show that the times for complete dissolution could be several orders of
magnitude greater than those currently forecast on the basis of laboratory experiments. Radionuclide release rates would be correspondingly reduced. The delay and dilution effects brought about could be highly significant in the containment of long-lived radionuclides which tend to be poorly sorbed during groundwater transport; see 18.5 below.

It is generally assumed that the area term, A, in equation 1 is the only time-dependent variable. In practice, however, the release rate is likely to be governed by a diffusion process, depending upon concentration gradients through the waste matrix itself. Thus, the concentration term, C, is unlikely to remain constant. In addition, the leach rate, l, will vary according to the concentration levels and solubility saturation levels of different radionuclides within the groundwater. These factors in turn depend on the groundwater composition and the groundwater residence time, \( S \), which is defined by:

\[
S = \frac{n}{ki} \tag{4}
\]

where

- \( n \) is the effective porosity of the backfill
- \( k \) is the backfill permeability
- \( i \) is the hydraulic gradient

Assuming that congruent dissolution dominates, the parameters \( n \) and \( k \) will also be time-dependent. However, if the rate of diffusion of radionuclides although the matrix is faster than the rate of dissolution, then time-dependent variation of the hydraulic properties of the backfill need not be considered.
These various factors clearly require more thorough investigation than hitherto (40). It appears that present theoretical models and experimental methods have been over-simplified. A modified dissolution model, which incorporates a time-dependent empirical leach rate term, would provide a more reasonable basis for estimates of release rates and dissolution times. However, as previously suggested, any form of dissolution model could have serious adverse implications in terms of the long-term hydraulic properties of the near-field backfill system if voids are created by the leaching process.

Current conceptual design proposals generally envisage the use of cylindrical units for high-level wastes and cylindrical or cuboidal-shaped units for intermediate-level wastes. However, equation 1 suggests that the leach rate is sensitive to the geometrical design of the waste units. It is therefore appropriate to examine the factors involved. Although the validity of the dissolution model has been questioned, it is retained here as a basis for outline assessments.

Three basic geometrical shapes are considered, namely:

- the sphere
- the cylinder
- the cuboid

Each of these shapes may be represented by a characteristic dimension. For spheres and cylinders, the chosen characteristic dimension is the diameter, d; and for cuboids, it is the breadth, b. For cylinders and cuboids, it is also necessary to specify a parameter k; which is the ratio of the height to the diameter or breadth of the solid.
For each of these shapes, the author has derived release rate expressions from equation 1. Ignoring the concentration coefficient, C, the expressions are:

**Sphere**

\[ R = \pi l(d - 2t/l)^2 \] ............................... (5)

**Cylinder**

\[ R = \pi l(d - 2t/l)^2 \left( \frac{1}{d} + kd - 2t/l \right) \] ............................... (6)

**Cuboid**

\[ R = 2l(b - 2t/l)^2 \] ............................... (7)

The above expressions show how the release rates vary with the geometrical parameters \( d, b \) and \( k \). However, where \( k > 1 \), the time required for complete release/dissolution, \( T \), is independent of \( k \). The appropriate expressions then are:

**Sphere**

\[ T = \frac{d\rho}{2l} \] ............................... (8)

**Cylinder**

\[ T = \frac{d\rho}{2l} \] ............................... (9)
\[ T = \frac{\rho}{21} \] (10)

i.e. the time required for complete release is directly proportional to the smallest dimension, and the constant of proportionality is \( \frac{\rho}{21} \).

For this reason, all radionuclide migration studies have utilised the sphere as a simple reference shape for the assessment of the time required to complete the release/dissolution process. It is also generally assumed that the glass matrix in high-level waste units will be fractured as a result of handling or insitu de-vitrification, so that the material is broken-down into fist-sized lumps at the time of emplacement (100). In this event, the release times would be more or less as predicted earlier for the 10cm diameter spheres. However, if sufficient precautions are taken during the design, manufacture and handling of the units, it should be possible to ensure that the waste matrix remains intact, and retains its initial geometrical form at the time of emplacement. If the indications are that borosilicate glass will inevitably break down into discrete lumps, due to de-vitrification in the long-term, it would appear that there are strong technical arguments in favour of the section of a more stable form of material, so that release times could be extended over much longer periods.

Figure 106a shows the times required for complete dissolution (i.e. total release of radionuclides) for waste units of different sizes and shapes, based on equations 8, 9 and 10; assuming a constant leach rate of \( 10^{-6} \text{ gm/cm}^2/\text{day} \). It is apparent that the preferred
shape is the sphere, followed by the cylinder and then the cuboid. It is also evident that any elongation (i.e. increasing the value of k) reduces the release time for a waste unit of given shape and volume. However, irrespective of shape, the release time increases linearly with the minimum dimension of the waste unit. Clearly, therefore, large units are to be preferred, from the present viewpoint, although in practice the maximum size will also depend upon the size of the repository excavations, the volume of backfill required and the design of the waste emplacement/backfill configuration.

However, it is also interesting to note the variation of release rate with time, for a particular shape of unit. Figure 106b shows the release rate/time relationships for the same range of shapes, based on equations 5, 6 and 7; assuming a waste volume of 1 m³. It is apparent that the cuboidal shape promotes a rapid initial leach rate by comparison with cylinders and spheres. The graph indicates the actual quantity of waste material released as a function of time, and may obviously be multiplied by the number of waste units in a given repository zone to obtain the total release quantity over a specified period.

The relevance of release rate/waste unit shape relationships to the design of engineered barriers is self-evident. Based on current assumptions concerning the mechanism of the leaching process, it is apparent that relatively large diameter cylindrical units are to be preferred for high-level waste disposal; a conclusion which coincides with the results of the optimisation study described in Chapter 15. For intermediate-level waste units, where in-room stacking arrangements are envisaged, shapes approximating to
spheres provide the best resistance to leaching; suggesting that a review of current proposals for the adoption of cuboidal or cylindrical shapes for intermediate-level waste packaging may be warranted.

In general, waste units with large volumes and low aspect ratios provide the most effective long-term containment. However, ease of fabrication and the practical aspects of waste emplacement and backfilling must be considered; and overpack and shielding costs are also relevant. Further consideration of these factors is included in Chapter 19.

18.5 Migration Phase

Following the release of radionuclides, due to the leaching action of groundwater, their migration through the backfilling and sealing system could occur in two ways, namely:

(a) transverse migration; in which there is a significant component normal to the line of the excavations

(b) longitudinal migration; in which the direction of movement is predominantly parallel to the line of the excavations.

The parameters which influence longitudinal migration determine the extent to which the repository system could provide 'short-circuit' connections to the biosphere; avoiding the relatively slow process of migration through the host rock. Therefore, excessive longitudinal migration is undesirable, although it could be significant in both active and redundant parts of the repository system.
The actual migration route through the backfill may be regarded as the vector sum of the transverse and longitudinal components. In optimising the parameters involved, the objective is to maximise the travel-time of radionuclides passing through the backfilling and sealing system, without reducing the length of the subsequent migration path through the host rock.

Groundwater transport is promoted by the hydraulic gradient established after saturation of the system. In the absence of physico-chemical retention effects, the rate of radionuclide migration will therefore be governed solely by the release rate, the permeability and porosity of the fill materials, and the hydraulic boundary conditions. However, the retention properties of backfills may exert a significant retarding influence on certain radionuclides, so that their travel times are much longer than those of the groundwater.

The relative significance of hydraulic and physico-chemical retention properties of the fill may vary according to the dominant mode of transport, and the flow boundary conditions. The influence of hydraulic properties may be illustrated, at a very simple level, by means of flow nets. The diagrams shown in figure 107 illustrate the simple case of a backfilled circular opening in ground having an isotropic permeability, \( k_g \) under the influence of an ambient hydraulic gradient, \( i \). It is assumed that the backfill is homogenous, with an isotropic permeability, \( k_b \). Where \( k_g = k_b \), the rates of groundwater flow through the ground and the backfilled tunnel are identical; see figure 107a. The lowest rate of flow through the backfill is obtained where \( k_g \gg k_b \), causing the greatest divergence of flow lines; see figure 107b. Analyses usually show that permeability
differences of at least one order of magnitude are necessary to produce a significant influence on the flow system.

However, more sophisticated analyses are required to examine parametric influences under more realistic situations. These would involve:

- layered backfill systems
- anisotropic permeabilities
- different orientations of the excavations with respect to the principal hydraulic flow direction
- 'fractured' as well as 'porous' ground
- non-uniform boundary conditions

Meaningful studies of this type could only be carried out by computer methods.

Figure 107d illustrates the ideal flow situation for a totally impermeable backfill. Impermeable barriers were discussed in Chapter 17 as a means of extending the delay to backfill saturation; although it was noted that, in practice, saturation may ultimately occur due to imperfections in the impermeable barrier; see figure 100a.

Under these circumstances, the backfilling and sealing system would create an internal three-dimensional flow system, as illustrated in figure 108a. Modelling the effects of varying the density and distribution of imperfections could provide a useful indication of the long-term effectiveness of 'impermeable' barriers within
the backfilling and sealing system. For backfill systems which incorporate permanent linings, the spacing of circumferential and longitudinal joints, and the positions of grout holes, bolt holes, etc., could also be examined in order to determine the influence of lining design on transverse groundwater movement.

No firm design conclusions regarding these factors are put forward in this thesis. However, it is thought likely that the use of impermeable barriers could have a significant influence on the effectiveness of engineered backfill systems, and there is likely to be considerable scope for optimising relevant backfill design parameters, based on the use of currently available computer techniques.

When flow occurs through the backfill within an in-room disposal system, the configuration of the units will also influence the flow pattern and flow rates. In general, a waste emplacement system which incorporates tortuous flow paths would be preferred. Further discussion of appropriate waste unit shape and stacking arrangements is given in Chapter 19.

Radionuclide retention properties are also likely to have a significant influence. However, as described in Chapter 9.2, the lack of a satisfactory data-base precludes the use of truly representative retention parameters at the present stage. Ideally, it would be desirable to distinguish the retention effects attributable to molecular diffusion, sorption and chemical retention, in conditions similar to those anticipated in the repository system. Unfortunately, the $K_d$ concept currently provides the only basis for empirical assessments.
The distribution coefficient $K_d$ is defined as:

$$K_d = \frac{\text{mass of sorbed radionuclide R/gm of backfill}}{\text{mass of dissolved radionuclide R/cm}^3 \text{ of solution}} \quad \ldots (1)$$

It has previously been noted that values of $K_d$ for particular radionuclides are strictly applicable only to sorption/de-sorption reactions occurring in solutions of low concentration, at the Eh/pH and temperature/pressure conditions which prevailed during the original experiments. Nevertheless, in the absence of any satisfactory alternative approach, it must be assumed for present purposes that experimental $K_d$ values are representative for repository conditions, and take full account of the various mechanisms which could operate in practice.

Where retention effects occur, the travel time for a radionuclide, $R$, is greater than that of the groundwater by an amount:

$$\frac{T_R}{T_W} = 1 + K_d \frac{\gamma}{n} \quad \ldots (2)$$

Where $T_R$ is the travel time of the radionuclide
$T_W$ is the travel time of the groundwater
$\gamma$ is the density of the backfill
$n$ is the porosity of the backfill

For a given system, the values $K_d$, $\gamma$ and $n$ are assumed to be constant. Hence, it is convenient to define another empirical sorption constant, $K$, (100) where:

$$K = 1 + K_d \frac{\gamma}{n} \quad \ldots (3)$$
Thus,
\[ \frac{T_R}{T_W} = K \quad \text{............................. (4)} \]
and
\[ \frac{V_W}{V_R} = K \quad \text{............................. (5)} \]

Where \( V_W \) is the velocity of the groundwater
\( V_R \) is the velocity of the radionuclide

Based on Darcy's law, the true mean velocity of the groundwater is given by:
\[ V_W = \frac{ki}{n} \quad \text{............................. (6)} \]

Where \( k \) is the permeability
\( i \) is the hydraulic gradient

Hence for a flow path of length \( z \):
\[ T_W = \frac{zn}{ki} \quad \text{............................. (7)} \]
and
\[ T_R = \frac{zKn}{ki} \quad \text{............................. (8)} \]

It is apparent from equations 3 and 8, that the factors which increase the transverse radionuclide migration
time within the backfill are:

(a) a high density
(b) a low effective porosity
(c) a low permeability
(d) a small hydraulic gradient
(e) a long flow path
(f) a high $K_d$ value

The physical and hydraulic parameters (a), (b) and (c) are inter-related, and may readily be assessed for a given fill material. By reference to figure 108a, it is also apparent that the use of impermeable membranes or barriers can greatly increase the length of the flow path and reduce the hydraulic gradient; whereas, in the absence of an impermeable barrier, the general direction of flow would coincide with the direction of the regional hydraulic gradient.

It is clearly important that the retention of backfill materials should be designed to complement those of the host rock. In addition, it is desirable that the retention properties of the fill should ensure a significant retardation of all radionuclides which present a significant potential health hazard. No single backfill material or additive is likely to satisfy these requirements. Therefore, it is likely that the required retention properties must be obtained by introducing complex sorptive mixtures.

However, it is likely that the improvement in backfill retention properties achieved by the introduction of these additives could also cause an increase in backfill permeability. In addition (recognising that difficulties may occur in achieving adequate fill compaction in certain regions) it may be desirable to
counter potential deficiencies in hydraulic properties by increasing retention capacity. Thus, the design of engineered fill barriers must consider the optimisation of retention properties and hydraulic performance.

A range of travel times for transverse migration through an engineered fill barrier is plotted on figure 109. The analysis is based on equation 8 and assumes that the ratio $zn/i = 100$. The travel time isochrones show the combined influence of variations in fill permeability and retention properties on the containment of radionuclides. The ratio $zn/i = 100$ is arbitrarily chosen and could, for example, represent a situation in which:

- flow path length $z = 4.0m$
- backfill porosity $n = 0.25$
- hydraulic gradient $i = 0.01$

Clearly, suites of many curves would have to be generated to represent the full range of backfill properties and hydraulic boundary conditions which could be envisaged in practice. The modelling process is thus ideally suited to a computer-based analysis, which would form a logical extension of the studies previously described.

In figure 109, another series of lines is shown running parallel to the travel-time isochrones. These represent the effective toxic lives of a number of radionuclides selected from the inventory shown in table 3. For each radionuclide, the time period shown is equivalent to 30 half-lives; at which stage the activity will have decayed to $10^{-9}$ times its original level, which is assumed to be negligible for present purposes; see figure 4. The lines therefore indicate the ranges of $K$ and $k$ which would ensure total containment of the wastes under the conditions assumed in the example.

As previously noted, the lack of representative data concerning values of the retention parameter, $K$, is the
main problem in modelling the real repository situation. However, Neretnieks has quoted some insitu $K$ values, based on radionuclide migration measurements in western U.S. desert sub-soils (151). Relevant values have therefore been plotted on the figure for illustrative purposes. The values shown support the generally accepted conclusion that the radionuclides $^{129}$I, $^{99}$Tc and $^{123}$Cs are not significantly retarded by natural soils. Based on the retention data shown, a fill permeability of about $2 \times 10^{-14}$ m/sec would be required to contain $^{129}$I for the conditions assumed. For higher fill permeabilities, an additive would be required having a high sorptive affinity for iodine.

It must be noted that the value of the insitu sorption constant $K$, is influenced by the ratio $\gamma/n$. For backfills, the value of this parameter depends upon the method of placement; and for a given material, is likely to be empirically related to the permeability $k$. Hence, it may be desirable to examine the influence of more readily measured physical properties, such as $\gamma$ and $n$, and to relate these to laboratory-based measurements of $K_d$. Parametric studies based on measurements of these relatively simple properties, using systematic and standardised testing procedures for a wide range of materials, could provide a valuable indication of requirements for materials formulation studies and placement trials.

As described earlier, it may be desirable to promote a substantial longitudinal component during transverse migration within the active areas of the repository system. This may be most readily achieved by modifying the flow boundary conditions, using impermeable barriers, as illustrated in figure 108a.
However, under certain circumstances, longitudinal migration could provide a weak link in the system. This applies to situations where longitudinal flow paths allow the migration of radionuclides from active to redundant areas of the repository, and also provide a direct route for continued migration towards the biosphere.

The problem of longitudinal migration has been discussed in Chapters 9 and 10, and the potential flow zones to be considered are illustrated in figure 34, comprising:

1. the bulk fill within the central region of the excavations
2. the contact zone around the periphery of the excavations (including a lining, where relevant)
3. the zone of peripheral disturbance induced by over-stress; due to loss of internal support pressure or inadequate control over methods of excavation.

In order to establish the conditions under which a preferential flow path could develop, it is convenient to consider the properties of the bulk fill (zone 1) as essentially constant. The sensitivity of the following parameters may then be investigated:

- the amount of separation at the contact between the bulk fill and the host rock or lining (zone 2)
- the effect of peripheral ground disturbances (zone 3)
As an illustrative example, consider the case of a circular excavation containing a bulk fill of mass permeability $10^{-12}$ m/sec. This low value would be difficult, though not impossible, to achieve in practice; and lies within the design range of permeabilities envisaged for the repository situation.

Assuming a uniform hydraulic gradient, $i$, in the longitudinal direction, the volumetric flow rates through the bulk fill and the disturbed zone (zones 1 and 3) are both predicted by the Darcy relationship:

$$Q = kAi$$ ........................................ (9)

Assuming a continuous plane of separation at the peripheral contact (zone 2), the interface flow must be predicted on the basis of a fissure flow analogy. Using the expression for fissure flow derived in Chapter 9.1, the flow along the separation plane is given by:

$$Q_s = \frac{b^2 \rho g D}{12\eta}$$ ........................................ (10)

Where

- $D$ is the diameter of the excavation
- $b$ is the thickness of the plane of separation

Figure 110a shows the effects of increasing the separation thickness, $b$, for a range of excavation diameters. It is apparent that for a given separation thickness, the interface flow becomes increasingly pronounced as the diameter of the excavation decreases. For a given diameter, a small increase in the separation distance produces a large increase in preferential flow. Due to the cubic form of the relationship in equation 10, it implies that a separation of only a few microns
can give rise to a disproportionately high flow along the interface.

The example illustrated assumes a continuous separation along the periphery of the opening. In tunnel situations this is unlikely, since separation planes would tend to develop mainly in the crown and shoulder areas. For a given mean separation thickness, $b$, a pro-rata adjustment can be made to the ratio $Q_s/Q_f$, according to the proportion of the circumference in which separation has occurred. This type of modelling may therefore be used to assess the effects of fill shrinkage, consolidation, etc.

Figure 110b illustrates the influence of the thickness and permeability of the disturbed zone on longitudinal flow, based on equation 9. Where the peripheral ground disturbance is attributable to stress re-distribution, the disturbed zone thickness, $t$, depends upon the diameter of the excavation; see Chapter 10.2. Hence, it is appropriate to express variations in the thickness of the disturbed zone, $t$, in terms of the dimensionless parameter $t/D$. The volumetric flow rates through the disturbed zone and the bulk fill are also expressed as a dimensionless ratio $Q_d/Q_f$. Parametric relationships are indicated for a range of disturbed zone permeabilities.

The diagram indicates that the thickness of the disturbed zone and the relative increase in permeability are highly significant, and it is apparent that a wide range of anomalous flow conditions could arise. It must be noted that the higher values of the ratio $kd/kf$ are likely to occur in relatively brittle rocks, with low rock competence factors. The development of fractures in the disturbed zone implies a large increase in permeability, and thus the provision of rigid internal
support systems would assist in minimising longitudinal flow anomalies in host rocks of this type.

The simple analysis described indicates the potentially adverse effects due to differences in the hydraulic properties of the three longitudinal flow zones identified. However, it is important to note that the rate of release of radionuclides into the three flow zones may be different in each case, and will vary as a function of time. Furthermore, to satisfy the requirement for continuity of flow, there may be significant dilution of radionuclide concentration levels in the peripheral regions. Further consideration of these factors is beyond the scope of this thesis, but again indicates a need for more detailed optimisation studies.

However, the analysis of volumetric groundwater flow rates does not provide the complete picture. Groundwater velocities must also be evaluated in order to assess the travel times for the radionuclides of concern. The plane of separation acts as a single fissure, and therefore the assessment is relatively straightforward. Presumably, an assessment of the increase in fracture porosity of the disturbed zone could also be made by predicting the amount of dilatancy associated with localised brittle failure of the rock mass.

An outline assessment of combined flow rates/travel times has been made for a 100m length of circular repository opening, based on the following parameters:

- diameter of opening $D = 5\text{m}$
- disturbed zone thickness $t = 0.5\text{m}$ (i.e. $t/D = 0.1$)
- longitudinal hydraulic gradient $i = 0.001$
bulk fill permeability \( k_f = 10^{-12} \) to \( 10^{-8} \) m/sec
disturbed zone permeability \( k_d = 10^{-7} \) m/sec
disturbed zone

surface separation \( b = 5 \) to \( 20 \mu \)
effective porosity of bulk fill \( n = 0.20 \)

\( \rho_g \) for water \( \eta = 1.78 \times 10^7 \) (m.sec\(^{-1}\))

The results of the analysis for this illustrative example are presented in figure 111. The diagram indicates the cumulative volume of contaminated groundwater passing the end of the 100m flow path, as a function of time, for each flow zone. The equations used to assess volumetric flow rates and velocities have utilised a combined porous medium/fissured medium approach, as shown in table 39. The flow velocity calculation for the disturbed zone assumes a fracture flow model, in which the flow is concentrated in an equivalent single fissure. This is considered to be a pessimistic 'first-approach'. More sophisticated studies should consider the probable increase in fracture porosity due to dilatancy, as previously described.

Figure 111 shows that the first arrival times are predicted within the plane of separation and the disturbed zone, and may occur rapidly after the first release of radionuclides. As noted above, the first arrival times shown for the disturbed zone are probably unduely pessimistic. However, the volumes of water involved are relatively small. The first arrival time for contaminated water in the bulk fill area is many orders of magnitude greater, but much larger volumes are involved.
The example demonstrates the difficulty in providing adequate hydraulic barriers in the peripheral zones of the flow system. It is apparent that further design studies are required to assess the extent to which the repository backfilling and sealing system could provide an escape route for radionuclides.

A number of design expedients should be carefully considered in future studies. Cut-off collars, as illustrated in figure 98, could be effective in reducing peripheral flow, by interrupting the flow path. The constriction of flow would tend to force contaminated water into the surrounding rock or the bulk fill, incorporating delay and dilution factors into the transport model.

However, the geometry and permeability characteristics of cut-off collar designs is likely to be critical. For example, a simple low-permeability plug, which fills the opening but does not fully penetrate the disturbed zone, could actually encourage peripheral flow. For fully penetrating cut-off collars, the diameter, thickness and permeability of the plug will determine its effectiveness in forcing water into a transverse flow path. Parametric studies of this type must therefore consider a large range of design variables and may best be performed using analogue or computer methods.

Purely hydraulic studies of this type could provide a valuable insight into the backfill design requirements for crystalline and argillaceous host rock repositories. It must also be noted that the analysis of longitudinal groundwater transport provides the basis for deriving a delay model for saliferous rocks.
For crystalline and argillaceous host rocks, studies must also examine the scope for countering deficiencies in hydraulic performance, by means of retentive backfills. For the example shown in figure 111, it would be desirable to assess the scope for reducing the first arrival times for radionuclide migration in the peripheral zones.

Consider the separation plane, whose thickness has been assumed to vary from 5 to 20μ. As shown in table 39, the fissure flow equation predicts that the velocity of water \( v_w \) is given by:

\[
\frac{b g}{12 \eta} \text{ ........................................ (11)}
\]

As described in Chapter 9.2, the travel time of a particular radionuclide, \( T_R \), and the travel time of the water, \( T_W \), along a fissure are related by the expression:

\[
\frac{T_R}{T_W} = K \text{ ........................................ (12)}
\]

where the sorption-equilibrium constant, \( K \), for the fissure flow system is defined by:

\[
K = 1 + K_{df} M_s \text{ ........................................ (13)}
\]

where \( K_{df} \) is the sorption distribution coefficient for the fissure system

\( M_s \) is the specific surface of the fissure

\( \text{ratio of surface area to volume} \)

Utilising equations 11 and 12, it is possible to assess the backfill sorption properties required to achieve a
given migration time along the separation plane for the 100m flow length used in the previous flow analysis. Results are shown in figure 112.

The separation thickness, $b$, is plotted on a linear scale, and the curved form of the travel time isochrones reflects the quadratic form of the velocity expression in equation 3.

The results may be compared with the transverse migration example shown in figure 109. It must be noted, however, that the sorption equilibrium constants are not defined in the same way. The present example involves retention in a fissure system, and involves the parameter $K_{df}$, as shown in equation 13. Unfortunately, no empirical data are available to indicate the corresponding ranges of K. This is clearly an area which merits practical experimental research.
19. PRACTICAL DESIGN AND OPERATIONAL ASPECTS

19.1 Introduction

Previous chapters have considered various fundamental engineering aspects of repository design and construction, with emphasis on waste containment and the provision of engineered barriers. The interaction of the principal containment barriers within repository systems, namely the host rock, the waste units and the backfill have been emphasised repeatedly. The minimisation of repository size and the maximisation of volumetric efficiency have been advocated as a means of enhancing the containment properties of the system and reducing costs; the emphasis on each varying according to the host rock type.

However, various practical aspects of waste emplacement and backfilling require further engineering consideration. In Chapter 18.4, it has been shown that even small physical imperfections in the backfilling system, particularly at interfaces, can cause a serious impairment of long term performance. Relevant aspects of the waste emplacement and backfilling repository design proposals have been critically examined in Chapter 13, and it is thought that this topic has received insufficient detailed attention to date. In the author's view, it is essential that the design of engineered barriers must also incorporate a methodology for placement which will provide consistent and predictable results, to ensure that the desired objectives are attained throughout the repository system.

In relation to backfilling and sealing, the minimum repository size philosophy will help; since reducing
unnecessary excess in backfill volumes will also reduce the probability of human errors or faulty workmanship. Separation of construction and waste emplacement activities in time (where feasible) will also reduce potential conflict between the very different types of operation involved in underground construction, and the handling and emplacement of radioactive waste units.

Despite the general minimisation of repository size, however, it is necessary to evaluate the volume required for creation of effective engineered backfill barriers. For active regions of crystalline and argillaceous repositories, logic would suggest that the optimum barrier size and distribution would depend upon the volume and distribution of the wastes as well as the backfill properties. This topic is briefly considered in section 19.2 below.

However, operational difficulties must also be evaluated. The need for concurrent waste emplacement and backfilling and sealing in active regions of crystalline and argillaceous rock repositories, brings special problems in reconciling the conflicting needs for strict environmental and radiological controls in handling and emplacing the waste units, and the traditionally robust approach normally associated with underground backfilling and sealing operations.

Sections 19.3 and 19.4 below outline some of the factors which could benefit from a more detailed engineering appraisal than hitherto, and outline some practical design measures and operational procedures which could improve overall safety during the short-term emplacement phase, and increase confidence in predictions of long-term repository performance. Once again, the underlying theme is the need for an integrated approach
to the design of the repository, the waste units, and the backfilling and sealing system, and the adoption of techniques which provide maximum scope for direct engineering control, rather than long-range predictions of complex and uncontrolled interactive phenomena.

19.2 Backfill/Waste Volumes

Chapter 18.4 has demonstrated the importance of geochemical retention as a barrier to radionuclide migration through the near-field repository system. Particularly for the long-lived radionuclides, it is desirable to ensure that the sorptive properties of the backfill complement those of the host rock, so that all of the radionuclides travel at a much slower rate than would be predicted on the basis of groundwater movement alone.

The nature of the various geochemical retention mechanisms described in Chapters 9.2 and 18.5, suggests that a relationship should exist between the volume of waste and the volume of sorptive constituents within the backfill required to achieve a given retention effect. As outlined in Chapter 18.5, there should also be scope for optimising the design of backfill systems in terms of their permeability and geochemical retention properties.

For backfill additives included by virtue of their geochemical retention properties, it is apparent that an upper limit is likely to exist to the quantity of a given radionuclide which can be effectively retained in a given quantity of the additive. Hence, in order to give full effect to the retention properties of the backfill, not only should the backfill volume/waste
volume requirement be established, but the sorptive elements of the backfill must be placed in a position whereby contaminated water will have adequate contact time, whatever the flow path.

For high-level wastes placed in boreholes, with an annular backfill, the minimum flow path length for contaminated water from a given point on a waste unit surface is equal to the thickness of the annulus. However, for a stack of intermediate-level waste units, contaminated water from the outermost units may travel through a smaller backfill barrier thickness than contaminated water emerging from the more central units. The retention capacity of fill placed at the periphery of the excavation may therefore be exceeded as more radionuclides reach it from the centre of the waste stack.

Hence, it appears to the author that intermediate-level waste stacks must include a fixed volume of sorptive backfill around each unit, so that the retention capacity of the peripheral fill barrier may be designed to cater for a comparatively modest level of contamination. It follows also, that to ensure consistent residence times necessary for geochemical retention properties to take effect, the thickness of the fill around each waste unit should be uniform. If this is not the case, then retention will be greatest in pockets of fill where the thickness is greatest, and a more rapid migration will occur via pathways where the separation between units is relatively small. The solution to this problem must lie in the geometrical design of the waste units and the waste stacking system, as discussed in section 19.4 below.

The determination of the optimum waste volume/backfill
volume ratio, for fill of a given composition, is beyond the scope of this thesis; indeed it appears that there is inadequate empirical data, at present, for such estimates to be made. Nevertheless, the determination of such relationships would clearly have an important impact upon repository design.

Consider, for example, the spiral tunnel concept previously advocated by the author for the in-room disposal of high-level and intermediate-level wastes in a plastic clay host formation; see Chapter 16.3. Based on table 18, the volume of intermediate-level waste for disposal is 37,500m$^3$. The use of 4.0m internal diameter tunnels, with centrally-placed high-level waste units, would therefore leave an available volume of 280,000m$^3$ for placement of intermediate-level wastes and backfill. Supposing that a peripheral impermeable fill membrane were provided, comprising 0.15m thick bitumen blocks (see figure 100), then the remaining available volume would be 240,333m$^3$, and would provide for a waste volume/sorptive backfill ratio of about 6.

However, as described in Chapter 16.3, it would be desirable to reduce the amount of tunnel construction required by increasing the diameter of the high-level waste units to 0.425m, and increasing the storage period accordingly, so that the enlarged waste units would produce the same initial heat output. The length of tunnelling would then be halved, and the space available for placement of the intermediate-level waste units and the backfill would be reduced to approximately 140,000m$^3$. Assuming the incorporation of a 0.15m thick impermeable bitumen fill barrier at the periphery, as before, the remaining available volume would be about 120,000m$^3$, implying a waste/backfill volume ratio of about 3. In order to restore this ratio to the original
value of 6, it would then be necessary to increase the internal tunnel diameter to about 4.4m.

The above example illustrates that considerable scope exists for optimising the overall design of repository systems, on the basis of the performance of engineered barriers. Further research to determine the optimum formulation of retentive backfills, and the relationship between fill admixture volumes and waste volumes, is seen as a pre-requisite for a practical evaluation of more detailed design requirements.

19.3 High-Level Waste Emplacement Systems

The international proposals reviewed in Chapter 13 generally envisage that high-level waste units will be lowered into emplacement holes, and the annular space around the units will subsequently be backfilled by a special remote-controlled device for fill delivery and compaction. Several of the proposals envisage the use of pre-batched, dry intimate mixtures of bentonite powder and quartz sand, poured from above and compacted insitu by vibrators. The slow ingress of groundwater is then relied upon to provide a high density fill of suitably low permeability, due to the long-term swelling action of the bentonite.

This swelling action is crucial. It must not only ensure the elimination of imperfections at the rock-backfill and backfill/waste unit interfaces, but must ensure that sufficient swell potential remains to prevent any fissure development in the backfill, due to contraction effects, as waste heat output and local temperatures reduce towards the end of the thermal phase. Moreover, as outlined in 18.4 above, the
backfill must be capable of taking up any void space which may be created due to the waste leaching process, without serious impairment in its containment properties. In view of these factors, reliance upon a remote-controlled fill delivery and compaction procedure within the awkward annular space around the units, especially in deep holes, poses serious doubts concerning the effectiveness and reliability of systems proposed to date.

Some work carried out by Pusch suggest that highly compacted pre-formed solid blocks of 'dry' bentonite could provide a more satisfactory means of filling high-level waste emplacement holes (176). His experiments show that compaction of Na-bentonite powder at 10% moisture content, under statically applied pressures in the range 50 MN/m$^2$ to 90 MN/m$^2$, results in the formation of solid blocks with corresponding bulk densities in the range 2.0 to 2.3 t/m$^3$, and values of degree of saturation in the range 60% to 78%. The unconfined compressive strength of a block compacted at 50 MN/m$^2$ is found to be about 10 MN/m$^2$, with a deformation modulus of about 300 MN/m$^2$ (values which are representative of a weak sedimentary rock; see figure 24).

Pusch's laboratory experiments indicate that clear empirical relationships exist between block density and swelling pressure, permeability and shear strength, where the blocks are allowed access to water under confined conditions. For example, a block having an initial bulk density of 2.0 t/m$^3$, when allowed access to water, without increase in volume, attains a bulk density of about 2.2 t/m$^2$; and its permeability is reduced to nearly $10^{-14}$ m/sec (170).
Pusch has therefore suggested that pre-formed annular-shaped blocks of this type could be used in the backfilling of high-level waste emplacement holes (172). He advocates the placement of 'under-sized' annular blocks within the hole, prior to insertion of the waste unit. The waste unit is then placed in the central void, leaving a 3cm gap between the waste unit and the inside circumference of the fill blocks, and a similar gap is left between the outer circumference of the fill blocks and the borehole walls. Dry bentonite powder is then placed to fill the two small annular gaps, with delivery and compaction by means of a remote controlled ring-shaped device, as in other concepts (176).

The above has the advantage that direct control may be exercised, ensuring quality control to laboratory standards, and maximum compaction of the majority of the fill. Experiments have shown that the 'shelf-life' of the pre-formed blocks is likely to be measurable in weeks, before absorption of atmospheric moisture initiates cracking due to swelling processes (176). Presumably, some form of vacuum packaging, with an air-tight membrane, could bring about substantial improvements.

However, the author tentatively suggests a practical refinement of Pusch's pre-formed block concept, which could provide a more effective means of waste emplacement and fill compaction. The objective of the proposal is to eliminate the need for annular spaces which require in situ filling and compaction to a necessarily lower standard, and to ensure maximum confinement of the pre-formed blocks, with consequent improvements in predicted performance.

The author's proposal involves the fabrication of
pre-formed annular fill blocks, in which the outer circumference is only marginally smaller than the hole diameter; the actual tolerance being the maximum necessary for insertion of the fill blocks. The inner circumference, however, would be smaller than that of the waste units, so that insertion of the waste units would require an insitu lateral displacement of the fill material.

In order to ensure lateral displacement (compaction) of the fill during insertion, it would be necessary to form a special profile at the 'nose' of the waste unit, such that simple shear failure of the fill material would not occur. If this could be achieved, insertion of the waste unit would create a lateral displacement of the inner circumference of the fill blocks, causing the outer circumference to increase sufficiently to form intimate contact with the rock, and imparting a small but significant additional compaction effort to the fill material.

For given strength-deformation properties of the fill, it is possible to show that a relationship exists between the amount of lateral displacement required at the inner circumference of the fill and the wedge angle at the nose of the waste unit, such that shear failure of the fill material below the point of contact does not occur. Further, if a confining air-pressure is allowed to develop ahead of the waste unit, lateral displacement is encouraged. The force required to insert the unit may then be regarded as the sum of three components, comprising the vertical component of force at the 'nose', the force required to generate the 'confining pressure', and the force required to overcome the friction resistance at the sides. The first two would be effectively constant, whilst the third would increase...
with the penetration length.

A comprehensive analysis of the mechanics of the 'push-fit' concept for high-level waste emplacement is outside the scope of this thesis. The author has considered the problem based on the assumption of a simple wedge-shaped 'nose' profile, and has concluded that a relationship could be found between the longitudinal forces, confining pressures and wedge angles necessary to achieve the desired result for given waste unit and borehole dimensions and fill properties.

An illustration of the concept as applied to vertical waste emplacement holes is shown in figure 113, and it is thought likely that, in practice, a curved 'nose' profile of the type illustrated would provide the optimum shape. The use of finite element techniques would provide a useful means of assessing the effectiveness of alternative curved shapes, and evaluating the sensitivity of the various parameters involved, including fill properties. However, the concept is also amenable to experimental methods of development, and it is thought that a combined theoretical and empirical approach could lead to the development of a practical and effective system which would eliminate many uncertainties inherent in emplacement procedures advocated to date.

The 'push-fit' concept is clearly not restricted to vertical boreholes, and may be similarly applied to in-room methods of high-level waste disposal. Refering to figure 95, it is apparent that sufficient thrust reaction could be achieved by clamping the waste emplacement device to an 'emplacement shield', or by extending anchor rams to the tunnel walls.
A further variation, which could merit detailed study involves the use of a screw system, in which rotation of the waste unit coupled with a relatively small thrust, could achieve the desired lateral fill displacement. For such a system, the size and pitch of the waste units would have to be designed to achieve fill compaction without shear failure.

It is apparent that both the 'push-fit' and 'screw-fit' concepts would demand detailed attention to the design of the waste units. Chapter 13 has indicated that relatively short waste units are to be preferred for high-level waste disposal in crystalline rocks, and it has also been shown that the provision of overpacks offers advantages in various respects. Both characteristics are desirable for the development of the concepts described, since insertion forces would have to be sustained entirely by an overpack, and short lengths would be desirable in order to minimise friction forces. Further detailed aspects to be considered include the provision of low-friction coatings, and the control of confining pressures ahead of the waste unit by means of a pressure regulating valve.

19.2 Intermediate-Level Waste Emplacement Systems

The design of intermediate-level waste emplacement systems is poorly treated in the international literature. As noted in previous chapters, the majority of the proposals for crystalline and argillaceous rock repositories envisage the use of cylindrical or cuboidal-shaped waste units, with high-density stacking arrangements. However, specified methods of backfilling the awkward shapes between the waste units and the tunnel crown area (which is generally inaccessible) do
not inspire confidence in the long-term containment properties of the systems which have been advocated thus far.

Figure 63 illustrates arrangements which have been proposed for the Belgian intermediate-level waste disposal concept (see Chapter 13) and may be used as a basis for the critical examination of some typical problems inherent in all the proposals reviewed by the author. The hexagonal stacking arrangement of cylinders as shown, provides a close-packed configuration, consistent with constraints imposed by the circular tunnel profile. However, the void spaces between the units do not provide a uniform thickness of fill around each unit. Indeed the units are placed in direct physical contact, so that if a migration path is developed from one of the outer units, this could extend to all the other units in the cross-section, via interface contacts.

Furthermore, the void spaces contain sharp re-entrant angles which the proposed fluid-form fill would be most unlikely to penetrate. This aspect is worsened by the proposed delivery of fill material from a pipe mounted in the crown of the tunnel; since material falling downwards would be unlikely to penetrate the vertical re-entrant angles below each unit and, in the absence of any mechanical compaction, arching effects would exacerbate the situation. Finally, the crown area itself would be unlikely to receive satisfactory treatment based on the system as shown.

In contrast to the above, it would be desirable to establish an intermediate-level waste emplacement and backfilling system which embodies all the following principles:
1. The stacking arrangement should result in the maximum volumetric efficiency

2. A uniform fill thickness should be provided around each waste unit, in addition to any fill barriers placed at the periphery of the tunnel

3. Fluid form fills should be capable of penetrating all voids at the same rate, so that no cavitation pockets are introduced; and no re-entrant angles should be present

4. The shape of the backfilled voids around the units should present a tortuous flow path, so that the length of the migration route through the near field backfill system is at a maximum

5. The waste stack should extend into the crown area of the tunnel (see 1 above), and the void space at the crown should be filled to the same standards of compaction as elsewhere

6. The amount of fluid form fill required should be minimised in order to simplify backfilling operations and reduce radiological hazards

7. Backfilling crews should be subjected to minimal radiological exposure

8. Emphasis should be placed on simple and reliable procedures, in which maximum control can be exercised over fill properties and insitu characteristics

The author considers that, under appropriate
circumstances, all of these requirements can be fulfilled by judicious design of the waste units, the backfill and the repository excavations. Considering first of all the waste units, it is apparent that waste unit size and shape will largely determine the available stacking options. In order to satisfy the first four requirements listed above, it would be necessary to select a tessallating solid shape; i.e. one which is capable of being stacked in such a way that it will fill the whole of the three-dimensional space occupied by the stack. By spacing the units an equal distance apart, the widths of the voids between adjacent units would then be equal in all directions; see 19.2 above.

Figure 114 indicates the four principal tessallating solid shapes; comprising the cube, the rhombic dodecahedron, the hexagonal prism and the triangular prism (55). The cuboctahedron and the octahedron are also shown; these shapes being capable of forming solid tessallations when used in conjunction with one another. Also shown for comparison are the cylinder and the sphere, neither of which are capable of forming solid tessallations.

The aspect ratios (surface area/volume ratios) of each noted shape, for a 1m³ solid, are also indicated on figure 114, and the variation of surface area with increasing volume is shown on figure 115. It is apparent that the rhombic dodecahedron has the lowest aspect ratio of the tessallating solids, since it approximates closely to the spherical shape. Thus, based on the discussion in Chapter 18.4, the rhombic dodecahedron would be expected to have a correspondingly low maximum leach rate.

However, for a given fill volume/waste volume ratio, low
aspect-ratio waste units such as the rhombic dodecahedron would also require a relatively large thickness of fill around each unit by comparison with units having higher aspect ratios, and they would tend to increase the ground-water residence time; see Chapter 19.2 above. For a given tunnel cross-section, and fill volume to waste volume ratio, it would be possible to adjust waste unit dimensions and spacings for any of the tessallating solid shapes so as to achieve the most efficient volume utilisation.

However, the consistently accurate stacking of units (including the tunnel crown area) presents practical difficulties. Furthermore, the horizontal planes which separate each layer of units are likely to prevent the full penetration of fluid-form fills.

To overcome these and other difficulties, the author proposes a waste unit and backfill emplacement system based on the use of 'palletising chambers', as illustrated in figure 116. The palletising chambers would be located at one end of each emplacement tunnel and would be constructed to comparatively large dimensions in order to provide access all around a completed waste stack. The supporting base would comprise a cast insitu concrete slab, with provision for temporary mounting on a rail bogey or air skates for transporting the completed slab/waste stack assembly into the emplacement tunnel.

Within the palletising chamber, radiation workers would receive the waste units and would stack them accurately on the pallet slab, in a pre-determined configuration to provide a close fit to the explacement tunnel profile; including the crown area. Separation at the horizontal surfaces between waste units would be achieved by using
solid pre-formed compressed fill blocks of the type described in 19.3 above. The lateral spacing between units would be similarly achieved by using strategically-placed solid fill block spacers.

Simultaneously, a length of emplacement tunnel, equivalent to the length of the finished waste stack dimensions, would be prepared by non-radiation workers. This would involve the construction of peripheral fill barriers over the floor walls and crown of the tunnel, based on concepts similar to those illustrated in figure 100. Maximum use would be made of solid pre-formed fill blocks; including materials based on bitumen, bentonite, MgO, etc., as appropriate. These would be placed so as to form a tunnel 'lining' including the crown area. However, in the crown, the fill blocks would be of a special design, incorporating a channel or groove to allow for the escape of air during the final backfilling stages.

The final waste emplacement and backfilling process would then follow. The doors of the palletising chamber would be opened, and radiation workers would transfer the entire pallet assembly into the prepared section of the emplacement tunnel. A diaphragm of fill blocks would then be erected on the trailing-edge of the pallet slab so as to provide a 'dam' at the rear of the assembly. A pressure release valve would be incorporated at the top of this dam and a temporary rigid vertical support would be provided to resist grouting pressures.

The injecting of fluid-form fill to complete the backfilling and sealing operation would then be carried out through ports at the trailing edge of the pallet slab. The latter would be designed to convey the fluid
equally to the various vertical flow channels existing within the palletised waste assembly. This operation would again be carried out by non-radiation workers; the specialised crew having returned to the palletising chamber to commence the construction of the next waste stack.

Since the volume of the prepared section of the emplacement tunnel and the volume of the waste stack would be pre-determined, and effectively constant, the volume of fluid form fill necessary to complete the backfilling operation would be known. However, the accuracy with which the required backfill volume could be calculated could be improved by photogrametric measurements of the prepared section of the emplacement tunnel, prior to the introduction of the waste pallet.

The required volume of fluid-form fill could therefore be prepared and pre-batched at a surface facility and delivered to the emplacement chamber in special enclosed tanks. The injection process need not, therefore, involve any spillage or discharge of excess material. Long pipe lines would also be avoided, and empty tanks and hose attachments could simply be returned to the surface for cleaning and re-use.

By judicious design, the size of the waste stack could be adjusted so that the required minimal volume of fluid-form fill could be injected as a single shift operation. The careful metering and control of delivery rates, volumes and pressures, combined with a fully optimised and carefully controlled waste stacking arrangement, should ensure consistent and reliable results without the need for any embedded instruments.

Considering the shapes shown in figure 114, it is
considered that the rhombic dodecahedron would probably provide the most stable stacking arrangement; since the solid tessellation involves a highly interlocking structure. Furthermore, the shape is such that the spaces around the units would provide a relatively tortuous flow path, although no sharp re-entrant angles would be present to impede the upward flow of fluid-form fill during the backfilling process.

The combination of cuboctahedron and octahedron shapes also offers interesting alternative possibilities where it is envisaged that waste units may be of two different sizes. This situation may arise where the waste inventory includes decommissioning wastes (which are likely to require some relatively large waste units). Like the rhombic dodecahedron, these units approximate to the spherical shape and have small aspect ratios, suggesting relatively low leach rates; see Chapter 18.4.

In practice, the ease of fabrication of the waste units must also be considered. However, the practical design aspects outlined above, in conjunction with the influence of waste unit size and shape on the level of containment (as discussed in Chapter 18), require most careful consideration in the design of intermediate-level waste repository systems. It appears, therefore, that many of the current international proposals could benefit from a re-appraisal of current waste production strategies.
PART 5
CONCLUSIONS
It has been shown that nuclear power production by fission reactors is likely to continue well into the next century, and that increasing emphasis will be placed on the technical and industrial evolution of advanced reactor fuel cycles. American initiatives in the late 1970's, advocating the universal adoption of a once-through cycle with disposal of spent fuel, have been rejected; and it is apparent that all nuclear power producing countries will seek to reduce their dependence on the somewhat volatile international uranium market, achieve a greater measure of self-sufficiency, and increase their efficiency, by adopting advanced nuclear fuel cycles.

In consequence, it may be anticipated that the number, size and sophistication of enrichment, reprocessing and fuel fabrication plants will continue to grow; although the need for enrichment facilities could diminish as third generation fast breeder reactors assume a more significant role. These factors will have the following important effects on the production levels and characteristics of high-activity radioactive wastes:

- High-level wastes will comprise the unwanted highly-active residue from fuel reprocessing, the composition and physical characteristics of which can be accurately controlled by currently available techniques for vitrification, packaging and interim storage. It appears most unlikely that unreprocessed nuclear fuel will be considered as a waste product.

- The volume and heterogeneity of intermediate-level
wastes associated with the various secondary industries (enrichment, reprocessing and fuel fabrication) will continue to grow and will be centered on a relatively few, large-scale plants in America and Europe. The attendant problems of waste immobilisation, conditioning and packaging may be expected to increase.

- The size and complexity of facilities requiring decommissioning at the end of their useful lives will increase, posing new and difficult radioactive waste disposal problems in the early part of the next century; in addition to those already associated with present and planned nuclear reactors.

In addition to the above, large volumes of military wastes already exist. Although the problem of disposing of these wastes has not received a comparable level of public attention, the establishment of an effective and safe disposal solution will pose problems at least as great as those associated with civilian nuclear wastes.

Based on the above, the projected nuclear waste inventories described in the available literature appear incomplete, and may grossly underestimate the ultimate scale of the problem; particularly as regards decommissioning wastes. With regard to the latter, it is desirable that the design and construction of nuclear plant should take into account the logistics of decommissioning at the end of the useful life of the facilities concerned; with a specific disposal concept in view.

It has been shown that deep underground burial represents the most satisfactory method of disposing of
high-activity radioactive wastes, based on a suitable combination of the following concepts:

- matrix of drillhole concepts for high-level wastes, provided that the foreseen waste arisings are sufficiently small
- tunnel networks with in-room disposal of high-level and/or intermediate-level wastes
- tunnel networks with in-floor disposal of high-level wastes

The above conclusions broadly coincide with the consensus of international opinion, based on waste disposal research studies carried out to date. However, the author has shown that the detailed repository design proposals put forward generally fail to achieve an optimum solution in terms of waste containment.

Whilst the nuclear establishment has achieved significant advances in devising sophisticated techniques for managing, storing and conditioning wastes, repository design studies appear to have placed excessive emphasis on the use of simple adaptations of conventional underground engineering and mining technology. In commissioning repository design studies, it also appears that an over-simplistic view has been taken in assigning generic construction and containment properties to the host rock types under consideration, and in specifying that construction principles should be based on a simple extrapolation of conventional underground construction experience.

It has been shown that the shape, size, heat output and packaging of waste units, coupled with the overall
repository layout and development concept, can have a significant influence on the long-term effectiveness of the disposal system.

These factors indicate the need for a more integrated engineering approach than hitherto, since the development of an optimum repository design and an appropriate pre-disposal strategy are mutually dependent. Based on the optimisation of variables for high-level waste disposal in jointed rocks, using a representative U.K. waste inventory, it has been shown that the following pre-disposal objectives are desirable:

- high-level wastes should be immobilised at the maximum practicable fission product concentration
- waste units should be based on AVM type canisters approximately 1m in length and 0.5m in diameter (larger diameters could be desirable, subject to constraints imposed by the dimensions of the underground openings)
- the provision of 100mm thick overpacks is desirable not only in terms of corrosion protection, but to minimise repository size, due to reduction in surface heat flux at the time of disposal
- waste units should be stored for a period of about 90 years prior to disposal

Although detailed optimisation studies have not been carried out for argillaceous and saliferous rocks, it has been shown that long periods of storage, coupled with the provision of overpacks, can provide comparable benefits.
For intermediate-level wastes, cylindrical and cuboidal shapes are not ideal. In crystalline and argillaceous rock repositories, tessallating solid shapes approximating to the sphere (such as the rhombic dodecahedron) are to be preferred; and waste unit sizes should be as large as possible, consistent with the planned dimensions of the underground openings.

For saliferous rock repositories the pre-disposal requirements are different. For high-level wastes, long-term storage and provision of overpacks are advocated as a means of reducing repository size requirements (contrary to the proposals described in the available literature). Cuboidal-shaped units are recommended for intermediate-level waste disposal, to allow a dense, planar emplacement system, without vertical stacking. Shapes approximating to the sphere are not necessary in saliferous repositories, since there is no incentive to reduce the surface area : volume ratio of the waste units. However, cylindrical intermediate-level waste units are undesirable since they reduce volumetric efficiency and promote the formation of void spaces and high stress concentrations when placed in a 'dense' stacking configuration. Cuboidal waste unit shapes are particularly favourable for use in conjunction with the disposal techniques for saliferous rock repositories advocated in this thesis.

Host rock properties and corresponding outline repository design requirements should be determined, as far as possible, in advance of the waste conditioning process, since these will determine relevant design parameters for engineered barriers required to mitigate against undesirable or uncertain aspects of host rock
performance. The importance of the latter is emphasised by the fact that, in several countries, it has been found expedient to choose sites underlying nuclear or government-controlled land, so avoiding socio-political problems associated with a broader approach to site selection. This approach clearly places an even greater level of importance on the need for an engineering optimisation of waste variables as part of the overall disposal strategy.

Since it has been shown that repository performance can only be assessed in terms of long-term risk assessments, it is evident that every effort should be made to improve the level of containment, by exercising maximum engineering control over parameters which are subject to the lowest levels of uncertainty. This implies that greater attention should be focussed on near-field phenomena and the role of engineered barriers than hitherto. Pre-disposal strategies, in particular, should be formulated on the basis of a more thorough examination of these aspects.
By means of a detailed examination of host rock properties and repository design proposals, it has been shown that considerable scope exists for modifying repository design and construction techniques, adjusting waste variables, and introducing engineered barriers in order to improve repository performance. It has been shown that the principal mechanisms of waste containment in crystalline, argillaceous and saliferous rocks are dissimilar; but that, in each case, the repository construction process can impair long-term performance, unless special preventative design measures are taken.

The following major factors have been highlighted:

- The geochemical and physical environment in a repository, modified by the transient and permanent effects of construction and waste emplacement, creates a system which is highly dynamic over the time-scales involved.

- The construction of a repository creates a potential for the development of longitudinal migration paths, via zones of peripheral disturbance, separation planes, structural linings (if present) and bulk infill materials.

- Conventional methods of designing underground openings which maximise the support contribution made by the ground, are generally inappropriate; especially in brittle rocks having low competence factors.

- Structural linings must be expected to deteriorate
in the long-term, and their failure could have adverse consequences in terms of waste containment.

- Where groundwater flow is concentrated in fissures, transport velocities and volumetric flow rates are several orders of magnitude greater than those predicted by a single generic permeability value assigned on the basis of an 'equivalent porous medium' approach.

- General uncertainties inherent in the assessment of rock mass properties are exacerbated by the need to avoid an excessive number of site investigation boreholes. Since waste containment performance must be expressed in terms of probabilities, it is necessary to enhance the overall level of waste containment attributable to the near-field system by reducing rock disturbance, reducing the risk of short-circuit migration paths, and introducing engineered barriers whose performance may be controlled and predicted with relative certainty.

- Current proposals for waste emplacement and backfilling are poorly developed in general, and are unlikely to produce reliable and predictable results.

Each of the above aspects has been examined in detail to show the sensitivity of relevant parameters, and new practical design proposals have been advocated for each of the principal host rock types under consideration.

For crystalline and hard argillaceous rocks, the need to avoid peripheral disturbance indicates that repository depths should be selected, where possible, so that the rock is completely self-supporting. Based on
assumptions made in the various international repository design proposals, a construction depth of 1000m may be too great in many instances, and shallower depths of about 500m would generally be more appropriate. Where high competence factors cannot be achieved in these rock types, it would be desirable to install rigid linings, close behind the working face, so that sufficient ground support pressure is generated to avoid peripheral rock fracture. Generally, however, it is preferable to select a formation and emplacement horizon where self-supporting openings are possible.

For crystalline and hard argillaceous rocks, the minimisation of repository size will reduce the risk of rapid radionuclide migration via natural discontinuities in the host rock. For high competence rocks, the adoption of separate phasing for repository construction and waste emplacement has been shown to be a feasible proposal which would achieve considerable reductions in overall size and the number of shaft penetrations.

Based on the minimum repository size philosophy, the optimum design parameters for a high-level waste repository in jointed rock have been established by reference to a simple thermal design model, in order to evaluate the effects of emplacement configuration, waste unit size and shape, overpack thickness, and storage period on the plan area, characteristic rock volume, scan-line intensity, volumetric efficiency and overall cost of the disposal system. A total of 232 data sets have been evaluated and various trends have been demonstrated. The study has shown that the following provides the most satisfactory basis for more detailed design studies:

- the use of a moderate depth, cuboidal, in-floor
emplacement configuration with hole depths in the range 0-50m

- the use of AVM-type high-level waste units containing the maximum practicable fission product concentration

- the provision of 100mm thick metal overpacks

- temporary storage of the waste units for about 90 years prior to disposal

It has been shown that the size reductions achieved by this optimisation technique can be more than an order of magnitude less than comparable proposals in the available literature, with a corresponding increase in volumetric efficiency. The overall costs, including temporary storage and provision of overpacks, for the preferred pre-disposal strategy and repository system described above, are of the order of £180M for predicted arisings in the United Kingdom, at 1982 prices.

For plastic clay host formations, it has been shown that severe construction difficulties may be anticipated at the depths envisaged, and that rectangular repository configurations based on the in-floor concept for high-level waste disposal are generally inappropriate. A radical design alternative, based on spiralling shield-driven tunnels, has been proposed, involving the in-room emplacement of both high-level and intermediate-level wastes and the removal of temporary linings. The outline design requirements for a special tunnelling shield for concurrent waste emplacement and lining removal have also been put forward.
Based on this concept, opportunities for reducing repository size requirements, by adjusting the size of the waste units and the interim storage period, have been described. It has also been shown that the removal of linings could significantly increase the level of waste containment by modifying the pore pressure regime over a wide area around the waste emplacement tunnels.

A radical design alternative has also been put forward for saliferous rock repositories using the minimum repository size criterion. This avoids the use of caverns for intermediate-level waste disposal (which are deprecated in general) and advocates a system based on the adaptation of long-wall mining techniques to create a dense emplacement configuration in which controlled roof closure by visco-elastic creep, without rock fracture, is utilised to avoid the need for long stand-up times in repository excavations and to eliminate the need for near-field backfill barriers.

The use of special construction techniques is also advocated as a means of further reducing the risks of longitudinal migration. The use of cut-off collars is advocated as a means of preventing migration along peripheral zones of disturbance in crystalline and hard argillaceous host rocks, and providing seals to prevent groundwater invasion in saliferous host rocks. A new procedure for repository shaft construction is unstable water-bearing overburden strata has been proposed as a means of mitigating against localised increases in permeability which may be associated with traditional ground-freezing methods.

A review of the properties of a wide range of candidate repository backfilling and sealing materials has indicated that several materials in current use for
conventional engineering applications cannot be regarded as suitable in the repository engineering context, due to the absence of reliable data concerning their geochemical longevity. Of the remaining materials, it is concluded that the performance of only a limited number can be predicted with confidence, based on the present geoscientific data-base. The majority of the more specialised materials, generally additives incorporated on account of their geochemical retention properties, require more fundamental geoscientific research; see table 30.

The cost of repository backfilling and sealing has been assessed in generic terms, according to host rock type, and it has been shown that backfilling costs may be up to four times the basic repository excavation costs. The backfilling material costs per m$^3$ have been shown to vary significantly according to the emphasis placed on the use of engineered barriers. The notional costs are £51 per m$^3$ for crystalline and hard argillaceous rocks, £39 per m$^3$ for plastic clay formations and £21 per m$^3$ for saliferous rocks. For an optimised repository system, in a jointed host rock, it has been shown that the total backfilling cost could be about 30% of the overall cost of the disposal system, and for the more extensive repository systems described in the international literature, the proportion could be considerably greater. These high backfilling costs are seen to confirm the appropriateness of the minimum repository size philosophy.

Some important parametric influences in the design of engineered barriers have been demonstrated, with reference to the sequence of radionuclide release and migration in deep underground repositories. It has been shown that swelling materials (particularly
partially-saturated pre-formed blocks of highly compressed material) can play a significant part in delaying the time taken to establish the through-flow conditions which lead to waste cladding corrosion. The use of blocks of swelling materials, in conjunction with hydrophobic materials, such as bitumen, has been advocated for the backfilling of crystalline and argillaceous rock repositories. For saliferous host rocks, fill materials with no free water content are preferred, and it is suggested that fill assemblages should incorporate anhydrite blocks as a potential swelling agent, in conjunction with bitumen 'mortar'. The extensive use of cut-off collars is also advocated.

The attributes of different metals when used in the fabrication of corrosion-resistant claddings for high-level waste units have been examined, and the relevance of Eh-pH stability fields has been demonstrated. It has been shown that aluminium, nickel, titanium, copper and lead can offer enhanced corrosion resistance, due to passivation, under appropriate Eh-pH conditions. Paradoxically, it is found that finely divided metals within the near-field backfill may also provide the best means of buffering the Eh-pH regime, and removing aggressive chemical species so as to increase corrosion resistance. The empirical data-base is shown to be generally inadequate for meaningful predictions of long-term corrosion resistance at present. This situation is exacerbated for intermediate-level waste units, where a much wider variety of waste packaging materials is envisaged.

It has been shown that the waste leaching process is strongly dependent on temperature, reinforcing the desirability of overpack protection around high-level waste units. However, it has also been shown that the
leaching mechanism is poorly understood, and that dissolution models which assume congruent leaching may imply the creation of voids in the long term, with consequent adverse effects on the backfilling and sealing system. The shape and size of waste units has been shown to exert a significant influence on leach rates, and a hierarchy exists whereby spherical shapes are to be preferred, followed by cylinders and cuboids. For non-spherical shapes, elongate units should be avoided, and for all shapes, an increase in waste unit size implies a reduction in overall leach rate.

By reference to theoretical groundwater flow equations, it has been shown that the creation of a zone of peripheral disturbance around a repository excavation, or the development of a separation plane at rock/backfill or backfill/waste interfaces, can cause a significant radionuclide migration potential. Relationships have been established, on the basis of published $K_D$ values, for some of the longer-lived radionuclides, to show how sorptive backfill components could be introduced to compensate for possible physical imperfections in backfill systems.

By reference to requirements for geochemical radionuclide retention in engineered backfills, and the need to ensure a near-field swelling potential to avoid impairment of fill properties due to the waste leaching process, it has been shown that a minimum backfill/waste volume design ratio should be established. Furthermore, to ensure consistent groundwater/fill residence times, the thickness of fill around each unit should be effectively constant. On this basis, it has been shown that tessallating solids represent the preferred shapes for intermediate-level waste units. Tessallating solid shapes approximating to the sphere are particularly
favourable in terms of structural stability and tortuosity of flow paths around possible stacking arrangements, as typified by the rhombic dodecahedron. It is suggested that this aspect of waste unit design merits detailed attention, in relation to the feasibility, and economics of waste unit fabrication.

Examination of the operational aspects of waste emplacement and near-field backfilling has shown that many of the current proposals have severe shortcomings, and are unlikely to achieve consistent and reliable results. For high-level waste disposal, a 'push-fit' (or 'screw-fit') emplacement concept has been advocated as a basis for further development, whereby the unit is inserted into a preformed assemblage of solid-form annular fill blocks. Deformation at a specially profiled 'nose' on the leading edge of the waste unit, or by screw blades along the periphery, coupled with the generation of confining pressure, could achieve a dense insitu fill compaction with no interface defects, enabling insitu fill properties to be properly engineered to laboratory standards.

A palletising concept has been advocated for intermediate-level waste emplacement and backfilling, as a means of eliminating many of the uncertainties and potential conflicts inherent in current repository proposals. This involves the construction of carefully designed waste 'stacks', on palletising slabs within special chambers at the end of each emplacement tunnel. Completed stacks may then be transported into specially prepared sections of the emplacement tunnel and pre-batched and pre-mixed quantities of fluid-form fill may be injected to complete the emplacement and backfilling cycle in a single operation. The procedure requires no elaborate instrumentation systems, since
full quality control may be achieved by reference to known fill volume requirements.
Several areas requiring further research and development have been indicated throughout this thesis. It is proposed to summarise these briefly, under the following headings:

(a) pre-disposal strategies  
(b) waste conditioning  
(c) host rocks  
(d) backfills  
(e) construction methods  
(f) geochemical aspects  
(g) migration modelling  
(h) operational aspects

(a) Pre-Disposal Strategies

The research requirements under this heading are self-evident from the discussion in Chapter 20, and will clearly be specific to the particular country concerned. This thesis has demonstrated the interaction and inter-dependence between decisions concerning the design, management, conditioning and storage of waste units, and the design, construction and operation of an underground disposal system.

It is thought that considerable scope exists for the further detailed optimisation of variables, based on procedures similar to those which have been outlined by this author. The optimisation of high-level waste variables and outline repository design parameters, based on a computer simulation of thermal design requirements for a repository model similar to that described in...
Chapter 15 would be valuable.

Further investigation of the available options for intermediate-level waste fabrication could also lead to an improved basis for the development of disposal strategies, utilising the results of parametric sensitivity studies similar to those outlined in Chapters 18 and 19.

(b) Waste Conditioning

It has been shown that the current models which describe the leaching process for borosilicate glass, generally assume that the glass-water composite will fracture during handling, or will de-vitrify so as to create discrete lumps of material; see Chapter 18.4. It has also been shown that the latter implies a greatly increased leach rate, by comparison with the intact material. The desirability of preventing degradation of this type is therefore self-evident. It would be desirable to research methods of preventing degradation, either by modifying the design of the waste units or by developing an alternative immobilising material which is more stable in the long-term; e.g. 'Synroc'.

General inadequacies in the understanding of the mechanism of leaching are also apparent, and it would be particularly desirable to establish:

- whether leaching occurs as a dissolution process, implying volume reduction
- whether speciation of radionuclides will occur as a result of the leaching process
The above factors are clearly relevant, not only to high-level waste immobilisation and conditioning, but also to intermediate-level wastes.

(c) Host Rocks

Despite considerable geoscientific research, it is apparent that significant uncertainties remain concerning the containment properties of potential repository host rocks at great depth. To a large extent, these uncertainties reflect the practical difficulties in measuring or extrapolating the measured properties of small rock samples to those of the rock on a km$^3$ scale. However, various specific areas have been identified which merit a sustained programme of fundamental geotechnical research. These are:

- the application of scan-line theories in the development of optimised design solutions for repositories in jointed rock masses
- the effects of peripheral overstressing on localised permeability in brittle rocks
- the effects of pore pressure changes around tunnels in plastic clays upon the time required for re-establishment of a steady-state hydraulic regime
- the propensity for anealing of induced fractures in saliferous rocks, or re-crystallisation of crushed saliferous spoil during ground-fill interaction
At a more fundamental level, it is evident that current knowledge concerning the properties of crystalline and argillaceous rocks at great depth is inadequate, particularly with respect to hydraulic properties. In the author's view, investigative drilling programmes should be re-established (notwithstanding the desirability of extending the proposed storage periods for high-level wastes), so that an adequate geotechnical data-base can be established at a sufficiently early stage in the overall research and development programmes.

(d) Backfills

It has been shown that there is an inadequate level of knowledge concerning the properties of a variety of potentially promising backfilling and sealing materials. This applies particularly to materials whose geochemical properties are of greatest relevance in the waste disposal context. Materials indicated in categories 2 and 3 in table 30 merit serious consideration as components of engineered backfills in high-activity radioactive waste repositories; and a systematic evaluation of their relevant properties, under appropriate operating conditions, is warranted.

(e). Construction Methods

Three radical construction methods have been advocated which could provide a suitable basis for the development of practical repository construction techniques. These are:
the spiral tunnel concept for repository construction, waste emplacement and backfilling in plastic clay host formations, incorporating shield tunnelling methods; see figures 92, 93 and 95

the long-wall system for intermediate level waste emplacement and backfilling in saliferous rock repositories; see figure 96

a special ground-freezing technique for shaft construction in unstable, water-bearing overburden materials; see figure 97

The author considers that each of the above merits serious technical consideration as a possible basis for practical design and development. In each case, this would be dependent upon the successful outcome of detailed design studies and full-scale in situ validation trials.

(f) Geochemical Aspects

It has been shown that chemical buffering and geochemical retention can play a vital role in providing a near-field engineered containment system which complements the properties of the natural host-rock barrier. However, it is apparent that, thus far, experiments have generally failed to simulate the chemical and physical conditions likely to be experienced in the deep underground environment.

Furthermore, retention experiments based on the $K_D$
concept have generally failed to include the longer-lived radionuclides which represent the greatest hazard in terms of their mobility (and radiological toxicity) over the time-scales to be considered; notably the anionic species. It is also apparent that insufficient data exists concerning the retentive capacities of different materials, so that at present it is not possible to predict what volume of a particular sorptive component is required to retain given quantities of radionuclides.

In view of the above, it is considered that a systematic evaluation of the buffering and retention properties of potential backfill additives should receive priority attention. It appears that the Eh-pH concept could be a valuable tool in the evaluation of the influence of geochemical environment upon geochemical buffering and radionuclide retention phenomena.

(g) Migration Modelling

The relevance of fissure-flow phenomena associated with separation planes, and the influence of localised permeability increases associated with peripheral rock disturbance have been highlighted in Chapter 18.5. The need to counter-balance the hydraulic and geochemical retention properties of fills has been demonstrated, and the role of hydrophobic fills in creating imperfect hydraulic boundaries has been described.

To date, surprisingly little effort has been made to evaluate the performance requirements for near-field
backfill systems, in terms of a combination of the above, despite a plethora of sophisticated migration modelling studies through host rock barriers. Since near-field conditions are subject to engineering control, whereas conditions within the host rock are not, this author advocates an extension of migration modelling studies in order to include a systematic evaluation of near-field migration phenomena, as an important adjunct to engineered barrier design studies.

Parametric sensitivity studies to assess the influence of hydrophobic fill barriers with small imperfections, cut-off collars, and sorptive fills with varying permeabilities, could provide a valuable insight into research and development requirements for highly effective engineered barrier systems.

(h) Operational Aspects

Specific proposals have been put forward for the development of operational systems for high-level and intermediate-level waste emplacement and backfilling. Both require research into the interaction between fill materials and waste units, at a theoretical and practical level.

The 'push-fit' and 'screw-fit' concepts advocated by this author, (see Chapter 19) for high-level waste emplacement and backfilling form suitable topics for theoretical studies to complement parallel research into the properties of compressed 'dry' fill blocks of different compositions. It has been shown in Chapter 19, that these emplacement concepts can only operate successfully under specific conditions, which
prevent the occurrence of slabbing failure or spalling of material from the sides of the fill; see figure 116.

Finite element studies should be performed to examine the effects of different waste unit 'nose' profiles, confining pressures and fill properties on the forces required to insert waste units, so as to achieve a dense, in situ compaction of pre-formed annular-shaped fill blocks without shear failure. Similar studies related to the screw-fit concept should be carried out to establish the optimum blade dimensions, screw blade pitch dimensions, and corresponding emplacement forces under given conditions.

The results of such studies should be used as a basis for full-scale experimental trials, using emplacement holes formed in mass concrete, with different sizes and shapes of fill blocks and waste units. Such experiments could also indicate the scope for more detailed practical design improvements, by provision of low-friction coatings, varying borehole tolerances and surface roughnesses, etc.

Proposals put forward in Chapter 19 for palletising intermediate-level waste units, in stacks which incorporate solid-form fill blocks, should be studied in terms of the optimum relationships between tunnel size and shape, waste unit size and shape, and waste unit spacing, for a range of tessallating solids. Appropriate geometrical relationships should be established for standard horseshoe and circular shaped tunnel profiles, for different waste/backfill volumes and fill thicknesses, so that optimum waste
unit designs and stacking configurations can be established in terms of volumetric efficiencies.

Based on the results of such studies, it would be desirable to perform large-scale validation trials in instrumented 'mock-up' tunnel sections, using instrumented lightweight concrete blocks as a simulant for intermediate-level waste units. Irrigation points could be introduced at the periphery of the 'mock-up' tunnel section, to investigate the swelling pressures generated by solid-form fill blocks included in the waste stack and at the periphery of the test chamber; and rotary coring techniques could be used to examine the integrity of the filling system.
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THE UNDERGROUND DISPOSAL OF
HIGH-ACTIVITY RADIOACTIVE WASTES

A thesis submitted to the University of Surrey
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Relationship between radioactivity and elapsed time
Relationship between binding energy per nucleon and mass number, A.

**FIG 5**

- **DECREASING STABILITY**
- **MAXIMUM STABILITY**
- **DECREASING STABILITY**

binding energy per nucleon \( \frac{\Delta E}{A} \)
FIG 6

Principal components of a thermal nuclear reactor
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1. Groundwater inflow phase leading to saturation of emplacement area
2. Corrosion of waste containers by groundwater
3. Dissolution of wastes by leaching action of groundwater
4. Migration of wastes by groundwater flow and diffusion processes

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Shear strength $C_u$ & effective overburden pressure $p_0$

b) effect of overconsolidation due to overburden removal

$C_u / p_0 = 0.11 + 0.0037(\text{PI})$

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FIG 18

Scale effects associated with rock mass discontinuities
a. flow in an idealised fissure of average width \( b \).

\[
\frac{b_1 + b_2 + b_3 + \ldots + b_n}{n} = b
\]

Number of fissures/unit length = \( \lambda \)

b. flow through an idealised set of parallel fissures of average width \( b \).

FIG 19

Idealised models for the assessment of fissure flow phenomena
a) Diffusion into micro-cracks in fissured rocks

b) Diffusion into micro-cracks in fissured rocks

c) Effect of diffusive retention on radionuclide concentration front

**FIG 20**

Diffusive retention at the boundaries of flow channels in fissured rocks
FIG 21

Ionic potential diagram for radionuclides in the pure water environment
Natural geothermal gradients and variations in leach rate with temperature based on standard laboratory tests on borosilicate glass.

FIG 22

(a) typical ambient repository temperature at different waste emplacement horizons.

(b) effect of temperature on leach rate (adapted from Hill and Grimwood; 1978)
Near-field temperature profile associated with high-level waste emplacement

FIG 23

c. theoretical radial temperature profile assuming uniform thermal conductivities throughout the system

a. typical high-level waste unit surrounded by backfill

b. horizontal section through waste unit
SOIL STRENGTH CLASSIFICATION

<table>
<thead>
<tr>
<th>V.Soft</th>
<th>Soft</th>
<th>Firm</th>
<th>Stiff</th>
<th>Hard</th>
</tr>
</thead>
</table>

ROCK STRENGTH CLASSIFICATION

<table>
<thead>
<tr>
<th>V.Weak</th>
<th>Weak</th>
<th>Mod. Weak</th>
<th>Mod. Strong</th>
<th>Strong</th>
<th>V.Strong</th>
<th>Extremely Strong</th>
</tr>
</thead>
</table>

Deformation Modulus MN/m²

Compressive Strength MN/m²

FIG 24

Outline comparison of strengths and deformation moduli of potential host rocks

- A: Crystalline rocks
- B: Saliferous rocks
- C: Indurated argillaceous rocks
- D: Plastic clays
\[ k_0 = \frac{\sigma_H}{\sigma_V} \]

(adapted from Hoek & Brown, 1980)

<table>
<thead>
<tr>
<th>Depth Range</th>
<th>Typical K₀ Range</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Crystalline +</td>
</tr>
<tr>
<td>0 - 500</td>
<td>1.0 - 4.0</td>
</tr>
<tr>
<td>500 - 1000</td>
<td>0.5 - 2.5</td>
</tr>
<tr>
<td>1000 - 1500</td>
<td>0.5 - 2.0</td>
</tr>
</tbody>
</table>

Variation of K₀ with depth (after Hoek & Brown E T; 1980)
Normally consolidated clay.

Increasing $K_q$ with increasing values of OCR.

Figure 26: Assessment of $K_0$ for unindurated plastic clays.
FIG. 27
Influence of \( K_0 \) on the intensity and distribution of induced stress around circular openings in elastic, homogeneous, isotropic rock (after Poulos H G, Davies E H; 1974)
Outline indication of ground support requirements in different host rocks based on stability factors
Examples of the influence of stress redistribution on the zone of plastic disturbance or yielding around circular openings in rock
a) load deformation curves for brittle rocks with different competence factors

b) effect of rapid support installation in brittle rocks

c) transfer of stress to backfill due to ultimate failure of support system in brittle rocks

d) controlled transfer of stress to stiff backfill

FIG 30

Ground-support interaction diagrams for repository excavations
FIG 31

Generalised time-dependent constitutive stress - strain behaviour of saliferous rocks
Radial development of plastic, visco-elastic and elastic zones around openings in saliferous rocks
**FIG 33**

Permeability and fissure-size limits of currently available methods of ground water control and ground treatment.
Potential longitudinal flow path anomalies in and around backfilled repository excavations.
reprocessing plant

vapour condensing and treatment facility

(a) Liquid waste emplacement in a deep hole with rock melting and conversion to rock-waste matrix

5000 to 10,000 m

crystalline host rock

limit of rock-waste melt

waste reception and emplacement facility

5000 to 10,000 m

treated units of solid waste

crystalline host rock

(b) Solid waste emplacement in a deep hole without rock melting.

FIG 35

The very deep drillhole concept
(adapted from Schneider K J, Platt A M; 1974)
(a) Facilities for waste emplacement

(b) Section showing emplacement holes

FIG 36

The matrix of drillholes concept
(adapted from Schneider K J, Platt A M; 1974)
Reprocessing plant Vapor condensing and treatment facility. (see fig 2)

Two sealed and cased holes containing waste, vapour, monitoring and instrument lines.

Exploded cavity for liquid waste storage.

2000 to 5000 m

Limit of melt and final waste solid

b) Liquid waste emplacement in an exploded cavity. Rock melting and conversion to rock waste matrix.

Casing
Cement
Gas or liquid monitor fill
Steam flow casing
Cement to seal bottom of monitoring annulus.
Rubble filled cavity
Water waste inflow pipe

SECTION

PLAN

b) Pipe and hole configuration.

FIG 37

The exploded cavity concept (adapted from Schneider K J, Platt A M; 1974)
reprocessing plant  
your condensing and treatment facility (see fig 2)  

(a) Liquid waste emplacement with interim cooling and subsequent conversion to rock-waste matrix

(b) Solid waste emplacement with interim cooling and subsequent conversion to rock-waste matrix

FIG 38

The mined cavern concept with rock melting due to high-level waste emplacement
(adapted from Schneider K J, Platt A M; 1974)
The mined cavern 'WP-cave' concept
(adapted from Akesson B A, Hok J H; 1977)
(a) Direct emplacement on tunnel floor with mechanical fixing at base of waste unit (after Chapwood and Gnirk); 1977

(b) Emplacement by grouting bases into recess in tunnel floor (after Schnieder and Platt); 1974

(c) Emplacement in 'shelves' in tunnel walls (after Beale, Griffin and Burton); 1979

(d) Possible high-density stacking arrangement for intermediate-level waste units, depending on tunnel and waste unit geometry.

FIG 40

The in-room disposal concept
a) Emplacement in vertical drillholes.

b) Emplacement in vertical holes formed in concrete filled trench.

c) Emplacement in long inclined drillholes.

d) Emplacement in raise-borings, with interim air circulation.

FIG 41

The in-floor disposal concept
FIG 42

Sweden; distribution of potentially suitable host rock formations (adapted from KBS; 1977)
Canada; distribution of potentially suitable host rock formations (adapted from Barnes C. R.; 1979)
a) Principle areas of intrusive igneous rocks in the U.S.A.

b) Principle areas of volcanic igneous rocks in the U.S.A.

c) Principal rock salt deposits in the U.S.A.

FIG 44

United States; distribution of potentially suitable host rock formations (adapted from Ekren E R, Dinwiddie G A, et al; 1974)
United Kingdom; distribution of potentially suitable granitic host rock formations
(adapted from Anderson L J, Barbreau A, et al; 1980)

FIG 45
FIG 46

FIG 47

Representation of a typical deep underground repository system.
6.14 km
3.6 km
2.9 km

Vault boundary

Exclusion zone boundary

a) Plan layout of repository area

b) Schematic view of repository

FIG 48

Canadian reference design proposals; repository configuration
(after Acres Consulting Services Ltd, et al; 1980)
FIG 49

French reference design proposals; repository configuration
(adapted from 1st European Comunity Conf. on radioactive waste disposal; 1980)
United Kingdom reference design proposals; repository configuration
FIG 52

United States reference design proposals; repository configuration (after Parsons, Brinckerhoff, Quade and Douglas Inc; 1978)

LLW = low level waste
ILW = intermediate level waste
HLW = high level waste
M&M = men & materials
FIG 53

Belgian reference design proposals; repository configuration
(adapted from 3rd Annual Progress Report of the Commission of the European Communities; 1980)
FIG 54

Canadian reference design proposals; principal dimensions of drifts, emplacement rooms and drillholes
(after Acres Consulting Services Ltd, et al; 1980)
Canadian reference design proposals; ventilation flow system
(after Acres Consulting Services Ltd, et al; 1980)
Swedish reference design proposals; principal dimensions of emplacement tunnels and drillholes (after KBS; 1977)
Swedish reference design proposals; ventilation flow system
(after KBS; 1977)
Swedish reference design proposals; waste emplacement and backfilling sequence (after KBS; 1977)
Dimensions of emplacement tunnels

(b) Arrangement of emplacement holes in full panel

Holes -- section A-A

Canister emplacement

FIG 59

United States reference design proposals; principal dimensions of emplacement tunnels and drillholes (after Parsons, Brinckerhoff, Quade and Douglas Inc; 1978)
FIG 60

West German reference proposals; shaft lining
(adapted from Rothemeyer H; 1980)
Swedish reference design proposals; tunnel backfilling procedure (after KBS; 1977)
FIG 63

a. Intermediate level waste backfilling system

b. High level waste backfilling system

Belgian reference design proposals; intermediate-level waste stacking system and backfilling arrangements (adapted from Heremens R, Manfroy P, et al; 1980)
Netherlands reference design proposals; backfilling system for high-level waste emplacement holes
(adapted from Hamstra J; 1980)
FIG 65

Relationships between emplacement hole depth and number of waste units per hole for different waste canister types and overpack thicknesses.
FIG 66

Variation in total heat output with time for different waste canister types
Characteristic heat conduction times for different radial distances from a heat source in granite having a thermal diffusivity of $10^{-6} \text{ m}^2/\text{sec}$.
Temperature rise at centre of an array of approx 7000 Harvest canisters in granite; heat output 1kW per unit; unit spacing 20m
(adapted from Bourke P J, Hodgkinson D P; 1977)
Temperature rise at the centre of an array of approx 7000 Harvest canisters in granite; heat output 1kW per unit; unit spacing 15m (adapted from Bourke P J, Hodgkinson D P; 1977)
where

- $I_1 = \text{length of emplacement tunnel} = \frac{xs}{y} + 25$
- $I_2 = \text{length of peripheral tunnel} = t(y-1) + 40$

where
- $x = \text{No. of emplacement holes}$
- $y = \text{No. of emplacement tunnels}$
- $s = \text{emplacement hole spacing}$
- $t = \text{tunnel spacing}$
FIG 71 Characteristic areal dimension (km)

Repository design optimisation; variations in characteristic areal repository dimension for waste unit types 1, 2, 3 (half-AVM canisters)
FIG 72

Repository design optimisation; variations in characteristic areal repository dimension for waste unit types 4, 5, 6 (half-Harvest canisters)
FIG 73

Repository design optimisation; variations in characteristic areal repository dimension for waste unit types 7, 8, 9 (AVM canisters)
Repository design optimisation; variations in characteristic areal repository dimension for waste unit types 10, 11, 12 (Harvest canisters)
Repository design optimisation; variations in characteristic rock volume and scan-line length for planar arrays of waste unit types 1, 2, 3 (half-AVM canisters)
Repository design optimisation; variations in characteristic rock volume and scan-line length for planar arrays of waste unit types 4, 5, 6 (half-Harvest canisters)
Repository design optimisation; variations in characteristic rock volume and scan-line length for planar arrays of waste unit types 7, 8, 9 (AVM canisters)
FIG 79

Repository design optimisation; variations in characteristic rock volume and scan-line length for cuboidal arrays of waste unit types 1, 2, 3 (half-AVM canisters)
Repository design optimisation: variations in characteristic rock volume and scan-line length for cuboidal arrays of waste unit types 4, 5, 6 (half-Harvest canisters)
Repository design optimisation: variations in characteristic rock volume and scan-line length for cuboidal arrays of waste unit types 7, 8, 9 (AVM canisters)
Repository design optimisation: variations in characteristic rock volume and scan-line length for cuboidal arrays of waste unit types 10, 11, 12 (Harvest canisters)
Repository design optimisation: variations in gross volumetric efficiency for waste unit types 1, 2, 3 (half-AVM canisters)
Repository design optimisation; variations in gross volumetric efficiency for waste unit types 7, 8, 9 (AVM canisters)
Repository design optimisation; variations in gross volumetric efficiency for waste unit types 10, 11, 12 (Harvest canisters)
Repository design optimisation; variations in net volumetric efficiency for waste unit types 1, 2, 3 (half-AVM canisters)
Repository design optimisation; variations in net volumetric efficiency for waste unit types 4, 5, 6 (half-Harvest canisters)
Repository design optimisation; variations in net volumetric efficiency for waste unit types 7, 8, 9 (AVM canisters)
Repository design optimisation; variations in net volumetric efficiency for waste unit types 10, 11, 12 (Harvest canisters)
In situ planning and development of optimum layout for repository construction in crystalline rocks
FIG 92

Outline design concept for a repository in unindurated plastic clay; based on a spiral tunnel system
Stage 1: construction of first 2 loops; shaft B used as construction shaft.

Stage 2: shaft A adapted for use as construction shaft; shaft B adapted for use as waste delivery shaft; waste emplacement shield constructed in chamber D.

Stage 3: bulkhead inserted between C and D; waste emplacement shield commences first loop from D to B, simultaneously removing linings and backfilling.

Stage 4: as waste emplacement and backfilling progresses in first loop from D to B, linings removed are used in the construction of the 3rd loop.

FIG 93
Outline sequence for construction, waste emplacement and backfilling; based on the spiral tunnel design concept for unindurated plastic clay host formations.

note: in subsequent stages, the bulkhead is moved progressively towards A as successive loops of emplacement tunnel are filled.
FIG 94

Theoretical stress redistribution and pore pressure response around a circular tunnel in clay for $Cv/Cu = 5$; assuming $K_o = 1$
FIG 95

Essential features of an articulated shield system for simultaneous lining removal, waste emplacement and backfilling
Direction of longwall advance and intermediate level waste emplacement

Waste delivery tunnel 'man gate'

Temporary tunnel backfill

High level waste emplacement tunnels

Access tunnels for high-level waste disposal (in floor method)

Access tunnel for intermediate level waste disposal (in room long wall method)

Self advancing hydraulic roof support

Shear cutter

Intermediate level waste units

Fully mechanised long wall mining

HLW emplacement holes

FIG 96

Outline design concept for a saliferous rock repository based on the minimum area/minimum backfill philosophy
Outline concept for repository shaft construction using ground-freezing overburden strata

(a) alternative ground-freezing concept for repository shaft construction

(b) detail of special freezing-tube design

Fig 97
Undisturbed host rock

Longitudinal flow through disturbed zone 'cut off' by enlargement

Bulk excavation tunnel or shaft

Minimal disturbance around enlargement

Zone of peripheral disturbance

FIG 98
Cut-off collar concept for the prevention of peripheral groundwater flow
a) Estimate of time required for saturation of fill comprising dry compacted bentonite blocks within circular tunnels (radial inflow) assuming $C_{ys} = 2.9 \times 10^{-10} \text{ m}^2 \text{ yr}^{-1}$

b) Estimate of time required for structure of peripheral fill comprising dry compacted bentonite blocks (unidirectional inflow), based on a direct extrapolation of swelling tests on small oedometer samples.

FIG 99

Estimates of lower-bound saturation times for dry compacted Na-bentonite fill
a) Possible arrangement of backfill assemblages to create an engineered barrier in self supporting crystalline or argillaceous host rocks

Rigid lining incorporating impermeable coatings, caulked joints etc and outer layer of swelling material.

Thicker inner layer of swelling material.

Incompressible fill.

Flexible impermeable barrier (eg. bitumen blocks).

Swelling fill material (eg. dry bentonite blocks).

Incompressible fill.

Principal sources of groundwater inflow from rock.

Imperfections in impermeable barrier.

Induced swelling pressures seals off fissures at point of intersection.

b) Possible arrangement of backfill assemblages in brittle argillaceous rocks with low competence factor

Rigid lining installed close to working during construction.

Simultaneous filling and removal of rigid lining.

Flexible impermeable barrier (eg. bitumen blocks).

Bulk fill of dense pre-fabricated fill blocks (eg. bentonite clay).

High negative pore-pressure response in surrounding clay.

High internal support pressure maintained by fill.

Swelling pressures.

c) Possible arrangement of backfill assemblages in clays and uncemented argillaceous rocks with high over-load factor

Fill assemblages designed to act as engineered barriers in active areas of crystalline and argillaceous rock repositories
Fill assemblages designed to act as engineered barriers in redundant areas of saliferous rock repositories.
FIG 102

Schematic diagram of anodic current density \( I \) versus electrode potential \( E \) for stainless steel
FIG 103

Eh-pH corrosion stability fields for aluminium and nickel
FIG 104
Eh-pH corrosion stability fields for titanium and copper

(a) Eh-pH diagram for the titanium-water system at 25°C

(b) Eh-pH diagram for the copper-water system in the presence of S and CO at 25°C (total dissolved sulphur = 10^{-1} mol/l)
(a) Eh-pH diagram for the lead-water system in the presence of sulphur at 25°C (total dissolved sulphur = 10⁻⁶ mol/l)

(b) Eh-pH diagram for the lead-water system in the presence of S and CO₃ (total dissolved sulphur = 10⁻⁶ mol/l, total dissolved carbonate = 10⁻¹ mol/l).

FIG 105

Eh-pH corrosion stability fields for lead
FIG 106

Influence of waste unit size and shape on release rates, assuming congruent leaching at $10^{-6}$ gm/gm²/day.

(a) release periods for waste units of various shapes and sizes

(b) release rate as a function of time for $1\text{m}^3$ waste according to shape
FIG 107

2-dimensional flow nets for transverse groundwater flow through a backfilled circular opening; assuming perfect hydraulic boundaries and homogeneous isotropic permeabilities in the host rock and fill materials.
Imperfection on downstream side

Imperfection on upstream side

principal direction of hydraulic gradient

(a) three-dimensional flow system through backfill due to imperfections in impermeable barrier.

(b) three-dimensional flow system through backfill due to intersection of a plane continuous fissure.

FIG 108

Idealised representation of 3-dimensional flow patterns in a backfilled circular opening with non-uniform boundary conditions
Travel time isochrons

Effective toxic life for typical radionuclides

Published $K$ values for desert sub-soil [31]

Note: travel time of radionuclide $T_r = C_k^k$; where $C = \frac{z}{10}$.

Diagram assumes $C = 100$

**FIG 109**

Combined permeability and sorption effects on radionuclide travel times through an engineered backfill barrier
Effects of variations in backfill permeability, interface separation and peripheral disturbance on longitudinal groundwater flow phenomena
Influence of variations in hydraulic properties on groundwater travel times and flow volumes for longitudinal migration along a 100m length of 5m diameter circular opening; assuming $i = 0.001$
Thrust ram jacked off tunnel crown.

Shielded transport and emplacement flask

Temporary steel collar

Special profile at nose of waste unit

Waste unit with thick metal overpack and low-friction coating

Pressure release valve

Radial fill compression during emplacement

Engineered backfill comprising pre-fabricated dry annular blocks compressed during insertion of waste unit

Emplacement borehole

FIG 113

Push-fit emplacement concept for high-level waste units
Comparison of geometric forms for intermediate-level waste units in terms of stacking and surface area to volume relationships.
Surface to volume relationships for various geometric waste forms

FIG 115
<table>
<thead>
<tr>
<th>REACTOR</th>
<th>MAGNOX</th>
<th>AGR</th>
<th>HTGR</th>
<th>PWR</th>
<th>BWR</th>
<th>CANDU</th>
<th>SGHWR</th>
</tr>
</thead>
<tbody>
<tr>
<td>County of Origin</td>
<td>Britain</td>
<td>Britain</td>
<td>Various</td>
<td>U.S.A.</td>
<td>U.S.A.</td>
<td>Canada</td>
<td>Britain</td>
</tr>
<tr>
<td>Moderator</td>
<td>Graphite</td>
<td>Graphite</td>
<td>Graphite</td>
<td>H₂O</td>
<td>H₂O</td>
<td>D₂O</td>
<td>D₂O</td>
</tr>
<tr>
<td>Fuel</td>
<td>Natural uranium metal</td>
<td>Uranium oxide</td>
<td>Uranium carbide</td>
<td>Uranium oxide</td>
<td>Uranium oxide</td>
<td>Uranium oxide</td>
<td>Uranium oxide</td>
</tr>
<tr>
<td>Enrichment</td>
<td>Nil</td>
<td>2.3%</td>
<td>10%</td>
<td>3.3%</td>
<td>2.6%</td>
<td>Nil</td>
<td>2.25%</td>
</tr>
<tr>
<td>Coolant</td>
<td>CO₂</td>
<td>CO₂</td>
<td>He</td>
<td>H₂O</td>
<td>H₂O</td>
<td>D₂O</td>
<td>H₂O</td>
</tr>
<tr>
<td>Fuel Cladding</td>
<td>Magnox</td>
<td>Stainless steel</td>
<td>Silicon carbide</td>
<td>Zircaloy</td>
<td>Zircaloy</td>
<td>Zircaloy</td>
<td>Zircaloy</td>
</tr>
<tr>
<td>Coolant Pressure (Mn/m²)</td>
<td>1.9</td>
<td>3.9</td>
<td>4.7</td>
<td>14.8</td>
<td>6.9</td>
<td>8.5</td>
<td>5.9</td>
</tr>
<tr>
<td>Coolant Outlet Temperature (°C)</td>
<td>400</td>
<td>650</td>
<td>720</td>
<td>317</td>
<td>286</td>
<td>305</td>
<td>272</td>
</tr>
<tr>
<td>Approx. Core Volume (m³)</td>
<td>382</td>
<td>176</td>
<td>80</td>
<td>8.3</td>
<td>12.7</td>
<td>74.4</td>
<td>39</td>
</tr>
<tr>
<td>Burn-up (MW days/tonne) of uranium</td>
<td>4000</td>
<td>13,000</td>
<td>100,000</td>
<td>20,000</td>
<td>20,000</td>
<td>9000</td>
<td>21,000</td>
</tr>
<tr>
<td>Power Density (kW/litre)</td>
<td>1</td>
<td>4.5</td>
<td>6</td>
<td>50–100</td>
<td>50–100</td>
<td>16</td>
<td>11</td>
</tr>
</tbody>
</table>
TABLE 2  TYPICAL COMPOSITION OF PWR REACTOR FUEL BEFORE AND AFTER DISCHARGE; ASSUMING 3 YEARS REACTOR OPERATION (Adapted from Cohen BL; 1977).

<table>
<thead>
<tr>
<th>Radionuclides</th>
<th>% weight</th>
<th>kg/tonne of fuel</th>
</tr>
</thead>
<tbody>
<tr>
<td>Original Composition</td>
<td></td>
<td></td>
</tr>
<tr>
<td>238 U</td>
<td>96.7</td>
<td>967</td>
</tr>
<tr>
<td>235 U</td>
<td>3.3</td>
<td>33</td>
</tr>
<tr>
<td>Final Composition</td>
<td></td>
<td></td>
</tr>
<tr>
<td>238 U</td>
<td>94.3</td>
<td>943</td>
</tr>
<tr>
<td>235 U</td>
<td>0.8</td>
<td>8</td>
</tr>
<tr>
<td>236 U</td>
<td>0.46</td>
<td>4.6</td>
</tr>
<tr>
<td>Pu isotopes</td>
<td>0.89</td>
<td>8.9</td>
</tr>
<tr>
<td>other actinides</td>
<td>0.05</td>
<td>0.5</td>
</tr>
<tr>
<td>fission products</td>
<td>3.5</td>
<td>35</td>
</tr>
</tbody>
</table>
### Table 3: Principal Radionuclides in High-Activity Radioactive Wastes

<table>
<thead>
<tr>
<th><strong>Fission Products</strong> (+ daughters)</th>
<th><strong>Approx. Radioactive Half-Life (years)</strong></th>
<th><strong>Ionic Form</strong></th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>ELEMENT</strong></td>
<td><strong>RADIONUCLIDE</strong></td>
<td><strong>Selenium</strong></td>
</tr>
<tr>
<td>Selenium</td>
<td>$^{79}_{\text{Se}}$</td>
<td>$6.5 \times 10^4$</td>
</tr>
<tr>
<td>Rubidium</td>
<td>$^{87}_{\text{Rb}}$</td>
<td>$4.8 \times 10^4$</td>
</tr>
<tr>
<td>Zirconium</td>
<td>$^{93}_{\text{Zr}}$</td>
<td>$1.5 \times 10^5$</td>
</tr>
<tr>
<td>Niobium</td>
<td>$^{93}_{\text{Nb}}$ (d)</td>
<td>$13.6$</td>
</tr>
<tr>
<td>Niobium</td>
<td>$^{94}_{\text{Nb}}$</td>
<td>$2.0 \times 10^4$</td>
</tr>
<tr>
<td>Technetium</td>
<td>$^{99}_{\text{Tc}}$</td>
<td>$2.1 \times 10^5$</td>
</tr>
<tr>
<td>Palladium</td>
<td>$^{107}_{\text{Pd}}$</td>
<td>$6.5 \times 10^5$</td>
</tr>
<tr>
<td>Tin</td>
<td>$^{126}_{\text{Sn}}$</td>
<td>$1.0 \times 10^5$</td>
</tr>
<tr>
<td>Antimony</td>
<td>$^{126m}_{\text{Sb}}$ (d)</td>
<td>$19.0$</td>
</tr>
<tr>
<td>Iodine*</td>
<td>$^{129}_{\text{I}}$</td>
<td>$1.7 \times 10^7$</td>
</tr>
<tr>
<td>Caesium</td>
<td>$^{137}_{\text{Cs}}$</td>
<td>$30.0$</td>
</tr>
<tr>
<td>Samarium</td>
<td>$^{147}_{\text{Sm}}$</td>
<td>$1.0 \times 10^{11}$</td>
</tr>
<tr>
<td>Samarium</td>
<td>$^{161}_{\text{Sm}}$</td>
<td>$93.0$</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th><strong>Actinides</strong> (+ daughters)</th>
<th><strong>Approx. Radioactive Half-Life (years)</strong></th>
<th><strong>Ionic Form</strong></th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>ELEMENT</strong></td>
<td><strong>RADIONUCLIDE</strong></td>
<td><strong>Lead</strong></td>
</tr>
<tr>
<td>Lead</td>
<td>$^{210}_{\text{Pb}}$</td>
<td>$21.0$</td>
</tr>
<tr>
<td>Polonium</td>
<td>$^{210}_{\text{Po}}$ (d)</td>
<td>$138$ days</td>
</tr>
<tr>
<td>Radium</td>
<td>$^{226}_{\text{Ra}}$</td>
<td>$1.6 \times 10^3$</td>
</tr>
<tr>
<td>Radium</td>
<td>$^{228}_{\text{Ra}}$ (d)</td>
<td>$5.7$</td>
</tr>
<tr>
<td>Actinium</td>
<td>$^{227}_{\text{Ac}}$ (d)</td>
<td>$21.5$</td>
</tr>
<tr>
<td>Thorium</td>
<td>$^{228}_{\text{Th}}$ (d)</td>
<td>$1.9$</td>
</tr>
<tr>
<td>Thorium</td>
<td>$^{228m}_{\text{Th}}$</td>
<td>$7.3 \times 10^3$</td>
</tr>
<tr>
<td>Thorium</td>
<td>$^{230}_{\text{Th}}$</td>
<td>$8.0 \times 10^4$</td>
</tr>
<tr>
<td>Thorium</td>
<td>$^{232}_{\text{Th}}$</td>
<td>$1.4 \times 10^4$</td>
</tr>
<tr>
<td>Protoactinium</td>
<td>$^{231}_{\text{Pa}}$</td>
<td>$3.3 \times 10^4$</td>
</tr>
<tr>
<td>Uranium</td>
<td>$^{232}_{\text{U}}$ (d)</td>
<td>$70$</td>
</tr>
<tr>
<td>Uranium</td>
<td>$^{233}_{\text{U}}$</td>
<td>$1.6 \times 10^3$</td>
</tr>
<tr>
<td>Uranium</td>
<td>$^{234}_{\text{U}}$</td>
<td>$2.6 \times 10^5$</td>
</tr>
<tr>
<td>Uranium</td>
<td>$^{235}_{\text{U}}$</td>
<td>$7.1 \times 10^4$</td>
</tr>
<tr>
<td>Uranium</td>
<td>$^{236}_{\text{U}}$</td>
<td>$2.4 \times 10^7$</td>
</tr>
<tr>
<td>Uranium</td>
<td>$^{238}_{\text{U}}$</td>
<td>$4.5 \times 10^4$</td>
</tr>
<tr>
<td>Neptunium</td>
<td>$^{237}_{\text{Np}}$</td>
<td>$2.1 \times 10^5$</td>
</tr>
<tr>
<td>Plutonium</td>
<td>$^{238}_{\text{Pu}}$ (d)</td>
<td>$89.0$</td>
</tr>
<tr>
<td>Plutonium</td>
<td>$^{239}_{\text{Pu}}$</td>
<td>$2.4 \times 10^4$</td>
</tr>
<tr>
<td>Plutonium</td>
<td>$^{240}_{\text{Pu}}$ (d)</td>
<td>$6.8 \times 10^3$</td>
</tr>
<tr>
<td>Plutonium</td>
<td>$^{241}_{\text{Pu}}$ (d)</td>
<td>$14.6$</td>
</tr>
<tr>
<td>Plutonium</td>
<td>$^{242}_{\text{Pu}}$</td>
<td>$3.8 \times 10^5$</td>
</tr>
<tr>
<td>Plutonium</td>
<td>$^{244}_{\text{Pu}}$</td>
<td>$8.3 \times 10^4$</td>
</tr>
<tr>
<td>Americium</td>
<td>$^{241}_{\text{Am}}$</td>
<td>$4.3 \times 10^2$</td>
</tr>
<tr>
<td>Americium</td>
<td>$^{242m}_{\text{Am}}$</td>
<td>$1.41 \times 10^2$</td>
</tr>
<tr>
<td>Americium</td>
<td>$^{242}_{\text{Am}}$ (d)</td>
<td>$16$ hrs</td>
</tr>
<tr>
<td>Curium</td>
<td>$^{242}_{\text{Cm}}$ (d)</td>
<td>$163$ days</td>
</tr>
<tr>
<td>Curium</td>
<td>$^{243}_{\text{Cm}}$ (d)</td>
<td>$38$</td>
</tr>
<tr>
<td>Curium</td>
<td>$^{246}_{\text{Cm}}$ (d)</td>
<td>$8.5 \times 10^3$</td>
</tr>
<tr>
<td>Curium</td>
<td>$^{247}_{\text{Cm}}$</td>
<td>$4.7 \times 10^6$</td>
</tr>
<tr>
<td>Curium</td>
<td>$^{248}_{\text{Cm}}$ (d)</td>
<td>$1.8 \times 10^7$</td>
</tr>
<tr>
<td>Curium</td>
<td>$^{248}_{\text{Cm}}$</td>
<td>$1.7 \times 10^4$</td>
</tr>
<tr>
<td>Curium</td>
<td>$^{250}_{\text{Cm}}$</td>
<td>$3.5 \times 10^5$</td>
</tr>
<tr>
<td>Californium</td>
<td>$^{249}_{\text{Cf}}$</td>
<td>$13.1$</td>
</tr>
<tr>
<td>Californium</td>
<td>$^{250}_{\text{Cf}}$ (d)</td>
<td>$3.5 \times 10^2$</td>
</tr>
<tr>
<td>Californium</td>
<td>$^{251}_{\text{Cf}}$</td>
<td>$8.0 \times 10^2$</td>
</tr>
</tbody>
</table>

### Notes:

1. Radionuclides listed are those present in 1000 year old waste. Only those with a half-life greater than 30 days, and significant activity levels are included.
2. (d) indicates a short-lived daughter product of a radioactive decay chain.
3. * Iodine is a volatile fission product, most of which is removed during reprocessing. Nevertheless it constitutes waste and is therefore included in the inventory.
4. Information compiled from a variety of international sources.
## TABLE 4  RISK FACTORS FOR FATAL CANCERS OR SERIOUS HEREDITARY DEFECTS

<table>
<thead>
<tr>
<th>TISSUE OR ORGAN</th>
<th>HEALTH DEFECT</th>
<th>RISK FACTOR</th>
<th>FRACTIONAL CONTRIBUTION (risk weighting factor)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>Risk</td>
<td>$(Sv)^{-1}$</td>
</tr>
<tr>
<td>Gonads</td>
<td>Serious hereditary+ effect in first two generations</td>
<td>1 in 250</td>
<td>$4 \times 10^{-3}$</td>
</tr>
<tr>
<td>Breast</td>
<td>Fatal cancer</td>
<td>1 in 400</td>
<td>$2.5 \times 10^{-3}$</td>
</tr>
<tr>
<td>Red bone marrow</td>
<td>&quot; &quot; &quot; &quot;</td>
<td>1 in 500</td>
<td>$2 \times 10^{-3}$</td>
</tr>
<tr>
<td>Lung</td>
<td>&quot; &quot; &quot; &quot;</td>
<td>1 in 500</td>
<td>$2 \times 10^{-3}$</td>
</tr>
<tr>
<td>Thyroid</td>
<td>&quot; &quot; &quot; &quot;</td>
<td>1 in 2000</td>
<td>$5 \times 10^{-4}$</td>
</tr>
<tr>
<td>Bone surfaces</td>
<td>&quot; &quot; &quot; &quot;</td>
<td>1 in 2000</td>
<td>$5 \times 10^{-4}$</td>
</tr>
<tr>
<td>Remainder</td>
<td>&quot; &quot; &quot; &quot;</td>
<td>1 in 200</td>
<td>$5 \times 10^{-3}$</td>
</tr>
<tr>
<td>Whole body</td>
<td>Either of above</td>
<td>1 in 60</td>
<td>$1.65 \times 10^{-2}$</td>
</tr>
</tbody>
</table>

* Adapted from 'Living with Radiation'; 2nd edition, NRPB; 1981
+ Note: The average risk is reduced by a factor of about 0.4 since only a fraction of the population would actually produce children.
<table>
<thead>
<tr>
<th>Generic Rock Properties</th>
<th>Crystalline (igneous, intrusive)</th>
<th>Argillaceous</th>
<th>Saliferous</th>
</tr>
</thead>
<tbody>
<tr>
<td>Typical intact permeability, $k$ (m/sec)</td>
<td>low to very low $10^{-11}$ to $10^{-9}$</td>
<td>low to very low $10^{-12}$ to $10^{-7}$</td>
<td>extremely low $10^{-12}$ to $10^{-10}$</td>
</tr>
<tr>
<td>Typical bulk permeability, $k$ (m/sec)</td>
<td>depends on nature and extent of fissures</td>
<td>low $10^{-7}$</td>
<td>very low $10^{-10}$</td>
</tr>
<tr>
<td>Thermal conductivity (W/m°C)</td>
<td>medium 2.5 to 3.0</td>
<td>low to medium 0.5 to 2.5</td>
<td>high 5.0 to 8.0</td>
</tr>
<tr>
<td>Sorption capacity</td>
<td>depends on nature and extent of fracture infilling</td>
<td>very high</td>
<td>very low nil for pure salt</td>
</tr>
<tr>
<td>Uniformity</td>
<td>fairly high at m scale; variable at larger scales</td>
<td>high at m to km scale</td>
<td>fairly high at m scale</td>
</tr>
<tr>
<td>Solubility</td>
<td>very low</td>
<td>low</td>
<td>very high</td>
</tr>
<tr>
<td>Susceptibility to weathering</td>
<td>weathers slowly at the surface and along fractures</td>
<td>very low at depth; very high at surface</td>
<td>potentially soluble at the margins</td>
</tr>
<tr>
<td>Plasticity</td>
<td>low</td>
<td>potentially very high depends on degree of induration and construction depth</td>
<td>high</td>
</tr>
<tr>
<td>Stability/stand-up time during construction</td>
<td>very high</td>
<td>very low to medium</td>
<td>high</td>
</tr>
<tr>
<td>Intrinsic stability over geological time</td>
<td>high</td>
<td>high</td>
<td>salt beds: high</td>
</tr>
<tr>
<td>Potential mineral resource value</td>
<td>low</td>
<td>low</td>
<td>salt domes (a) recent origin; low (b) ancient origin; high</td>
</tr>
</tbody>
</table>


| TABLE 6 CLASSIFICATION SYSTEM FOR ARGILLACEOUS ROCKS  (Adapted from Stow DAV; 1981). |
|--------------------------------|---------------------------------|--------------------------------|-----------------|-----------------|-----------------|
| a.  **UNMETAMORPHOSED**     |                                  |                                |                  |                  |                  |
| UNLITHIFIED          | LITHIFIED (NON-FISSILE)           | LITHIFIED (FISSILE)            | APPROX GRAIN-SIZE|                  |                  |
| EQUIVALENT           |                                |                                | PROPORTIONS      |                  |                  |
| silt                | siltstone                       | silt-shale                     | >2/3 silt-sized  |                  |                  |
| mud                 | mudstone                        | mud-shale                      | silt/clay mixture|                  |                  |
| clay                | claystone                       | clay-shale                     | >2/3 clay-sized  |                  |                  |
| b.  **METAMORPHOSED**  |                                  |                                |                  |                  |                  |
| TYPE                | DEGREE OF METAMORPHISM           | COMPOSITION                    |                  |                  |                  |
| argillite           | slightly metamorphosed/non-fissile| silt/clay mixture             |                  |                  |                  |
| slate               | metamorphosed/fissile            | silt/clay mixture             |                  |                  |                  |
| c.  **COMPOSITIONAL DESCRIPTIONS** |                                |                                |                  |                  |                  |
| TERM                | APPROX PROPORTIONS               |                                |                  |                  |                  |
| calcareous*         | 10% CaCO₃                       |                                |                  |                  |                  |
| siliceous           | 10% SiO₂                        |                                |                  |                  |                  |
| carbonaceous        | 1% Organic carbon               |                                |                  |                  |                  |

*where carbonate content exceeds 30%, the rock may be termed a marl.
### Table 7: Characteristic Chemical Compositions of Host Rock Groundwaters

(Based on standard inventories adopted by CEC corrosion study group).

<table>
<thead>
<tr>
<th>Pore Water Type</th>
<th>Typical Species</th>
<th>Concentration (meq/l)</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Granitic</strong></td>
<td>Na&lt;sup&gt;+&lt;/sup&gt;</td>
<td>0.35 - 4.60</td>
</tr>
<tr>
<td></td>
<td>K&lt;sup&gt;+&lt;/sup&gt;</td>
<td>0.03 - 0.13</td>
</tr>
<tr>
<td></td>
<td>Ca&lt;sup&gt;2+&lt;/sup&gt;</td>
<td>0.08 - 1.00</td>
</tr>
<tr>
<td></td>
<td>Mg&lt;sup&gt;2+&lt;/sup&gt;</td>
<td>0.10 - 0.50</td>
</tr>
<tr>
<td></td>
<td>Cl&lt;sup&gt;-&lt;/sup&gt;</td>
<td>0.21 - 1.00</td>
</tr>
<tr>
<td></td>
<td>SO&lt;sub&gt;4&lt;/sub&gt;&lt;sup&gt;-2&lt;/sup&gt;</td>
<td>0.10 - 0.50</td>
</tr>
<tr>
<td></td>
<td>Si&lt;sub&gt;2&lt;/sub&gt;</td>
<td>0.16 - 0.83*</td>
</tr>
<tr>
<td></td>
<td>K&lt;sub&gt;2&lt;/sub&gt;SO&lt;sub&gt;4&lt;/sub&gt;</td>
<td>0.187</td>
</tr>
<tr>
<td></td>
<td>MgSO&lt;sub&gt;4&lt;/sub&gt;</td>
<td>0.021</td>
</tr>
<tr>
<td><strong>Argillaceous</strong></td>
<td>Na&lt;sub&gt;2&lt;/sub&gt;SO&lt;sub&gt;4&lt;/sub&gt;</td>
<td>0.20 - 1.207</td>
</tr>
<tr>
<td>(varies according</td>
<td>NaCl</td>
<td>0.058 - 0.60</td>
</tr>
<tr>
<td>to Eh/pH)</td>
<td>Na&lt;sub&gt;2&lt;/sub&gt;CO&lt;sub&gt;3&lt;/sub&gt;</td>
<td>0.339 - 0.72</td>
</tr>
<tr>
<td></td>
<td>NaHCO&lt;sub&gt;3&lt;/sub&gt;</td>
<td>1.529 - 20.0</td>
</tr>
<tr>
<td></td>
<td>CaCO&lt;sub&gt;3&lt;/sub&gt;</td>
<td>saturation</td>
</tr>
<tr>
<td></td>
<td>Na&lt;sup&gt;+&lt;/sup&gt;</td>
<td>saturation</td>
</tr>
<tr>
<td></td>
<td>K&lt;sup&gt;+&lt;/sup&gt;</td>
<td>&quot;</td>
</tr>
<tr>
<td></td>
<td>Ca&lt;sup&gt;2+&lt;/sup&gt;</td>
<td>&quot;</td>
</tr>
<tr>
<td><strong>Saliferous</strong></td>
<td>Mg&lt;sup&gt;2+&lt;/sup&gt;</td>
<td>&quot;</td>
</tr>
<tr>
<td></td>
<td>Cl&lt;sup&gt;-&lt;/sup&gt;</td>
<td>&quot;</td>
</tr>
<tr>
<td></td>
<td>SO&lt;sub&gt;4&lt;/sub&gt;&lt;sup&gt;-2&lt;/sup&gt;</td>
<td>&quot;</td>
</tr>
<tr>
<td></td>
<td>CO&lt;sub&gt;3&lt;/sub&gt;&lt;sup&gt;-2&lt;/sup&gt;</td>
<td>&quot;</td>
</tr>
<tr>
<td></td>
<td>HCO&lt;sub&gt;3&lt;/sub&gt;</td>
<td>&quot;</td>
</tr>
<tr>
<td></td>
<td>F&lt;sup&gt;-&lt;/sup&gt;</td>
<td>&quot;</td>
</tr>
</tbody>
</table>

* Upper bound value from measurements in the Canadian Shield
<table>
<thead>
<tr>
<th>Radionuclide</th>
<th>Behavior of radioactive nuclide in pure water (only anion present OH⁻)</th>
<th>Type of radionuclide</th>
<th>Approx. radionuclide half-life (years)</th>
<th>Behavior of radioactive nuclide in pure water (only anion present OH⁻)</th>
<th>Type of radionuclide</th>
<th>Approx. radionuclide half-life (years)</th>
</tr>
</thead>
<tbody>
<tr>
<td>²⁷³⁹</td>
<td>Highly mobile cation soluble complex ion</td>
<td>Fission P</td>
<td>6.9 x 10⁵</td>
<td>Insoluble Hydroxide</td>
<td>Fission P</td>
<td>1.5 x 10⁶</td>
</tr>
<tr>
<td>²⁷⁸⁹</td>
<td>Highly soluble cation</td>
<td>Fission P</td>
<td>6.2 x 10⁵</td>
<td>Insoluble Hydroxide</td>
<td>Fission P</td>
<td>2.0 x 10⁵</td>
</tr>
<tr>
<td>²⁷⁰⁷</td>
<td>Highly soluble cation</td>
<td>Fission P</td>
<td>4.2 x 10⁵</td>
<td>Insoluble Hydroxide</td>
<td>Fission P</td>
<td>6.5 x 10⁵</td>
</tr>
<tr>
<td>²⁷²⁵</td>
<td>Highly soluble cation</td>
<td>Fission P</td>
<td>1.5 x 10⁹</td>
<td>Insoluble Hydroxide</td>
<td>Fission P</td>
<td>3.0 x 10⁹</td>
</tr>
</tbody>
</table>

Notes:
- Fission P: Fission product
- Decay product in a radioactive decay chain series
- Not a decay product, usually a decay series precursor formed in nuclear reactors

**TABLE 8 IONIC POTENTIALS AND SOLUBILITY LEVELS OF RADIONUCLIDES IN THE AQUEOUS SYSTEM**
## Table 9: Heat Transfer Properties of Typical Host Rock, Waste Unit and Backfill Materials.

<table>
<thead>
<tr>
<th>Repository Component</th>
<th>Material</th>
<th>Density $t/m^3$</th>
<th>Specific Heat $J/Kg \ °C$</th>
<th>Thermal Conductivity $W/m \ °C$</th>
<th>Thermal Diffusivity $m^2/s \times 10^6$</th>
<th>Reference</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>HOST ROCK</strong></td>
<td>Crystalline</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>- granite</td>
<td>2.5 - 3.1</td>
<td>-</td>
<td>2.8 - 3.6</td>
<td>-</td>
<td>47,125</td>
</tr>
<tr>
<td></td>
<td>- gneiss</td>
<td>2.5 - 2.8</td>
<td>-</td>
<td>1.9 - 4.8</td>
<td>-</td>
<td>47,125</td>
</tr>
<tr>
<td></td>
<td>Argillaceous</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>- shale</td>
<td>2.06 - 2.66</td>
<td>-</td>
<td>1.1 - 2.4</td>
<td>-</td>
<td>47,125</td>
</tr>
<tr>
<td></td>
<td>- argillite</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>Saliferous</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>- halite</td>
<td>1.98 - 2.82</td>
<td>-</td>
<td>5.3 - 7.2</td>
<td>-</td>
<td>47,125</td>
</tr>
<tr>
<td></td>
<td>- anhydrite</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td><strong>WASTE UNIT</strong></td>
<td>Borosilicate glass</td>
<td>2.6 - 2.8</td>
<td>-</td>
<td>1.2 - 1.4</td>
<td>-</td>
<td>125</td>
</tr>
<tr>
<td></td>
<td>Cast steel</td>
<td>7.65</td>
<td>490</td>
<td>46.0</td>
<td>12.0</td>
<td>37,125</td>
</tr>
<tr>
<td></td>
<td>Stainless steel</td>
<td>7.7 - 7.9</td>
<td>490</td>
<td>14.5 - 25.0</td>
<td>3.75 - 6.63</td>
<td>37,125</td>
</tr>
<tr>
<td></td>
<td>Lead</td>
<td>11.34</td>
<td>126</td>
<td>35.1</td>
<td>24.5</td>
<td>125</td>
</tr>
<tr>
<td></td>
<td>Copper</td>
<td>8.94</td>
<td>302</td>
<td>309.1</td>
<td>114.9</td>
<td>47,125</td>
</tr>
<tr>
<td></td>
<td>Titanium</td>
<td>4.51</td>
<td>528</td>
<td>17.2</td>
<td>7.22</td>
<td>125</td>
</tr>
<tr>
<td><strong>BACKFILL</strong></td>
<td>Clay</td>
<td>1.7 - 2.4</td>
<td>910 - 980</td>
<td>0.3 - 2.0</td>
<td>0.15 - 1.24</td>
<td>125</td>
</tr>
<tr>
<td></td>
<td>Quartz crystal</td>
<td>2.65</td>
<td>-</td>
<td>5.56 - 13.60</td>
<td>-</td>
<td>47</td>
</tr>
<tr>
<td></td>
<td>Quartz aggregate</td>
<td>1.82</td>
<td>-</td>
<td>2.22</td>
<td>-</td>
<td>47</td>
</tr>
<tr>
<td></td>
<td>90% Quartz/10% Bentonite</td>
<td>2.50</td>
<td>-</td>
<td>0.5 - 2.0</td>
<td>-</td>
<td>177</td>
</tr>
<tr>
<td></td>
<td>Cement Paste</td>
<td>2.4 - 2.6</td>
<td>840 - 1260</td>
<td>0.5 - 0.7</td>
<td>-</td>
<td>60</td>
</tr>
<tr>
<td></td>
<td>Concrete</td>
<td>2.2 - 3.9</td>
<td>840 - 1050</td>
<td>1.44 - 3.60</td>
<td>0.42 - 1.55</td>
<td>37,60,152</td>
</tr>
<tr>
<td></td>
<td>Graphite</td>
<td>2.25</td>
<td>680</td>
<td>160.0</td>
<td>104.5</td>
<td></td>
</tr>
<tr>
<td>COUNTRY</td>
<td>HOST ROCK TYPE</td>
<td>REPOSITORY SYSTEM</td>
<td>NUCLEAR AUTHORITY</td>
<td>DESIGN BODY</td>
<td>REFERENCE</td>
<td></td>
</tr>
<tr>
<td>-------------</td>
<td>----------------</td>
<td>-------------------</td>
<td>----------------------------------</td>
<td>--------------------------------------</td>
<td>-----------------</td>
<td></td>
</tr>
<tr>
<td>Canada</td>
<td>Crystalline</td>
<td>in-floor</td>
<td>in-room</td>
<td>AECL/ONTARIO HYDRO</td>
<td>1, 2, 35</td>
<td></td>
</tr>
<tr>
<td>France</td>
<td>Crystalline*</td>
<td>in-floor</td>
<td>—</td>
<td>CEA</td>
<td>16A</td>
<td></td>
</tr>
<tr>
<td>Sweden</td>
<td>Crystalline</td>
<td>in-floor</td>
<td>in-floor and mined cavern</td>
<td>KBS</td>
<td>122,123,138,192</td>
<td></td>
</tr>
<tr>
<td>United</td>
<td>Crystalline*</td>
<td>in-floor</td>
<td>—</td>
<td>AERE</td>
<td>144</td>
<td></td>
</tr>
<tr>
<td>Kingdom</td>
<td>United</td>
<td>Crystalline*</td>
<td>in-floor</td>
<td>AEC</td>
<td>159</td>
<td></td>
</tr>
<tr>
<td>United</td>
<td>States</td>
<td>Clay</td>
<td>in-floor</td>
<td>CEN/SCK</td>
<td>52, 97, 98</td>
<td></td>
</tr>
<tr>
<td>States</td>
<td>Clay</td>
<td>in-floor</td>
<td>in-room and in-floor</td>
<td>CEN</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Denmark</td>
<td>Saliferous</td>
<td>matrix of drill-holes</td>
<td>—</td>
<td>ELSAM/ELKRAFT</td>
<td>67</td>
<td></td>
</tr>
<tr>
<td>Netherlands</td>
<td>Saliferous</td>
<td>in-floor</td>
<td>mined cavern</td>
<td>ECN</td>
<td>92, 91</td>
<td></td>
</tr>
<tr>
<td>W. Germany</td>
<td>Saliferous</td>
<td>in-floor</td>
<td>in-floor and mined cavern</td>
<td>ECN</td>
<td>188, 52</td>
<td></td>
</tr>
</tbody>
</table>

* In these countries, other host rock types are also under consideration.
<table>
<thead>
<tr>
<th>Country</th>
<th>Approximate Design Production Period (Years)</th>
<th>Assumed Maximum Installed Nuclear Capacity GW(e)</th>
<th>Approximate Number of Waste Units Produced</th>
<th>Size and Shape of Waste Unit</th>
<th>Approximate Weight of Each Complete Unit (kg)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Canada</td>
<td>36</td>
<td>194</td>
<td>186,000</td>
<td>Canister: 0.46 D, 3.28 L; Canister &amp; Overpack: - D, - L</td>
<td>1675</td>
</tr>
<tr>
<td>France</td>
<td>30</td>
<td>45</td>
<td>30,000</td>
<td>Canister: 0.36 D, 1.92 L; Canister &amp; Overpack: - D, - L</td>
<td>500</td>
</tr>
<tr>
<td>Sweden</td>
<td>30</td>
<td>30</td>
<td>9,000</td>
<td>Canister: 0.40 D, 1.50 L; Canister &amp; Overpack: 0.61 D, 1.80 L</td>
<td>3900</td>
</tr>
<tr>
<td>United Kingdom</td>
<td>20</td>
<td>48</td>
<td>4,000</td>
<td>Canister: 0.50 D, 3.00 L; Canister &amp; Overpack: - D, - L</td>
<td>1400</td>
</tr>
<tr>
<td>United States</td>
<td>28</td>
<td>487</td>
<td>87,300</td>
<td>Canister: 0.33 D, 3.05 L; Canister &amp; Overpack: - D, - L</td>
<td>780</td>
</tr>
<tr>
<td>Country</td>
<td>Volume of Glass Per Canister (cu m)</td>
<td>Period of Storage Prior to Disposal (years ex-reactor)</td>
<td>Proportion of Fission Products in Glass Matrix (% wt)</td>
<td>Heat Output at Disposal (kW)</td>
<td></td>
</tr>
<tr>
<td>---------------</td>
<td>-----------------------------------</td>
<td>--------------------------------------------------------</td>
<td>------------------------------------------------------</td>
<td>-----------------------------</td>
<td></td>
</tr>
<tr>
<td>Canada</td>
<td>0.43</td>
<td>10</td>
<td>1%</td>
<td>0.27</td>
<td></td>
</tr>
<tr>
<td>France</td>
<td>0.14</td>
<td>30</td>
<td>12.8%</td>
<td>0.90</td>
<td></td>
</tr>
<tr>
<td>Sweden</td>
<td>0.15</td>
<td>40</td>
<td>9%</td>
<td>0.52</td>
<td></td>
</tr>
<tr>
<td>United Kingdom</td>
<td>0.39</td>
<td>60</td>
<td>15%</td>
<td>1.00</td>
<td></td>
</tr>
<tr>
<td>United States</td>
<td>0.18</td>
<td>10</td>
<td>15%</td>
<td>1.70</td>
<td></td>
</tr>
<tr>
<td>Country</td>
<td>Total Number of Units</td>
<td>Mass of Fission Products in each Unit (kg)</td>
<td>Heat Output per Unit at Disposal (kW)</td>
<td>Curved Surface Area of each Unit (sq m)</td>
<td>Heat Output per Unit Surface Area (W/sq m)</td>
</tr>
<tr>
<td>---------------</td>
<td>-----------------------</td>
<td>------------------------------------------</td>
<td>--------------------------------------</td>
<td>----------------------------------------</td>
<td>-------------------------------------------</td>
</tr>
<tr>
<td>Canada</td>
<td>186,000</td>
<td>12</td>
<td>0.27</td>
<td>4.70</td>
<td>57</td>
</tr>
<tr>
<td>France</td>
<td>30,000</td>
<td>48</td>
<td>0.90</td>
<td>2.17</td>
<td>415</td>
</tr>
<tr>
<td>Sweden</td>
<td>9,000</td>
<td>36</td>
<td>0.52</td>
<td>3.45</td>
<td>150</td>
</tr>
<tr>
<td>United Kingdom</td>
<td>4,000</td>
<td>158</td>
<td>1.00</td>
<td>4.71</td>
<td>212</td>
</tr>
<tr>
<td>United States</td>
<td>87,300</td>
<td>73</td>
<td>1.70</td>
<td>3.16</td>
<td>538</td>
</tr>
</tbody>
</table>

Notes:  
1. Column b derived from Tables 11 and 12, assuming average glass specific gravity = 2.7; thus b = 2.7 x vol. glass per canister x % wt fission products.  
2. In Column c, curved surface area of Swedish unit includes overpack.  
3. Values in Column d indicate the area over which heat is dissipated to the surrounding rock.
### Table 14: High-Level Waste Forms and Characteristics; Comparisons of Design Emphasis in Crystalline Rock Repository Proposals

<table>
<thead>
<tr>
<th>Country</th>
<th>Repository Geometry</th>
<th>Degree of Confinement</th>
<th>Ease of Handling as function of canister weight</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Degree of Overall Compactness as function of heat output per unit mass of fission products</td>
<td>Waste Dilution based on % wt of fission products</td>
<td>Provision of Overpack</td>
</tr>
<tr>
<td>Canada</td>
<td>4</td>
<td>1</td>
<td>3.5</td>
</tr>
<tr>
<td>France</td>
<td>3</td>
<td>4</td>
<td>3.5</td>
</tr>
<tr>
<td>Sweden</td>
<td>2</td>
<td>2</td>
<td>1</td>
</tr>
<tr>
<td>United Kingdom</td>
<td>1</td>
<td>3</td>
<td>4.5</td>
</tr>
<tr>
<td>United States</td>
<td>5</td>
<td>5</td>
<td>4.5</td>
</tr>
<tr>
<td>Country</td>
<td>Host Rock</td>
<td>Size &amp; Shape of Waste Units</td>
<td>Proportion of Fission Products % by wt</td>
</tr>
<tr>
<td>-------------</td>
<td>-----------</td>
<td>-----------------------------</td>
<td>---------------------------------------</td>
</tr>
<tr>
<td>Belgium</td>
<td>Clay</td>
<td>0.3</td>
<td>12.8</td>
</tr>
<tr>
<td>Netherlands</td>
<td>Saliferous</td>
<td>0.20</td>
<td>*</td>
</tr>
<tr>
<td>Denmark</td>
<td>Saliferous</td>
<td>0.66</td>
<td>9</td>
</tr>
<tr>
<td>W. Germany</td>
<td>Saliferous</td>
<td>0.22</td>
<td>20</td>
</tr>
</tbody>
</table>

* Not found in literature
TABLE 16 COMPARISON OF ASSUMED CONSTRUCTION DEPTHS, THERMAL ROCK PROPERTIES AND TEMPERATURE DESIGN CONSTRAINTS FOR CRYSTALLINE ROCK HIGH-LEVEL WASTE REPOSITORY PROPOSALS

<table>
<thead>
<tr>
<th>Country</th>
<th>Assumed Depth of Emplacement Tunnels</th>
<th>Assumed Geothermal Gradient °C/km</th>
<th>Ambient Temperature at Emplacement Horizon °C</th>
<th>Approximate Maximum Rock Temperature Rise due to Waste °C</th>
<th>Approximate Maximum Rock Temperature °C</th>
<th>Assumed Rock Thermal Conductivity W/m°C</th>
<th>Assumed Coefficient of Thermal Expansion per °C</th>
</tr>
</thead>
<tbody>
<tr>
<td>Canada</td>
<td>1000</td>
<td>15</td>
<td>15</td>
<td>135**</td>
<td>150**</td>
<td>3.0 (granite)</td>
<td>6–8 x 10⁻⁶</td>
</tr>
<tr>
<td>France</td>
<td>1000–1400*</td>
<td>30</td>
<td>40</td>
<td>75</td>
<td>115</td>
<td>2.5</td>
<td>8 x 10⁻⁶</td>
</tr>
<tr>
<td>Sweden</td>
<td>500–600</td>
<td>15</td>
<td>15</td>
<td>65</td>
<td>80</td>
<td>3.0</td>
<td>8 x 10⁻⁶</td>
</tr>
<tr>
<td>United Kingdom</td>
<td>300–600*</td>
<td>30</td>
<td>20–30</td>
<td>80</td>
<td>100</td>
<td>2.51</td>
<td>8 x 10⁻⁶</td>
</tr>
<tr>
<td>United States</td>
<td>610</td>
<td>not known</td>
<td>not known</td>
<td>not known</td>
<td>260</td>
<td>2.63</td>
<td>9.5 x 10⁻⁶</td>
</tr>
</tbody>
</table>

* emplacement holes up to 300m deep (cuboidal configuration)

** actual configuration is estimated to induce maximum rock temperatures of 87°C and 102°C for granite and gabbro respectively.
### Table 17: Proposed Repository Construction Depths, Rock Strength Properties and Implied Ground Stability Conditions

<table>
<thead>
<tr>
<th>Country</th>
<th>Host Rock Type</th>
<th>Construction Depth Range (m)</th>
<th>Vertical In situ Stress $\sigma_v$ (MN/m²)</th>
<th>Assumed Unconfined Strength $\sigma_c$ (MN/m²)</th>
<th>Assumed $K_o$</th>
<th>Implied Rock Competence Factor/Overload Factor</th>
</tr>
</thead>
<tbody>
<tr>
<td>Canada</td>
<td>Crystalline</td>
<td>1000</td>
<td>30*</td>
<td>138 - 275</td>
<td>1.5</td>
<td>4.6 - 9.2</td>
</tr>
<tr>
<td>France</td>
<td>Crystalline</td>
<td>1000-1300</td>
<td>30 - 39*</td>
<td>150</td>
<td>0.7 - 1.3</td>
<td>3.8 - 5.0</td>
</tr>
<tr>
<td>Sweden</td>
<td>Crystalline</td>
<td>500-600</td>
<td>15 - 18*</td>
<td>150*</td>
<td>--</td>
<td>8.3 - 10.0</td>
</tr>
<tr>
<td>United Kingdom</td>
<td>Crystalline</td>
<td>300-600</td>
<td>9 - 18*</td>
<td>150</td>
<td>--</td>
<td>8.3 - 16.7</td>
</tr>
<tr>
<td>United States</td>
<td>Crystalline</td>
<td>610</td>
<td>18.3*</td>
<td>150</td>
<td>--</td>
<td>8.2</td>
</tr>
<tr>
<td>Belgium</td>
<td>Clay</td>
<td>220-235</td>
<td>4.6 - 5.0*</td>
<td>Cu = 0.85*</td>
<td>--</td>
<td>F = 5.4 - 5.9</td>
</tr>
<tr>
<td>Denmark</td>
<td>Saliferous</td>
<td>1200-2500</td>
<td>28.8 - 60*</td>
<td>40*</td>
<td>1.0*</td>
<td>0.7 - 1.4</td>
</tr>
<tr>
<td>Netherlands</td>
<td>Saliferous</td>
<td>600-900</td>
<td>14.4 - 21.6*</td>
<td>40*</td>
<td>1.0*</td>
<td>1.8 - 2.8</td>
</tr>
<tr>
<td>W. Germany</td>
<td>Saliferous</td>
<td>830-1200</td>
<td>20 - 28.8*</td>
<td>40*</td>
<td>1.0*</td>
<td>1.4 - 2.0</td>
</tr>
</tbody>
</table>

* typical values inserted by the author in the absence of published information.
### a. HIGH-LEVEL WASTE REPOSITORIES

<table>
<thead>
<tr>
<th>Country</th>
<th>Source of Info.</th>
<th>Proposed Repository System</th>
<th>Host Rock Type</th>
<th>Depth of Burial (m)</th>
<th>Waste Units (m³)</th>
<th>Repository Excavations</th>
<th>Volumes (m³)</th>
<th>Ratio x/y</th>
</tr>
</thead>
<tbody>
<tr>
<td>France 16A</td>
<td>in-floor</td>
<td>Crystalline</td>
<td>1000–1100</td>
<td>8,880</td>
<td>1,360,000</td>
<td>294,000</td>
<td>5,246,000</td>
<td>476,000</td>
</tr>
<tr>
<td>UK 144</td>
<td>in-floor</td>
<td>Crystalline</td>
<td>300–1000</td>
<td>2,400</td>
<td>26,000</td>
<td>20,000</td>
<td>110,000</td>
<td>4,800</td>
</tr>
<tr>
<td>Belgium 124,52,166</td>
<td>in-floor</td>
<td>Argillaceous</td>
<td>220–235</td>
<td>954</td>
<td>11,000</td>
<td>12,100</td>
<td>94,260</td>
<td>3,036</td>
</tr>
<tr>
<td>Germany 52,188</td>
<td>in-floor</td>
<td>Saliferous</td>
<td>820–1200</td>
<td>3,360</td>
<td>171,000</td>
<td>600,000</td>
<td>400,000</td>
<td>12,200</td>
</tr>
<tr>
<td>Netherlands 90,91,92</td>
<td>in-floor</td>
<td>Saliferous</td>
<td>600–800</td>
<td>3,000</td>
<td>71,000</td>
<td>392,000</td>
<td>1,890,000</td>
<td>8,000</td>
</tr>
<tr>
<td>Denmark 67</td>
<td>matrix of drill-holes</td>
<td>Saliferous</td>
<td>1200–2500</td>
<td>886</td>
<td>None</td>
<td>None</td>
<td>None</td>
<td>3,930</td>
</tr>
<tr>
<td>Canada 2</td>
<td>in-floor</td>
<td>Crystalline</td>
<td>1000</td>
<td>92,600</td>
<td>1,000,000</td>
<td>2,956,000</td>
<td>268,000</td>
<td>4,067,000</td>
</tr>
<tr>
<td>Sweden 122,153</td>
<td>in-floor</td>
<td>Crystalline</td>
<td>500–600</td>
<td>4,730</td>
<td>20,000</td>
<td>77,000</td>
<td>429,000</td>
<td>35,000</td>
</tr>
<tr>
<td>U.S. 159</td>
<td>in-floor</td>
<td>Crystalline</td>
<td>820</td>
<td>22,774</td>
<td>151,000</td>
<td>773,700</td>
<td>6,273,000</td>
<td>104,400</td>
</tr>
</tbody>
</table>

### b. INTERMEDIATE-LEVEL WASTE REPOSITORIES

<table>
<thead>
<tr>
<th>Country</th>
<th>Source of Info.</th>
<th>Proposed Repository System</th>
<th>Host Rock Type</th>
<th>Depth of Burial (m)</th>
<th>Waste Units (m³)</th>
<th>Repository Excavations</th>
<th>Volumes (m³)</th>
<th>Ratio x/y</th>
</tr>
</thead>
<tbody>
<tr>
<td>France 124,52,166</td>
<td>in-room</td>
<td>Argillaceous</td>
<td>270</td>
<td>36,700</td>
<td>Common with HLW</td>
<td>Common with HLW</td>
<td>65,300</td>
<td>85,300</td>
</tr>
<tr>
<td>Germany 62,188</td>
<td>in-floor &amp; in-room</td>
<td>Saliferous</td>
<td>830–900</td>
<td>320,000</td>
<td>Common with HLW</td>
<td>600,000</td>
<td>1,330,000</td>
<td>190,000</td>
</tr>
<tr>
<td>Netherlands 90,91,92</td>
<td>in-floor</td>
<td>Saliferous</td>
<td>500–650</td>
<td>25,000</td>
<td>Common with HLW</td>
<td>168,000</td>
<td>none</td>
<td>384,000</td>
</tr>
<tr>
<td>Denmark 204</td>
<td>in-room</td>
<td>Argillaceous &amp; others</td>
<td>500</td>
<td>20,000</td>
<td>9,000</td>
<td>528,000</td>
<td>none</td>
<td>574,000</td>
</tr>
<tr>
<td>Canada 2</td>
<td>in-room</td>
<td>Crystalline</td>
<td>1,000</td>
<td>783,100</td>
<td>Common with HLW</td>
<td>571,000</td>
<td>1,460,000</td>
<td>none</td>
</tr>
<tr>
<td>Sweden 192</td>
<td>in-floor &amp; in room</td>
<td>Crystalline</td>
<td>80</td>
<td>60,000</td>
<td>Adit</td>
<td>68,660</td>
<td>401,000</td>
<td>none</td>
</tr>
</tbody>
</table>

**Key**: Information not available from current literature includes Tertiary sandstones and marls.

**Notes concerning ratio x/y**
1. Varies according to concentration of fission products in HLW, and period of interim storage.
2. Relatively high for matrix of drill-holes concept.
3. Comparatively high for ILW repositories due to absence of heat effects and sharing of common access shafts and tunnels.
### Table 19: High-Level Waste Form Combinations Adopted for Optimisation Study

<table>
<thead>
<tr>
<th>Characteristics</th>
<th>Canister Type (a)</th>
<th>Canister Type (b)</th>
<th>Canister Type (c)</th>
<th>Canister Type (d)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Half-AVM</td>
<td>Half-Harvest</td>
<td>AVM</td>
<td>Harvest</td>
</tr>
<tr>
<td>Overpack thickness (mm)</td>
<td>100 300</td>
<td>100 300</td>
<td>100 300</td>
<td>100 300</td>
</tr>
<tr>
<td>Overall diameter (m)</td>
<td>0.25 0.50 0.90</td>
<td>0.25 0.50 0.90</td>
<td>0.50 0.75 1.15</td>
<td>0.50 0.75 1.15</td>
</tr>
<tr>
<td>Overall length (m)</td>
<td>1.00 1.25 1.65</td>
<td>3.00 3.25 1.65</td>
<td>1.00 1.25 3.65</td>
<td>3.00 3.25 3.65</td>
</tr>
<tr>
<td>Overall mass (tonnes)</td>
<td>0.16 1.60 8.00</td>
<td>0.46 4.00 17.00</td>
<td>0.50 3.00 12.00</td>
<td>1.40 7.00 26.00</td>
</tr>
<tr>
<td>Glass length (m)</td>
<td>0.70 0.70 0.70</td>
<td>2.00 2.00 2.00</td>
<td>0.70 0.70 0.70</td>
<td>2.00 2.00 2.00</td>
</tr>
<tr>
<td>Glass volume (cu m)</td>
<td>0.034 0.034 0.034</td>
<td>0.10 0.10 0.10</td>
<td>0.14 0.14 0.14</td>
<td>0.40 0.40 0.40</td>
</tr>
<tr>
<td>Glass mass (tonnes)</td>
<td>0.09 0.09 0.09</td>
<td>0.26 0.26 0.26</td>
<td>0.36 0.36 0.36</td>
<td>1.04 1.04 1.04</td>
</tr>
<tr>
<td>WASTE UNIT TYPE</td>
<td>TOTAL № WASTE UNITS</td>
<td>STORAGE PERIOD (YRS)</td>
<td>HEAT OUTPUT (W)</td>
<td>HEAT FLUX (W/m²)</td>
</tr>
<tr>
<td>----------------</td>
<td>---------------------</td>
<td>---------------------</td>
<td>-----------------</td>
<td>------------------</td>
</tr>
<tr>
<td>1</td>
<td>44,445</td>
<td>33</td>
<td>167</td>
<td>212</td>
</tr>
<tr>
<td>2</td>
<td>44,445</td>
<td>33</td>
<td>167</td>
<td>212</td>
</tr>
<tr>
<td>3</td>
<td>44,445</td>
<td>4</td>
<td>415</td>
<td>217</td>
</tr>
<tr>
<td>4</td>
<td>15,385</td>
<td>30</td>
<td>500</td>
<td>217</td>
</tr>
<tr>
<td>5</td>
<td>15,385</td>
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<td>1,088</td>
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</tr>
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</table>
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| TABLE 22 REPOSITORY DESIGN OPTIMISATION; RESULTS FOR 60 YEAR STORAGE PERIOD |
| --- | --- | --- | --- | --- | --- | --- | --- | --- | --- |
| WASTE QUANTITIES AND CHARACTERISTICS | EMPLACEMENT HOUSES | EMPLACEMENT HOUSE AND TUNNEL SPACING | COLLING | CONSTRUCTION QUANTITIES | TOTAL | REPOSITORY SIZE |
| WASTE UNIT TYPE | TOTAL WASTE (MT) | STORAGE PERIOD (YR) | HEAT OUTPUT (MW) | HEAT FLUX (MW/m²) | CONFIG | DEPTH (m) | DIA (m) | H1 PER CENT | NO. HOLES REQUIRED | HOLE SPACING (m) | TUNNEL SPACING (m) | NO. HOLES REQU | TOTAL LENGTH (km) | TOTAL LENGTH (km) | TOTAL LENGTH (km) | VOLUME (m³ x 10⁶) | AERIAL DIMENSION (km) | CHARACTERS TOTAL | STORAGE LENGTH (km) |
| 1 | 44,445 | 60 | 95 | 120 | PLANTAR | 6.05 | 0.45 | 13 | 3.67 | 20 | 14,415 | 88.89 | 69 | 96.99 | 5.58 | 101.67 | 2.35 | 1.976 | 11.69 | 69.98 |
| 2 | 44,445 | 60 | 95 | 120 | PLANTAR | 6.75 | 0.30 | 13 | 2.55 | 20 | 14,415 | 100.00 | 64 | 30.64 | 3.57 | 42.34 | 1.10 | 1.872 | 5.37 | 187.44 |
| 3 | 44,445 | 60 | 95 | 120 | PLANTAR | 7.95 | 0.30 | 13 | 12.7 | 20 | 24 | 71 | 1.872 | 5.37 | 42.34 | 1.10 | 1.872 | 5.37 | 187.44 |
| 4 | 55,385 | 60 | 260 | 110 | PLANTAR | 6.00 | 0.45 | 15 | 37.4 | 20 | 15,385 | 92.32 | 67 | 91.3 | 5.45 | 96.56 | 2.43 | 1.872 | 5.37 | 187.44 |
| 5 | 55,385 | 60 | 260 | 110 | PLANTAR | 6.45 | 0.30 | 13 | 2.7 | 20 | 15,385 | 96.18 | 65 | 42.64 | 3.74 | 46.54 | 1.20 | 1.872 | 5.37 | 187.44 |
| 6 | 55,385 | 60 | 260 | 110 | PLANTAR | 6.85 | 0.30 | 13 | 19.0 | 20 | 15,385 | 100.00 | 64 | 28.06 | 3.46 | 32.32 | 0.63 | 1.872 | 5.37 | 187.44 |
| 7 | 11,110 | 60 | 380 | 120 | PLANTAR | 6.75 | 0.95 | 13 | 6.0 | 20 | 3,703 | 25.00 | 26 | 21.66 | 2.70 | 27.26 | 0.71 | 0.684 | 3.25 | 52.57 |
| 8 | 11,110 | 60 | 380 | 120 | PLANTAR | 7.95 | 0.13 | 28 | 21.2 | 20 | 3,703 | 25.00 | 26 | 21.66 | 2.70 | 27.26 | 0.71 | 0.684 | 3.25 | 52.57 |
| 9 | 11,110 | 60 | 380 | 120 | PLANTAR | 8.25 | 0.13 | 28 | 21.2 | 20 | 3,703 | 25.00 | 26 | 21.66 | 2.70 | 27.26 | 0.71 | 0.684 | 3.25 | 52.57 |
| 10 | 2,615 | 60 | 1000 | 212 | PLANTAR | 6.00 | 0.70 | 1 | 4.0 | 20 | 2,615 | 24.64 | 27 | 16.19 | 2.32 | 18.51 | 0.59 | 0.579 | 2.22 | 44.48 |
| 11 | 2,615 | 60 | 1000 | 212 | PLANTAR | 6.75 | 0.95 | 13 | 6.0 | 20 | 2,615 | 24.64 | 27 | 16.19 | 2.32 | 18.51 | 0.59 | 0.579 | 2.22 | 44.48 |
| 12 | 2,615 | 60 | 1000 | 212 | PLANTAR | 6.85 | 0.95 | 13 | 4.0 | 20 | 2,615 | 25.95 | 28 | 16.19 | 2.32 | 18.51 | 0.59 | 0.579 | 2.22 | 44.48 |
### Table 23: Repository Design Optimisation; Results for 90 Year Storage Period

<table>
<thead>
<tr>
<th>Waste Unit Type</th>
<th>Total No. Waste Units</th>
<th>Storage Time (Yrs)</th>
<th>Heat Unit (W)</th>
<th>Heat Flux (W/m²)</th>
<th>Waste Quantities and Characteristics</th>
<th>Emplacement Holes</th>
<th>Emplacement Hole and Tunnel Spacing</th>
<th>Construction Quantities</th>
<th>Total</th>
<th>Repository Size</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Total No. Waste Units</td>
<td>Storage Time (Yrs)</td>
<td>Heat Unit (W)</td>
<td>Heat Flux (W/m²)</td>
<td>Waste Quantities and Characteristics</td>
<td>Emplacement Holes</td>
<td>Emplacement Hole and Tunnel Spacing</td>
<td>Construction Quantities</td>
<td>Total</td>
<td>Repository Size</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

**Columns:**
- **Waste Unit Type**
- **Total No. Waste Units**
- **Storage Time (Yrs)**
- **Heat Unit (W)**
- **Heat Flux (W/m²)**
- **Waste Quantities and Characteristics**
- **Emplacement Holes**
  - **Depth (m)**
  - **Dia (m)**
  - **No. Units Holes**
  - **No. Holes**
  - **Hole Spacing (m)**
  - **Tunnel Spacing (m)**
- **Emplacement Hole and Tunnel Spacing**
  - **No. Holes**
  - **No. Exp. Tunnels**
  - **Total Length (m)**
  - **Total Length (m)**
  - **Total Length (m)**
- **Construction Quantities**
  - **No. Tunnels**
  - **Volume (m³)**
- **Repository Size**
  - **Areal Diam.**
  - **Characteristics**
  - **Total Volume (m³)**
  - **Indices**

**Rows:**
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2. [Data Entry]
3. [Data Entry]
4. [Data Entry]
5. [Data Entry]
6. [Data Entry]
7. [Data Entry]
8. [Data Entry]
9. [Data Entry]
10. [Data Entry]
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12. [Data Entry]
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<thead>
<tr>
<th>WASTE QUANTITIES</th>
<th>EMPLACEMENT HOLES</th>
<th>DRILLING</th>
<th>CONSTRUCTION QUANTITIES</th>
<th>TOTAL VOLUME</th>
<th>REPOSITORY SIZE</th>
</tr>
</thead>
<tbody>
<tr>
<td>UNIT TYPE</td>
<td>TOTAL # WASTE UNITS</td>
<td>STORAGE PERIOD (YRS)</td>
<td>HEAT OUTPUT (W)</td>
<td>HEAT FLUX (W/m²)</td>
<td>CONFIG</td>
</tr>
<tr>
<td>1</td>
<td>4,445</td>
<td>120</td>
<td>30</td>
<td>38</td>
<td>PLANAR</td>
</tr>
<tr>
<td>2</td>
<td>4,445</td>
<td>120</td>
<td>30</td>
<td>38</td>
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</tr>
<tr>
<td>3</td>
<td>4,445</td>
<td>120</td>
<td>30</td>
<td>38</td>
<td>CUBICAL</td>
</tr>
<tr>
<td>4</td>
<td>4,445</td>
<td>120</td>
<td>30</td>
<td>38</td>
<td>CUBICAL</td>
</tr>
<tr>
<td>5</td>
<td>15,385</td>
<td>120</td>
<td>60</td>
<td>94</td>
<td>PLANAR</td>
</tr>
<tr>
<td>6</td>
<td>15,385</td>
<td>120</td>
<td>60</td>
<td>94</td>
<td>CUBICAL</td>
</tr>
<tr>
<td>7</td>
<td>15,385</td>
<td>120</td>
<td>60</td>
<td>94</td>
<td>CUBICAL</td>
</tr>
<tr>
<td>8</td>
<td>15,385</td>
<td>120</td>
<td>60</td>
<td>94</td>
<td>CUBICAL</td>
</tr>
<tr>
<td>9</td>
<td>15,385</td>
<td>120</td>
<td>60</td>
<td>94</td>
<td>CUBICAL</td>
</tr>
<tr>
<td>10</td>
<td>15,385</td>
<td>120</td>
<td>60</td>
<td>94</td>
<td>CUBICAL</td>
</tr>
<tr>
<td>11</td>
<td>15,385</td>
<td>120</td>
<td>60</td>
<td>94</td>
<td>CUBICAL</td>
</tr>
<tr>
<td>12</td>
<td>15,385</td>
<td>120</td>
<td>60</td>
<td>94</td>
<td>CUBICAL</td>
</tr>
<tr>
<td>STORAGE PERIOD</td>
<td>EMPLACEMENT CONFIGURATION</td>
<td>CANISTER TYPE (a)</td>
<td>CANISTER TYPE (b)</td>
<td>CANISTER TYPE (c)</td>
<td>CANISTER TYPE (d)</td>
</tr>
<tr>
<td>----------------</td>
<td>--------------------------</td>
<td>-------------------</td>
<td>-------------------</td>
<td>-------------------</td>
<td>-------------------</td>
</tr>
<tr>
<td>MINIMUM (see table 20)</td>
<td>PLANAR</td>
<td>610</td>
<td>620</td>
<td>1025</td>
<td>392</td>
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<tr>
<td></td>
<td>CUBOIDAL; 50a</td>
<td>221</td>
<td>270</td>
<td>344</td>
<td>274</td>
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<tr>
<td></td>
<td>CUBOIDAL; 100a</td>
<td>138</td>
<td>232</td>
<td>232</td>
<td>232</td>
</tr>
<tr>
<td></td>
<td>CUBOIDAL; 300a</td>
<td>82</td>
<td>604</td>
<td>604</td>
<td>604</td>
</tr>
<tr>
<td>30 YEARS (see table 21)</td>
<td>PLANAR</td>
<td>409</td>
<td>629</td>
<td>604</td>
<td>532</td>
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<tr>
<td></td>
<td>CUBOIDAL; 50a</td>
<td>267</td>
<td>616</td>
<td>232</td>
<td>458</td>
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<td>CUBOIDAL; 100a</td>
<td>260</td>
<td>135</td>
<td>135</td>
<td>135</td>
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<td>CUBOIDAL; 300a</td>
<td>131</td>
<td>77</td>
<td>77</td>
<td>77</td>
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<tr>
<td>60 YEARS (see table 22)</td>
<td>PLANAR</td>
<td>379</td>
<td>589</td>
<td>375</td>
<td>484</td>
</tr>
<tr>
<td></td>
<td>CUBOIDAL; 50a</td>
<td>203</td>
<td>511</td>
<td>204</td>
<td>445</td>
</tr>
<tr>
<td></td>
<td>CUBOIDAL; 100a</td>
<td>139</td>
<td>139</td>
<td>139</td>
<td>139</td>
</tr>
<tr>
<td></td>
<td>CUBOIDAL; 300a</td>
<td>98</td>
<td>98</td>
<td>98</td>
<td>98</td>
</tr>
<tr>
<td>90 YEARS (see table 23)</td>
<td>PLANAR</td>
<td>279</td>
<td>592</td>
<td>263</td>
<td>492</td>
</tr>
<tr>
<td></td>
<td>CUBOIDAL; 50a</td>
<td>197</td>
<td>519</td>
<td>192</td>
<td>443</td>
</tr>
<tr>
<td></td>
<td>CUBOIDAL; 100a</td>
<td>149</td>
<td>149</td>
<td>149</td>
<td>149</td>
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<tr>
<td></td>
<td>CUBOIDAL; 300a</td>
<td>117</td>
<td>117</td>
<td>117</td>
<td>117</td>
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<tr>
<td>120 YEARS (see table 24)</td>
<td>PLANAR</td>
<td>245</td>
<td>630</td>
<td>198</td>
<td>545</td>
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<tr>
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<td>CUBOIDAL; 50a</td>
<td>195</td>
<td>567</td>
<td>165</td>
<td>175</td>
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<tr>
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<td>CUBOIDAL; 100a</td>
<td>175</td>
<td>175</td>
<td>175</td>
<td>175</td>
</tr>
<tr>
<td></td>
<td>CUBOIDAL; 300a</td>
<td>165</td>
<td>165</td>
<td>165</td>
<td>165</td>
</tr>
</tbody>
</table>

**TOTAL COST OF OVERPACKS, INTERIM STORAGE AND REPOSITORY CONSTRUCTION IN £ M (TOTAL EXCLUDING OVERPACKS SHOWN IN BRACKETS)**
### Table 26: Repository Design Optimisation; Waste Emplacement Rates

<table>
<thead>
<tr>
<th>Canister Type</th>
<th>Corresponding Waste Unit Types</th>
<th>Total Number of Waste Units</th>
<th>Delivery/Emplacement Rate (Units/Day)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td></td>
<td>Average</td>
</tr>
<tr>
<td>a</td>
<td>1, 2, 3</td>
<td>45,450</td>
<td>7</td>
</tr>
<tr>
<td>b</td>
<td>4, 5, 6</td>
<td>16,000</td>
<td>2/3</td>
</tr>
<tr>
<td>c</td>
<td>7, 8, 9</td>
<td>11,250</td>
<td>1/2</td>
</tr>
<tr>
<td>d</td>
<td>10, 11, 12</td>
<td>4,000</td>
<td>less than 1</td>
</tr>
</tbody>
</table>
TABLE 27 PRINCIPAL ATTRIBUTES OF CANDIDATE BACKFILL MATERIALS

<table>
<thead>
<tr>
<th>MATERIAL GROUP</th>
<th>PRINCIPAL ATTRIBUTES</th>
<th>MATERIAL TYPES</th>
</tr>
</thead>
<tbody>
<tr>
<td>SPOIL</td>
<td>b, e</td>
<td>crystalline rock clay argillaceous rock saliferous rock</td>
</tr>
<tr>
<td>CLAYS</td>
<td>b, c, d, e</td>
<td>illites kandites polygonolites smectites vermiculite chlorites</td>
</tr>
<tr>
<td>ZEOLITES</td>
<td>d</td>
<td>analcime chabazite erionite clinoptilolite ferrierite mordenite phillipsite</td>
</tr>
<tr>
<td>POZZOLANAS</td>
<td>d, e</td>
<td>natural pozzolanas* pulverised fuel ash</td>
</tr>
<tr>
<td>HYDRAULIC CEMENTS</td>
<td>b, e</td>
<td>Portland cements polymer cements hydrothermal cements</td>
</tr>
<tr>
<td>MINERALS/AGGREGATES</td>
<td>a, e</td>
<td>anhydrite natural aggregates crushed aggregates</td>
</tr>
<tr>
<td>METALS AND METALLIC COMPOUNDS</td>
<td></td>
<td>lead copper iron magnesium manganese</td>
</tr>
<tr>
<td>BITUMENS</td>
<td>b, e</td>
<td>natural bitumens industrial bitumens*</td>
</tr>
<tr>
<td>CHEMICAL GROUTS</td>
<td>b, e</td>
<td>silicate-based* acrylic-based* formaldehyde-based* lignin-based epoxy-based*</td>
</tr>
<tr>
<td>CARBONS</td>
<td>a, b, e</td>
<td>graphite charcoal</td>
</tr>
</tbody>
</table>

Attributes
a  good, heat transfer properties
b  low permeability
c  favourable chemical buffering properties
d  favourable retention properties
e  favourable mechanical properties

* includes more than one variety.
<table>
<thead>
<tr>
<th>MATERIAL TYPE</th>
<th>RANKING</th>
<th>COMMENTS</th>
</tr>
</thead>
<tbody>
<tr>
<td>SPOIL</td>
<td>1</td>
<td>Principal attributes are (i) geochemical and physical compatibility with the host environment (ii) on-site availability at nominal basic cost. However technical and economic aspects of spoil processing may preclude their use in some instances.</td>
</tr>
<tr>
<td>CLAYS</td>
<td>1</td>
<td>Geological evidence suggests clays may exhibit excellent longevity. However, certain types may undergo alteration depending on chemical/physical conditions e.g. Na-bentonite Ca-bentonite. Diagenetic changes are precluded by temperature-pressure design constraints.</td>
</tr>
<tr>
<td>ZEOLITES</td>
<td>3</td>
<td>Potentially reactive except under extreme alkaline conditions. Research required to determine stabilities under ambient repository conditions. Dehydration precluded by temperature-pressure design constraints.</td>
</tr>
<tr>
<td>POZZOLANAS</td>
<td>2</td>
<td>Good geological and archaeological evidence of longevity for naturally occurring pozzolanas, although the influence of ambient repository conditions requires investigation. The long-term stability of PFA is likely to be comparable to or better than natural varieties.</td>
</tr>
<tr>
<td>HYDRAULIC CEMENTS</td>
<td>2</td>
<td>Favourable archaeological evidence exists although further research is required to determine optimum formulation. Hydrothermal cements are inherently more stable than Portland cements. Longevity of polymer-based cements is doubtful [R].</td>
</tr>
<tr>
<td>MINERALS/AGGREGATES</td>
<td>1</td>
<td>Geological evidence suggest excellent longevity provided mineralogy is matched with that of the host formation.</td>
</tr>
<tr>
<td>METALS/METALLIC COMPOUNDS</td>
<td>3</td>
<td>Longevity depends on geochemical environment. Some forms can be susceptible to groundwater transport.</td>
</tr>
<tr>
<td>BITUMENS</td>
<td>1</td>
<td>Geological evidence suggests excellent longevity, mainly due to lack of affinity to water.</td>
</tr>
<tr>
<td>CHEMICAL GROUTS</td>
<td>R</td>
<td>Silicate grouts likely to be stable based on comparable geological evidence [1]. However, longevity of all organic chemical grout is suspect and requires fundamental research [R].</td>
</tr>
<tr>
<td>CARBONS</td>
<td>1</td>
<td>Graphite and charcoal are essentially inert in the geochemical sense.</td>
</tr>
</tbody>
</table>

RANKINGS

1 documented evidence of geochemical stability over geological periods of time
2 documented evidence of stability over significant time-intervals
3 some doubt as to long-term stability under certain physico-chemical conditions
R fundamental uncertainties concerning longevity.
### Table 29: Candidate Backfill Material Groups: Rankings in Terms of Design Properties

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<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
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</thead>
<tbody>
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<td>CLAYS</td>
<td>3</td>
<td>1†</td>
<td>2</td>
<td>1</td>
<td>1†</td>
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<tr>
<td>ZEOLITES</td>
<td>3</td>
<td>3</td>
<td>2</td>
<td>1</td>
<td>3</td>
</tr>
<tr>
<td>POZZOLANAS</td>
<td>R</td>
<td>2</td>
<td>R</td>
<td>½</td>
<td>1</td>
</tr>
<tr>
<td>HYDRAULIC CEMENTS</td>
<td>3</td>
<td>1</td>
<td>2</td>
<td>2</td>
<td>1</td>
</tr>
<tr>
<td>MINERALS/AGGREGATES</td>
<td>½</td>
<td>3</td>
<td>2</td>
<td>2</td>
<td>1</td>
</tr>
<tr>
<td>METALS/METALLIC COMPOUNDS</td>
<td>1</td>
<td>3†</td>
<td>1</td>
<td>1</td>
<td>3†</td>
</tr>
<tr>
<td>BITUMEN</td>
<td>3</td>
<td>1</td>
<td>2</td>
<td>2</td>
<td>3</td>
</tr>
<tr>
<td>CHEMICAL GROUTS</td>
<td>R</td>
<td>1</td>
<td>R</td>
<td>R</td>
<td>1</td>
</tr>
<tr>
<td>CARBONS</td>
<td>a. Graphite</td>
<td>½</td>
<td>2</td>
<td>2</td>
<td>1/R</td>
</tr>
<tr>
<td></td>
<td>b. Charcoal</td>
<td>3</td>
<td>2</td>
<td></td>
<td>3</td>
</tr>
</tbody>
</table>

* Spoil materials too varied to be included in group classification (see Appendix A)
† Certain varieties may exhibit favourable swelling properties.
### TABLE 30 CANDIDATE BACKFILL MATERIALS: SUMMARY BASED ON SCREENING/RANKING PROCESS

<table>
<thead>
<tr>
<th>CATEGORY</th>
<th>DEFINITION</th>
<th>SCREENING &amp; CLASSIFICATION CRITERIA</th>
<th>MATERIAL GROUPS/TYPES</th>
</tr>
</thead>
</table>
| 1        | Materials whose properties are currently known to an extent which allows their design performance to be assessed on a quantitative basis | **LONGEVITY RANKINGS (SCREENING)** Predominantly 1 and 2  
**PROPERTY RANKINGS (CLASSIFICATION)** Predominantly 1 and 2. Ranking 3 tolerated if adverse property likely to be improved by mixing with other constituents | o crystalline rock spoil  
o argillaceous rock spoil  
o siliceous rock spoil  
o illite, kaolinite, smectites  
o Portland Cements  
o pulverised fuel ash (PFA)  
o natural sands and gravels  
o crushed siliceous aggregates  
o industrial bitumens |
| 2        | Materials whose properties are known to an extent which enables them to be tentatively incorporated in backfill designs, pending the outcome of further research regarding specific properties | **LONGEVITY RANKINGS (SCREENING)** Predominantly 1, 2 and R  
**PROPERTY RANKINGS (CLASSIFICATION)** Predominantly 1, 2 and R. Ranking 3 tolerated if adverse property likely to be improved by mixing with other constituents | o polygorskites  
o natural zeolites  
o natural pozzolanas  
o polymer cements, hydrothermal cements  
o anhydrite  
o metals and metallic compounds  
o chemical grouts  
o graphite |
| 3        | Materials which may possess desirable attributes but whose properties are poorly understood | **LONGEVITY RANKINGS (SCREENING)** 1, 2, 3 and R  
**PROPERTY RANKINGS (CLASSIFICATION)** High proportion of R rankings. | o vermiculite  
o synthetic zeolites  
o magnesium oxide/silica  
o charcoal |
<table>
<thead>
<tr>
<th>REGION</th>
<th>TYPE OF OPENING</th>
<th>VOID VOLUME (m³)</th>
<th>VOID VOLUME (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>REDUNDANT</td>
<td>Shafts</td>
<td>70,000</td>
<td>7%</td>
</tr>
<tr>
<td></td>
<td>Tunnels</td>
<td>200,000</td>
<td>20%</td>
</tr>
<tr>
<td>ACTIVE</td>
<td>Tunnels</td>
<td>700,000</td>
<td>70%</td>
</tr>
<tr>
<td></td>
<td>Boreholes</td>
<td>30,000</td>
<td>3%</td>
</tr>
<tr>
<td>TOTAL</td>
<td></td>
<td>1M m³</td>
<td>100%</td>
</tr>
</tbody>
</table>
**TABLE 32 SOURCES OF BACKFILL MATERIAL COST DATA**

<table>
<thead>
<tr>
<th>MATERIAL GROUP</th>
<th>MATERIAL TYPE</th>
<th>SOURCE OF INFORMATION</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>CLAYS</strong></td>
<td>Kaolinitic Ball Clay</td>
<td>8, 20, 22</td>
</tr>
<tr>
<td></td>
<td>Kaolinite</td>
<td>2, 22</td>
</tr>
<tr>
<td></td>
<td>Montmorillonite</td>
<td>3</td>
</tr>
<tr>
<td></td>
<td>Vermiculite</td>
<td>11</td>
</tr>
<tr>
<td><strong>ZEOLITES</strong></td>
<td>Clinoptilolite</td>
<td>4</td>
</tr>
<tr>
<td></td>
<td>Philipsite</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Analcime (mixture)</td>
<td>4</td>
</tr>
<tr>
<td></td>
<td>Chabazite</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Artificial zeolites</td>
<td>4</td>
</tr>
<tr>
<td><strong>HYDRAULIC CEMENTS</strong></td>
<td>Portland Cement</td>
<td>23</td>
</tr>
<tr>
<td></td>
<td>White</td>
<td>23</td>
</tr>
<tr>
<td></td>
<td>Sulphate resisting</td>
<td>23</td>
</tr>
<tr>
<td></td>
<td>High alumina Cement</td>
<td>23</td>
</tr>
<tr>
<td></td>
<td>Rapid hardening</td>
<td>23</td>
</tr>
<tr>
<td></td>
<td>Mass concrete</td>
<td>23</td>
</tr>
<tr>
<td><strong>POZZOLANAS</strong></td>
<td>Natural (Diatomite)</td>
<td>4</td>
</tr>
<tr>
<td></td>
<td>PFA (non-refined)</td>
<td>5</td>
</tr>
<tr>
<td></td>
<td>PFA (refined)</td>
<td>16, 17</td>
</tr>
<tr>
<td><strong>MINERALS/AGGREGATES</strong></td>
<td>Anhydrite</td>
<td>13, 18</td>
</tr>
<tr>
<td></td>
<td>Crushed Quartz Sand</td>
<td>11</td>
</tr>
<tr>
<td></td>
<td>Silica Flour</td>
<td>11</td>
</tr>
<tr>
<td></td>
<td>Natural Sand</td>
<td>23</td>
</tr>
<tr>
<td></td>
<td>Natural Gravel</td>
<td>23</td>
</tr>
<tr>
<td><strong>METALS</strong></td>
<td>Lead Minerals</td>
<td>24</td>
</tr>
<tr>
<td></td>
<td>Copper Minerals</td>
<td>7</td>
</tr>
<tr>
<td></td>
<td>Iron Oxide</td>
<td>1, 10, 12</td>
</tr>
<tr>
<td></td>
<td>Manganese Oxide</td>
<td>21</td>
</tr>
<tr>
<td></td>
<td>Magnesium Oxide</td>
<td>14</td>
</tr>
<tr>
<td><strong>BITUMENS</strong></td>
<td>Processed (road grade)</td>
<td>9, 16</td>
</tr>
<tr>
<td></td>
<td>Asphalt</td>
<td>23</td>
</tr>
<tr>
<td><strong>OTHERS</strong></td>
<td>Graphite (natural)</td>
<td>6</td>
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<tr>
<td></td>
<td>Graphite (artificial)</td>
<td>21</td>
</tr>
<tr>
<td></td>
<td>Charcoal</td>
<td>19</td>
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</tbody>
</table>

**KEY**

(a) SUPPLIERS

1. BAYER UK LTD, Bayer House, Richmond, Surrey, TW9 1SJ
2. BISMIN, Moneystone Quarry, Dukmoo, Staffs, ST10 3DX
3. BOOTH ENGINEERING LTD, Hawksworth Road, St Neots, Huntingdon, Cambs PE19 1NB
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13. IRWELL LTD, 1A Market Street, Altrincham, Cheshire WA14 1QG
14. THE MAGNESITE SYNDICATE LTD, Station Road, Calverley, Bradford, West Yorkshire
15. MOBIL OIL CO.LTD, 90 Victoria St, London, SW1E 6OB
16. PFA SEPARATIONS LTD, 24 Stafford Terrace, London W8 7DH
17. POZZOLANIC LTD, Nicholas St.Means, Chester CH1 2NS
18. RICHARD BAKER HARRISON LTD, 23 Cranbrook Road, Ilford, Essex IG1 4RG
19. SHIRLEY ALDRED & CO.LTD, Oakwood Chemical Works, Sandy Lane, Worksop, Notts, S80 2SY
20. STEETLEY MINERALS LTD, P.O.Box 2, Carlton Road, Worksop, Notts, S81 7OG
21. THOMAS HILL JONES, 15 High Street, Stratford, London E16 2WJ
22. WBL Park House, Courtown Park, Newton Abbott, Devon TQ12 4PS

(b) PUBLICATIONS

23. 'BUILDING SPECIFICATION' Cost Index, January 1982
24. 'FINANCIAL TIMES' 1/6/82

ADDITIONAL INFORMATION FROM:

CEMENTATION Ltd, Cementation House, Maple Cross, Rickmansworth, Herts. WD3 2SW
BRITISH QUARRYING AND SLAG FEDERATION, Carolyn House, Dingwall Rd, Croydon Surrey CR0 9XF
THE RESOURCES INSTITUTE, Atholl Estate Office, Blair Atholl, Perthshire.
<table>
<thead>
<tr>
<th>MATERIAL GROUP</th>
<th>MATERIAL TYPE</th>
<th>FORM</th>
<th>COST PER TONNE £</th>
<th>SPECIFIED MATERIAL SOURCE</th>
<th>OTHER SOURCES IN EUROPE</th>
</tr>
</thead>
<tbody>
<tr>
<td>CLAYS</td>
<td>Kaolinitic Ball Clay</td>
<td>bulk, shredded</td>
<td>9.00</td>
<td>Dorset, UK</td>
<td>France, Germany</td>
</tr>
<tr>
<td></td>
<td>Kaolinitic Ball Clay</td>
<td>pulverised, bagged</td>
<td>38.00</td>
<td>Dorset, UK</td>
<td>France, Germany</td>
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<tr>
<td></td>
<td>Kaolinite</td>
<td>bulk, powdered</td>
<td>45.65</td>
<td>Devon, UK</td>
<td>France, Germany</td>
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<tr>
<td></td>
<td>Montmorillonite</td>
<td>powdered, bagged</td>
<td>49.60</td>
<td>UK</td>
<td>Greece, Italy</td>
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<tr>
<td></td>
<td>Vermiculite</td>
<td>exfoliated, bagged</td>
<td>350.00</td>
<td></td>
<td>none</td>
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<tr>
<td>ZEOLITES</td>
<td>Clinoptilolite</td>
<td>powdered, bagged</td>
<td>150.00</td>
<td>Mexico</td>
<td>none</td>
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<td></td>
<td>Philipsite</td>
<td>powdered, bagged</td>
<td>80.00</td>
<td>Germany</td>
<td>Germany, Italy</td>
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<tr>
<td></td>
<td>Analcime</td>
<td>powdered, bagged</td>
<td>500.00</td>
<td></td>
<td>UK (manufactured)</td>
</tr>
<tr>
<td></td>
<td>Chabazite</td>
<td>powdered, bagged</td>
<td></td>
<td></td>
<td>many countries</td>
</tr>
<tr>
<td></td>
<td>Artificial zeolites</td>
<td>powdered, bagged</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>HYDRAULIC CEMENTS</td>
<td>Portland Cement</td>
<td>powdered, bagged</td>
<td>46.60</td>
<td>Throughout UK</td>
<td>many countries</td>
</tr>
<tr>
<td></td>
<td>Low heat cement</td>
<td>bulk, ready mixed</td>
<td>59.00</td>
<td></td>
<td></td>
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<tr>
<td></td>
<td>Sulphate resisting cement</td>
<td>bulk, ready mixed</td>
<td>54.60</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>High Alumina</td>
<td>powdered, bagged</td>
<td>53.00</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>Rapid hardening</td>
<td>powdered, bagged</td>
<td>51.00</td>
<td></td>
<td></td>
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<tr>
<td>FOZZOLANAS</td>
<td>Natural Diatomite</td>
<td>powdered</td>
<td>14.00</td>
<td>Skye Scotland</td>
<td>Germany, Italy</td>
</tr>
<tr>
<td></td>
<td>PFA (non-refined)</td>
<td>moistened, bulk</td>
<td>7.00</td>
<td>Throughout UK</td>
<td>many countries</td>
</tr>
<tr>
<td></td>
<td>PFA (refined)</td>
<td>dry, bulk</td>
<td>24.00</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>PFA (refined)</td>
<td>dry, bagged</td>
<td>14.00</td>
<td></td>
<td></td>
</tr>
<tr>
<td>MINERALS/AGGREGATES</td>
<td>Anhydrite</td>
<td>powdered, bagged</td>
<td>25.00</td>
<td>UK</td>
<td>France, Germany</td>
</tr>
<tr>
<td></td>
<td>Crushed Quartz Sand</td>
<td>graded, bagged</td>
<td>25.20</td>
<td>UK</td>
<td>many countries</td>
</tr>
<tr>
<td></td>
<td>Silica Flour</td>
<td>powdered, bagged</td>
<td>36.50</td>
<td>UK</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Natural Sand</td>
<td>washed, bulk</td>
<td>6.00</td>
<td>Throughout UK</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Natural Gravel</td>
<td>bulk</td>
<td>6.00</td>
<td></td>
<td></td>
</tr>
<tr>
<td>METALS</td>
<td>Lead Minerals</td>
<td>crushed, graded, bulk</td>
<td>290.00</td>
<td>(Estimate only)</td>
<td>Eire, France, Italy</td>
</tr>
<tr>
<td></td>
<td>Copper Minerals</td>
<td>crushed, graded, bulk</td>
<td>270.00</td>
<td>(Estimate only)</td>
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<tr>
<td></td>
<td>Iron Oxide</td>
<td>powdered, bagged</td>
<td>300.00</td>
<td>Germany</td>
<td>France, Germany</td>
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<tr>
<td></td>
<td>Manganese Dioxide</td>
<td>coarse sand, bulk</td>
<td>85.00</td>
<td>Not known</td>
<td>Belgium, France, Greece</td>
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<td>Magnesium Dioxide</td>
<td>powdered, bagged</td>
<td>140.00</td>
<td></td>
<td>Greece</td>
</tr>
<tr>
<td></td>
<td>Magnesium Oxide</td>
<td>powdered, bagged</td>
<td>110.00</td>
<td></td>
<td>France, Italy</td>
</tr>
<tr>
<td>BITUMENS</td>
<td>Processed (road grade)</td>
<td>tin drums</td>
<td>130.00</td>
<td>UK</td>
<td>many countries</td>
</tr>
<tr>
<td></td>
<td>Processed (road grade)</td>
<td>(not known)</td>
<td>140.00</td>
<td>UK</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Asphalt</td>
<td></td>
<td>110.00</td>
<td>UK</td>
<td></td>
</tr>
<tr>
<td>CARBONS</td>
<td>Graphite (natural)</td>
<td>powdered, bulk</td>
<td>150.00</td>
<td>China</td>
<td>Greece, Italy</td>
</tr>
<tr>
<td></td>
<td>Graphite (artificial)</td>
<td>powdered, bagged</td>
<td>330.00</td>
<td>UK</td>
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<tr>
<td></td>
<td>Charcoal</td>
<td>powdered, bagged</td>
<td>300.00</td>
<td>UK</td>
<td></td>
</tr>
</tbody>
</table>

*Note: Costs and specifications may vary depending on source and location.*
<table>
<thead>
<tr>
<th>MATERIAL ASSEMBLAGE</th>
<th>UTILIZATION % by volume</th>
<th>CONSTITUENTS</th>
<th>VOLUMES (m³)</th>
<th>TONNAGE REQUIRED (t)</th>
<th>COST PER TONNE (£)</th>
<th>TOTAL MATERIAL COST (£)</th>
<th>COST INDEX</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td></td>
<td>REDUNDANT</td>
<td>ACTIVE</td>
<td>REDUNDANT</td>
<td>ACTIVE</td>
<td>REDUNDANT</td>
</tr>
<tr>
<td>SPOIL / SAND / BENTONITE</td>
<td>100% in shafts and tunnels (e, f, g)</td>
<td>CRUSHED ROCK</td>
<td>35</td>
<td>70,000</td>
<td>200,000</td>
<td>700,000</td>
<td>1</td>
</tr>
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<td>SPOIL / BENTONITE</td>
<td>100% in shafts and tunnels (e, f, g)</td>
<td>SAND</td>
<td>20</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>BENTONITE</td>
<td>10%</td>
<td>BENTONITE</td>
<td>10</td>
<td>13,000</td>
<td>20,000</td>
<td>133,000</td>
<td>-</td>
</tr>
<tr>
<td>QUARTZ / BENTONITE</td>
<td>100% of repository (e, f, g, h)</td>
<td>QUARTZ</td>
<td>50</td>
<td>70,000</td>
<td>200,000</td>
<td>700,000</td>
<td>1</td>
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<tr>
<td>CONCRETE</td>
<td>45% in shafts and tunnels (e, f, g)</td>
<td>PORTLAND CEMENT</td>
<td>12</td>
<td>-</td>
<td>37,500</td>
<td>50,000</td>
<td>315,000</td>
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<tr>
<td>CONCRETE</td>
<td>45% in shafts and tunnels (e, f, g)</td>
<td>SILICA FLOUR</td>
<td>6</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>CONCRETE</td>
<td>50% in shafts and tunnels (e, f, g)</td>
<td>SAND</td>
<td>34</td>
<td>28,200</td>
<td>75,000</td>
<td>262,400</td>
<td>-</td>
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<tr>
<td>CONCRETE</td>
<td>50% in shafts and tunnels (e, f, g)</td>
<td>GRAVEL</td>
<td>48</td>
<td>-</td>
<td>37,000</td>
<td>106,800</td>
<td>376,400</td>
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<td>CONCRETE</td>
<td>50% in shafts and tunnels (e, f, g)</td>
<td>AS ABOVE</td>
<td>100</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>SPOIL / CLAY</td>
<td>100% in redundant areas (e, f, g, h)</td>
<td>CRUSHED ROCK</td>
<td>50</td>
<td>70,000</td>
<td>200,000</td>
<td>700,000</td>
<td>1</td>
</tr>
<tr>
<td>SPOIL / CLAY</td>
<td>100% in redundant areas (e, f, g, h)</td>
<td>BENTONITE</td>
<td>10</td>
<td>19,800</td>
<td>30,000</td>
<td>198,000</td>
<td>-</td>
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<tr>
<td>EXPANSIVE BLOCKS</td>
<td>20% shafts and tunnels</td>
<td>BENTONITE</td>
<td>100</td>
<td>14,000</td>
<td>40,000</td>
<td>140,000</td>
<td>1</td>
</tr>
<tr>
<td>SPOIL / CLAY</td>
<td>100% in redundant areas (e, f, g, h)</td>
<td>HIGH QUALITY CONCRETE</td>
<td>100</td>
<td>7,000</td>
<td>20,000</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>SORBS (alternatives)</td>
<td>5% in active areas (e, f, g, h)</td>
<td>MIXED ZEOLITES</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>SORBS (alternatives)</td>
<td>5% in active areas (e, f, g, h)</td>
<td>CLINOPTILULITE</td>
<td>5</td>
<td>-</td>
<td>-</td>
<td>-</td>
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<td>ADDITIVES</td>
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<td>CHARCOAL</td>
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<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>BUFFERS (alternatives)</td>
<td>2% in active areas (e, f, g, h)</td>
<td>MAGNESIUM*</td>
<td>-</td>
<td>25,200</td>
<td>1,080</td>
<td>25,280</td>
<td>-</td>
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<tr>
<td>BUFFERS (alternatives)</td>
<td>2% in active areas (e, f, g, h)</td>
<td>IRON*</td>
<td>-</td>
<td>25,200</td>
<td>1,080</td>
<td>25,280</td>
<td>-</td>
</tr>
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<td>BUFFERS (alternatives)</td>
<td>2% in active areas (e, f, g, h)</td>
<td>LEAD*</td>
<td>-</td>
<td>25,200</td>
<td>1,080</td>
<td>25,280</td>
<td>-</td>
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<tr>
<td>BUFFERS (alternatives)</td>
<td>2% in active areas (e, f, g, h)</td>
<td>MANGANESE*</td>
<td>-</td>
<td>25,200</td>
<td>1,080</td>
<td>25,280</td>
<td>-</td>
</tr>
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</table>

* initial or material compound

Total cost index for the construction of crystalline host rocks.
<table>
<thead>
<tr>
<th>MATERIAL ASSEMBLAGE</th>
<th>UTILIZATION % by volume</th>
<th>CONSTITUENTS MATERIAL TYPE</th>
<th>VOLUMES (m³)</th>
<th>TONNAGE REQUIRED (t)</th>
<th>COST PER TONNE</th>
<th>TOTAL MATERIAL COST (DM)</th>
<th>COST INDEX</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>REDUNDANT Active</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>Shells max 70,000 Tunnels max 200,000 Boreholes max 30,000</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Bentonite / Quartz Sand</td>
<td>100% in active areas</td>
<td>Bentonite 90</td>
<td>- -</td>
<td>700,000 30,000</td>
<td>1.70</td>
<td>- -</td>
<td>119,000 5,100</td>
</tr>
<tr>
<td>Concrete</td>
<td>50% in shafts and tunnels</td>
<td>Quartz Sand 35</td>
<td>- -</td>
<td>35,000 100,000</td>
<td>-</td>
<td>2.45</td>
<td>12,000 50,000</td>
</tr>
<tr>
<td>ROCK Fills</td>
<td>90% of repository</td>
<td>Portland Cement 5</td>
<td>- -</td>
<td>15,000 30,000</td>
<td>-</td>
<td>1.70</td>
<td>15,000 18,000</td>
</tr>
<tr>
<td>Pulverized Rock / Sand</td>
<td>100% in redundant areas</td>
<td>Rock Spoil 20</td>
<td>- -</td>
<td>70,000 200,000</td>
<td>-</td>
<td>1.70</td>
<td>20,000 60,000</td>
</tr>
<tr>
<td>EXPANSIVE BLOCKS</td>
<td>25% in shafts and tunnels</td>
<td>Sand 80</td>
<td>- -</td>
<td>95,000 272,000</td>
<td>-</td>
<td>-</td>
<td>307,000 272,000</td>
</tr>
<tr>
<td>SPECIAL FILL</td>
<td>10% in redundant areas</td>
<td>Bentonite 100</td>
<td>- -</td>
<td>140,000 140,000</td>
<td>-</td>
<td>2.45</td>
<td>17,200 49,000</td>
</tr>
<tr>
<td>CUT-OFF SEALS</td>
<td></td>
<td></td>
<td></td>
<td>REDUNDANT Active</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Sorbants (alternatives)</td>
<td>5% in active area</td>
<td>Bentonite 90</td>
<td>- -</td>
<td>35,000 15,000</td>
<td>1.80</td>
<td>- -</td>
<td>63,000 2,700</td>
</tr>
<tr>
<td>MIXED ZEOLITES</td>
<td></td>
<td></td>
<td></td>
<td>REDUNDANT Active</td>
<td></td>
<td></td>
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</tr>
<tr>
<td>CLINOPTILOLITE</td>
<td></td>
<td></td>
<td></td>
<td>graphite 25</td>
<td>- -</td>
<td>14,000 500</td>
<td>25,200 1,080</td>
</tr>
<tr>
<td>GRAPHITE</td>
<td></td>
<td></td>
<td></td>
<td>Graphite 25</td>
<td>- -</td>
<td>14,000 500</td>
<td>25,200 1,080</td>
</tr>
<tr>
<td>CHARCOAL</td>
<td></td>
<td></td>
<td></td>
<td>lead 25</td>
<td>- -</td>
<td>14,000 500</td>
<td>25,200 1,080</td>
</tr>
<tr>
<td>Additives</td>
<td></td>
<td></td>
<td></td>
<td>REDUNDANT Active</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Buffers (alternatives)</td>
<td>2% in active area</td>
<td>Graphite 25</td>
<td>- -</td>
<td>25,200 1,080</td>
<td>1.80</td>
<td>-</td>
<td>25,200 1,080</td>
</tr>
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</table>
### Table 36: Backfilling Cost Analysis; Clay Host Strata

<table>
<thead>
<tr>
<th>Material Assemblage</th>
<th>Utilization % by Volume</th>
<th>Constituents</th>
<th>Volumes (m³)</th>
<th>Tonnage Required (t)</th>
<th>Cost Per Tonne (£/t)</th>
<th>Total Material Cost (£M)</th>
<th>Cost Indices</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td></td>
<td>Redundant</td>
<td>Active</td>
<td>Redundant</td>
<td>Active</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Shaft max 70,000</td>
<td>Tunnel max 200,000</td>
<td>Shaft max 70,000</td>
<td>Tunnel max 200,000</td>
<td>Shaft max 70,000</td>
</tr>
<tr>
<td>Clay Spoil / Cement</td>
<td>100% in tunnels (f, g)</td>
<td>Clay Spoil</td>
<td>85</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Cement</td>
<td>15</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>Clay Spoil</td>
<td>100% in shafts (a)</td>
<td>Clay Spoil</td>
<td>100</td>
<td>70,000</td>
<td>-</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>Bentonite / Quartz</td>
<td>100% in boreholes (h)</td>
<td>Bentonite</td>
<td>10</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Quartz Sand</td>
<td>90</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>Cement / Clay GROUT</td>
<td>5% in tunnels and shafts (c, d)</td>
<td>Cement</td>
<td>10</td>
<td>3,500</td>
<td>10,000</td>
<td>35,000</td>
<td>1.56</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Bentonite</td>
<td>90</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>Buffers / Sorbants</td>
<td>2% in active area (g, h)</td>
<td>Mixed Zeolites</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Magnesium</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Iron*</td>
<td>2</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td></td>
<td></td>
<td>LEAD*</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Manganese</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
</tr>
</tbody>
</table>

* metal or metallic compound
### Table 37 Backfilling Cost Analysis: Saliferous Host Rocks

<table>
<thead>
<tr>
<th>Material Assemblage</th>
<th>Utilization % by Volume</th>
<th>Constituents</th>
<th>Volumes (m³)</th>
<th>Tonnage Required (t)</th>
<th>Cost Per Tonne (£)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td></td>
<td>Redundant</td>
<td>Active</td>
<td>Redundant</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Shafts max 20,000</td>
<td>Tunnels max 200,000</td>
<td>Boreholes max 30,000</td>
</tr>
<tr>
<td><strong>Spoil/Clay</strong></td>
<td>100% in tunnels (ft, g)</td>
<td>Crushed Salt</td>
<td>85%</td>
<td>200,000</td>
<td>700,000</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Bentonite</td>
<td>15%</td>
<td></td>
<td></td>
</tr>
<tr>
<td><strong>Spoil/PFA</strong></td>
<td>100% in tunnels (ft, g)</td>
<td>Crushed Salt</td>
<td>85%</td>
<td>200,000</td>
<td>700,000</td>
</tr>
<tr>
<td></td>
<td></td>
<td>PFA</td>
<td>15%</td>
<td></td>
<td></td>
</tr>
<tr>
<td><strong>Composite Fill</strong></td>
<td>50% in shafts (ft)</td>
<td>Crushed Salt</td>
<td>40%</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Concrete</td>
<td>40%</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Bitumen</td>
<td>15%</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Bentonite</td>
<td>5%</td>
<td></td>
<td></td>
</tr>
<tr>
<td><strong>Composite Fill</strong></td>
<td>50% in shafts (ft)</td>
<td>Quartz Sand</td>
<td>80%</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Bentonite</td>
<td>20%</td>
<td></td>
<td></td>
</tr>
<tr>
<td><strong>Composite Fill</strong></td>
<td>100% in boreholes (ft)</td>
<td>Concrete</td>
<td>25%</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Clay</td>
<td>25%</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Salt</td>
<td>15%</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>PFA</td>
<td>10%</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Quartz Sand</td>
<td>25%</td>
<td></td>
<td></td>
</tr>
<tr>
<td><strong>Bitumen</strong></td>
<td>15% in boreholes (ft)</td>
<td>Bitumen</td>
<td>100%</td>
<td>10,000</td>
<td>20,000</td>
</tr>
<tr>
<td><strong>Salt</strong></td>
<td>15% in shafts (ft)</td>
<td>OPC</td>
<td>15%</td>
<td>70,500</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Crushed Salt</td>
<td>40%</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Quartz Sand</td>
<td>35%</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Bentonite</td>
<td>10%</td>
<td></td>
<td></td>
</tr>
<tr>
<td><strong>Bitumen</strong></td>
<td>15% in boreholes (ft)</td>
<td>Bitumen</td>
<td>35%</td>
<td>10,000</td>
<td>30,000</td>
</tr>
<tr>
<td><strong>Concrete</strong></td>
<td>15% in boreholes (ft)</td>
<td>High Quality Concrete</td>
<td>65%</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

**Notes:**
- % by weight.
- Materials are mixed in specified volumes.

**Unit Cost Index:**
- Total Cost Index = Material Type + Material Assemblage.
### TABLE 38 BACKFILLING COST ANALYSIS; TOTAL COST COMPARISONS

<table>
<thead>
<tr>
<th>HOST ROCK TYPE</th>
<th>BACKFILL CATEGORY</th>
<th>MEAN COST PER TONNE (estimated)</th>
<th>% OF TOTAL BACKFILL WEIGHT</th>
<th>TOTAL CATEGORY COST (£M)</th>
<th>% OF TOTAL BACKFILL MATERIAL COST</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>CRYSTALLINE ROCKS</strong></td>
<td>Bulk Fills</td>
<td>15</td>
<td>80</td>
<td>22</td>
<td>44</td>
</tr>
<tr>
<td></td>
<td>Special Fills</td>
<td>45</td>
<td>15</td>
<td>12</td>
<td>23</td>
</tr>
<tr>
<td></td>
<td>Sorbants</td>
<td>182</td>
<td>3.5</td>
<td>12</td>
<td>23</td>
</tr>
<tr>
<td></td>
<td>Buffers</td>
<td>200</td>
<td>1.5</td>
<td>5</td>
<td>10</td>
</tr>
<tr>
<td><strong>TOTAL BACKFILL MATERIAL COST (T.B.M.C.)</strong></td>
<td></td>
<td></td>
<td></td>
<td>51</td>
<td>100%</td>
</tr>
<tr>
<td><strong>UNINDURATED PLASTIC CLAYS</strong></td>
<td>Bulk Fills</td>
<td>16</td>
<td>93</td>
<td>27</td>
<td>69</td>
</tr>
<tr>
<td></td>
<td>Special Fills</td>
<td>50</td>
<td>5</td>
<td>5</td>
<td>13</td>
</tr>
<tr>
<td></td>
<td>Sorbants/Buffers</td>
<td>186</td>
<td>2</td>
<td>7</td>
<td>18</td>
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<tr>
<td><strong>TOTAL BACKFILL MATERIAL COST (T.B.M.C.)</strong></td>
<td></td>
<td></td>
<td></td>
<td>39</td>
<td>100%</td>
</tr>
<tr>
<td><strong>STRONG INDURATED ARGILLACEOUS ROCKS</strong></td>
<td>Bulk Fills</td>
<td>15</td>
<td>80</td>
<td>22</td>
<td>44</td>
</tr>
<tr>
<td></td>
<td>Special Fills</td>
<td>45</td>
<td>15</td>
<td>12</td>
<td>23</td>
</tr>
<tr>
<td></td>
<td>Sorbants</td>
<td>182</td>
<td>3.5</td>
<td>12</td>
<td>23</td>
</tr>
<tr>
<td></td>
<td>Buffers</td>
<td>200</td>
<td>1.5</td>
<td>5</td>
<td>10</td>
</tr>
<tr>
<td><strong>TOTAL BACKFILL MATERIAL COST (T.B.M.C.)</strong></td>
<td></td>
<td></td>
<td></td>
<td>51</td>
<td>100%</td>
</tr>
<tr>
<td><strong>SALIFEROUS ROCKS</strong></td>
<td>Bulk Fills</td>
<td>8</td>
<td>95</td>
<td>14</td>
<td>67</td>
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<td>Seals</td>
<td>75</td>
<td>5</td>
<td>7</td>
<td>33</td>
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<tr>
<td><strong>TOTAL BACKFILL MATERIAL COST (T.B.M.C.)</strong></td>
<td></td>
<td></td>
<td></td>
<td>21</td>
<td>100%</td>
</tr>
</tbody>
</table>

*Total backfill weight assumed to be 1.8 M tonnes.
<table>
<thead>
<tr>
<th>FLOW ZONE</th>
<th>VOLUMETRIC FLOW RATE</th>
<th>FLOW VELOCITY</th>
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</thead>
<tbody>
<tr>
<td>Bulk Fill</td>
<td>$Q = A k_f i$</td>
<td>$v = \frac{k_f i}{\eta}$</td>
</tr>
<tr>
<td>Plane of Separation</td>
<td>$Q = b^3 \rho g \pi D i \frac{1}{12 \eta}$</td>
<td>$v = \frac{b^2 \rho g \cdot i}{12 \eta}$</td>
</tr>
<tr>
<td>Disturbed Zone</td>
<td>$Q = A k_0 i$</td>
<td>$v = \frac{b^2 \rho g \cdot i}{12 \eta}$</td>
</tr>
<tr>
<td></td>
<td>where $b = 3 \frac{Q \cdot 12 \eta}{\sqrt{\rho g \pi D \cdot i}}$</td>
<td></td>
</tr>
</tbody>
</table>
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