ERDA-76-43

VOL. 4 OF 5

Alternatives For Managing Wastes From Reactors And Post-Fission Operations In The LWR Fuel Cycle

MASTER

**Volume 4:
Alternatives For Waste Isolation And Disposal**

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Alternatives For Managing Wastes From Reactors And Post-Fission Operations In The LWR Fuel Cycle

MAY 1976

United States
Energy Research & Development
Administration

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Volume 4
(4 of 5 Volumes)

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Report Coordinated By
Battelle, Pacific Northwest Laboratories
At The Request Of The
Division Of Nuclear Fuel Cycle And Production

FOREWORD

This report describes technical alternatives for managing wastes from the back end of the commercial LWR fuel cycle. It describes waste types, the alternative technologies that can be used to manage the wastes, and the state of availability of those technologies.

The report does not select preferred waste management technologies or make comparative assessments. Its purpose is to provide a comprehensive digest of technical information expected to serve as a basis for future decisions, assessments, and environmental impact statements. The technology is still evolving, and this report will be subjected to public review and comment. Its contents therefore serve as a baseline subject to future supplementation and updating.

There are four major waste management functions: treatment, interim storage, transportation, and final storage or disposal. This report is structured in five volumes that reflect these functions. Volume 1 summarizes the report, describes alternatives for the back end of the LWR cycle, and provides background information on waste types and quantities. Volume 2 describes alternatives for waste treatment. Volume 3 discusses interim storage and transportation. Alternatives for final storage and disposal are described in Volume 4. The Appendices, Volume 5, provide supplementary information, with emphasis on characteristics of geologic formations that might be used for final storage or disposal.

Preparation of this report was started in August of 1975, in response to a directive from ERDA's Nuclear Fuel Cycle and Production Division. The report was authored by personnel of ERDA's contractors, of industry, and of universities; coordination was provided by Battelle's Pacific Northwest Laboratories. The report was completed in March of 1976. The cutoff date for characterizing the status of the technologies was September 1, 1975.

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ALTERNATIVES FOR WASTE ISOLATION AND DISPOSAL

INTRODUCTION

This volume describes alternatives for operations designed to remove radioactive wastes from man's environment. Such operations are conventionally called "disposal;" in this document, that term is used specifically for concepts that offer little or no chance for reversibility. Some concepts do permit reversibility, i.e., retrieval; these are described as "storage" operations. Storage and disposal alternatives that leave the wastes in existence on earth but remove them from man's environment are "isolation" concepts. Some disposal alternatives would eliminate the wastes from existence on earth.

Management by isolation implies emplacement in geologic media. Two basic isolation alternatives can be defined:

.Isolation by storage. The wastes would be put into geologic media so that they can be retrieved. "Provisional geologic storage" would use emplacement techniques and monitoring operations that permit retrieval with methods and rates essentially the same as those used for emplacement. Repositories designed for provisional storage could be converted to "permanent geologic storage" by backfilling and sealing.

.Isolation by disposal. The wastes would be put into geologic media using emplacement techniques that make retrieval highly difficult or impractical.

There are two other disposal alternatives that involve eliminating the wastes from existence on earth rather than isolation in geologic media. These are:

.Elimination by ejection from earth by rocket.

.Elimination by transmutation, which involves bombarding the radioactive nuclei with radiation or nuclear particles, thereby changing them into other isotopes.

Numerous specific concepts have been identified for each of the basic isolation and disposal alternatives.⁽¹⁾ Isolation concepts are made specific by choice of repository site, emplacement medium, and emplacement method. Rocket ejection is made specific mainly by choice of destination, and transmutation concepts are defined by the bombarding particles and their sources.

The wastes to be managed within the spectrum of these alternatives are the products of prior fuel cycle and waste management operations. Initially, the LWR fuel cycle wastes are radioactive solids, liquids, and gases with a wide range of physical and chemical properties as described in Volumes 1 and 2 of this document. After treatment and packaging for final storage or disposal, there are four basic waste categories:

<u>Waste Type</u>	<u>Example</u>
Wastes that contain transuranics and emitters of significant heat and penetrating radiation	High-level waste
Wastes that contain transuranics and emitters of significant penetrating radiation (high-gamma transuranic waste)	Chop-leach fuel bundle residues
Transuranic wastes that contain transuranics but no significant heat or penetrating radiation emitters (low-gamma transuranic waste)	Trash from MOX plant operations
Wastes that contain emitters of radiation but no transuranics	Immobilized ion-exchange resins from reactors

If the LWR fuel cycle is operated without fuel reprocessing and mixed oxide fuel fabrication, spent fuel and wastes of the above types that come from reactors and spent fuel storage basins would have to be managed.

Geologic isolation is expected to be capable of accommodating all LWR fuel cycle wastes; ERDA wastes and wastes from other commercial fuel cycles also could be managed using these techniques. Technical constraints restrict the rocket ejection and transmutation options to use for selected waste fractions. These would have to be separated from other waste constituents by methods such as are described in Sections 11 and 16 of Volume 2.

In this document, alternatives for geologic isolation are described in terms of four basic emplacement media:

- .near-surface continental geologic formations, i.e., burial grounds
- .deep continental geologic formations
- .seabeds
- .ice sheets.

Emplacement in near-surface geologies involves operations, geologies, and isolation techniques such as are used for nuclear waste burial grounds. Emplacement in deep continental geologies would use depths and geologic

formations selected to provide complete and long-term isolation from activities and processes in the biosphere. Seabed concepts are, in this document, restricted specifically to emplacement in the sea floor.

Discussion of the isolation and disposal alternatives is structured under five major headings.

- .Section 23, Basic Concepts for Geologic Isolation, describes generic principles and factors common to all geologic storage and disposal systems.
- .Section 24, Geologic Storage Alternatives, describes options for retrievable emplacement of wastes into burial grounds or deep continental geologic formations.
- .Section 25, Geologic Disposal Alternatives, describes options for irretrievable emplacement of wastes into deep continental geologic formations, seabeds, and ice sheets.
- .Section 26 describes alternatives for extraterrestrial disposal.
- .Section 27 describes alternatives for disposal by transmutation.

23.0 BASIC CONCEPTS FOR GEOLOGIC ISOLATION

23.0 BASIC CONCEPTS FOR GEOLOGIC ISOLATION

All geologic storage and disposal repositories will be located in tectonically, hydrologically, and mechanically stable geologic media. This stability is the primary means to maintain the wastes in isolation. Other characteristics of the storage or disposal system are selected to augment this stability.

Geologic disposal repositories, for which retrieval of wastes is not expected from the outset, would rely only on the stability and nuclide retention capability of the geology to keep the wastes in isolation. Storage repositories, which would be designed to permit retrieval of wastes, would attain their retrievability features by using waste forms, engineered structures, and emplacement techniques selected to preserve the waste configuration and access to the containers. Provisional geologic storage repositories could be converted to permanent geologic storage by backfilling and sealing.

23.1 SYSTEM CONCEPTS

Each geologic storage or disposal site is a system with the following significant elements:

- . geologic medium
- . engineered structures for retrievability, if any
- . waste form
- . quantities and types of wastes
- . logistics of waste handling, transportation, and support services
- . biologic environment
- . demographic environment.

These system characteristics are quantified and evaluated in a three-step selection, design, and qualification procedure:

1. Identify sites with appropriate environmental and geologic characteristics.
2. Design and size the site-specific system so that facilities, structures, and methods of waste emplacement satisfy design performance and licensing requirements.
3. Perform design evaluations and conduct experiments to confirm system performance.

Methods for evaluating geologic storage and disposal systems are discussed below in Section 23.3.

23.2 SITE SELECTION

This section describes factors considered and methods used in the search for sites for waste repositories in geologic media. The discussion is generic and aimed at selection of sites in deep continental geologic formations. Since selection of such sites includes consideration of surface and near-surface features, information relevant to selection of sites for burial grounds is also included.

Specific needs for information to characterize a geologic formation and select a specific site for a waste repository will depend on the type of geology being considered. Criteria and procedures such as described here would be used on an as-needed basis in order to minimize time and costs required to find acceptable repository sites.

23.2.1 Summary of Site Selection Criteria and Procedures

Sites for geologic isolation of high-level radioactive waste must be selected to assure that there will be no radiation exposure to the general public, either through direct contact or by indirect means involving other portions of man's ecosystem. Because geologic environments vary appreciably, the above criterion could be met in different ways at different sites, provided that all prospective sites were carefully evaluated and tested prior to selection.

Circulating groundwater is the most likely means by which radionuclides would be removed from a geologic repository. Hence each site must be carefully evaluated according to its own unique geologic and hydrologic features, as well as its geologic history. Site selection criteria can be considered under three categories:

- features and processes at the land surfaces
- subsurface features and processes and
- reactions or changes induced either by subsurface excavation or waste emplacement.

The first category includes topography, seismicity and surface-water hydrology. The second category, subsurface parameters, involves the depth, thickness and extent, homogeneity and mineralogy of the disposal zone, and the nature of surrounding rocks and the structural geology, especially the presence of faults and fractures. From a hydrologic viewpoint, the nature and occurrence of groundwater, and the latter's direction, velocity, and volume of flow would also be considered important subsurface characteristics. Porosity and permeability of the geology to be used for the disposal zone is similarly important. The third category, induced changes, would embrace topics such as the fate of gases and liquids present in certain rock types, rock mechanics before and after excavation, thermal and radiation effects from emplaced wastes, and the possible generation of water from hydrated minerals.

Topography. Repositories should be located in relatively simple geologies. The search for such structures will be facilitated by focusing on land areas with low relief. Such topography also minimizes possibilities for differential loadings on subsurface zones.

Seismicity-Tectonics. Since repository stability is the primary method for keeping wastes in geologic isolation, sites should be located in areas with low seismicity and corresponding high tectonic stability.

Surface Water Hydrology. Sites should not be located in flood plains; at elevations above a large repository of surface water, or near streams that are capable of eroding the subsurface. If noneroding streams are present, any future changes in stream regime should be predicted.

Subsurface Geologic-Hydrologic Conditions

Depth to Isolation Zone. The isolation zones should be located at depths sufficient to avoid surface phenomena such as erosion and biological processes. They should be selected from regions of normal geothermal gradient, i.e., where the increase in temperature per kilometer of depth does not exceed 30°C.

Below depths of 1550 m, stable openings in rock salt or shale could prove difficult to maintain. Although nonplastic brittle rocks such as limestone, granite, and basalt yield stable openings below that depth, their use at greater depths could be limited by excavation costs.

Thickness and Extent of Disposal Zone. Vertical and lateral repository dimensions should be sufficient to allow for adequate heat dissipation. They should also permit effective closure of fractures extending away from the actual waste emplacement location. Where brittle host rock such as limestone is surrounded by plastic-behaving units like shale, a reduced extent or thickness of the disposal zone could possibly be tolerated.

Homogeneity and Mineralogy of Disposal Zone. Generally, intervals of homogeneous rock are preferred for radioactive waste isolation. Such rock sequences are commonly more easily mineable than nonhomogeneous structures, especially where shale partings are widely spaced. They are also less apt to contain troublesome impurities such as hydrated minerals, and they generally lack features which can lead to complicating groundwater problems. Uniform rocks also better dissipate heat. Although possibly homogeneous, potential disposal zones which contain ore deposits or are themselves mineable resources are less attractive.

Contained mineral impurities such as gypsum ($\text{CaSO}_4 \cdot 2\text{H}_2\text{O}$), other hydrated minerals such as polyhalite, carnallite, and clay minerals all can release water when heated by emplaced wastes. The quantity and distribution of these minerals should be ascertained.

Nature of Contiguous Beds. Although the potential for subsurface isolation is based primarily on characteristics of the host rock, the occurrence of several hundred meters of impervious strata adjacent to the isolation rock zone further assures containment, and is especially attractive for shallower sites. A capacity for plastic deformation in these impervious units is an added benefit. Least desirable among contiguous strata would be water-bearing units capable of flooding the repository in the event of fracturing.

Dip or Inclination of Rock Units. Except for salt-intruded structures called diapirs, the preferred inclination for sedimentary strata is less than a few degrees. In nearly flat-lying units, excavations can be made more easily and maintained, and the potential host rock is much less likely to be faulted or fractured. Where steep dips do occur, the responsible geologic forces have generally fractured the strata unless the strata are capable of plastically healing any openings.

Inclinations within igneous and metamorphic rock zones may be less critical although local conditions could be adverse.

Other Structural Geology Elements. Joints, fractures, and faults in potential waste disposal rock types are deleterious mainly because they are potential major circulation paths for groundwater. These openings also can cause physical discontinuity in rock masses and complicate mining excavations.

Major regional fault-systems should be avoided, but minor localized rock openings may be tolerated if they are either confined to near-surface intervals or can be definitely located and circumvented during repository excavation.

Brittle rocks are much more likely to be adversely affected by structurally-induced openings than plastic-behaving types. The preferred repository rock and its surrounding strata would be totally devoid of openings.

Where thick bodies of salt or other plastically deformable rock exist under sufficient load pressure, mass flow of these rocks may result in diapiric structures. Salt domes are the best known of these features. If these subsurface structures can be shown to be stable, and flow will not be reactivated from natural causes, they are commonly large enough for waste storage. Deeply buried structures must first be evaluated to assure that present-day flow is not under way. Although faults and joints are found in rocks intruded by diapiric features, openings in salt have been fully sealed as a result of its plastic behavior.

Host Rock Porosity and Permeability. The more favored repository rock types will exhibit both low permeability and low porosity to preclude movement of groundwater. Dry excavations are thus preferred for waste emplacement. Even in rocks of low intergranular porosity and primary permeability, fracture porosity which leads to secondary permeability should be as low as possible. Rocks immediately adjacent to a soluble host rock type at any repository site should preferably exhibit similar porosity and permeability properties.

Nature and Occurrence of Groundwater. Because circulating groundwater poses the main threat to geologic containment of radioactive wastes, the nature of water-bearing intervals near potential disposal zones is of utmost importance. For plastic-behaving host rocks, the vertical and horizontal distances between circulating waters and actual disposal sites can be less than for brittle rocks. Even where soluble host rocks are being dissolved near the land surface, acceptable horizontal separative distances can be calculated based on projected rates of salt dissolution, even after considering climatic-hydrologic changes. Water zones below salt layers are not common, but, in any case, vertical separation of waste-emplacement zones should be several hundred meters.

If brittle host rocks are to be considered, thick intervening bodies of impermeable shales should separate any overlying and/or underlying aquifers.

Direction, Velocity and Flow Volume of Groundwater. Even if the repository site contains a dry host rock, determinations must be made concerning water-flow directions, linear velocities and flow rates in water-bearing rock types which intervene between the waste emplacement zone and the biosphere. Such assessments must include a three-dimension geometry so as to assure accurate knowledge in the event that any of these nonhost rocks is invaded by radionuclides.

Because many variables are involved and because a rock sequence containing possibly several different water-bearing zones may not lend itself to conventional hydrologic evaluation, special or even specifically tailored tests should be considered.

Although some consideration has been given to use of low-permeability host rocks with extremely slow flow velocities as repositories, the margin of error and uncertainty about the future persistence of these conditions makes such disposal horizons less reliable. Thus, host rocks that are virtually impermeable are considered much more favorable.

It is known that under certain natural and induced conditions, argillaceous units (clays and shales) can act as semipermeable membranes. Under these circumstances, osmotic pressure differences can lead to movement of fluids, albeit at very slow rates, and a salinity gradient in the migrating fluids. The possibility that this phenomenon could affect movement of radionuclides in a host rock of this lithologic character would need to be investigated.

Reactions and Changes Induced by Excavation and Waste Emplacement

Physical-Chemical Rock Properties. Although information on the physical and chemical properties of rocks would be acquired mainly in connection with other subsurface exploration data, its greatest utility is in determining such factors as the dissipation of decay heat from radioactivity, the effects of radiation on host-rock mineralogy and the stability of mined openings. Thus, these properties are discussed within this section on induced changes.

The porosity and permeability of potential repository host rocks are important to the physical characteristics of those rock units. Since, however, these parameters are of most significance with regard to water circulation, they were discussed in the previous section on groundwater.

Included Gases and Liquids. Many rock salt deposits contain numerous small inclusions of liquids and gases. Thermal gradients caused by waste emplacement can cause small brine-filled cavities to migrate toward areas of higher temperature which slightly increase the solubility of NaCl. If the cavities comprise less than 1% of the salt volume, the amount of brine inflow should prove to be small. Some container corrosion could result, but the overall consequences would probably be minor.

Small pockets of gas, when encountered in excavated sedimentary rock sequences, can cause blowouts or buckling to the floor and/or ceiling. These effects are generally minor, and the gas can easily be ventilated. Where larger pockets of more dangerous gases such as methane or hydrogen sulfide occur, hazardous conditions are possible; improved ventilation and more cautious excavation techniques are dictated. Preexcavation detection is beneficial.

Rock Mechanical Behavior. For all potential host rocks, it must be determined that the various rock deformations induced by either the mined excavation or thermal expansion from heat-generating wastes will not ultimately breach containment. Rock-mechanics measurements are used to study conditions of the floor, ceiling and support pillars.

When brittle host rock is excavated, rock spalling, collapse, and fracturing may occur. Plastic-behaving rocks such as salt and some shales are prone to long-term creep. Backfilling in brittle rock types could mitigate against deformation; in plastic-behaving lithologies, complete and unfractured encapsulation of the emplaced wastes would be produced.

The deeper the excavation in brittle rock, the greater the overburden pressure which must be supported by pillars to prevent collapse-fracturing. At depths greater than 1500 m, plastic-behaving rock may flow even with adequately sized pillars.

Thermal Effects. Host rocks for high-level waste repositories preferably should rapidly dissipate decay heat while remaining stable at elevated temperatures. Rock salt exhibits high thermal conductivity, but also is subject to accelerated rates of plastic creep at higher temperatures. This latter property can, however, prove beneficial to sealing excavated, backfilled cavities that contain wastes. For certain brittle rocks, accelerated deformation from thermal effects could be deleterious through fracturing and possible loss of containment. For plastic, clay-rich units, elevated temperatures can lead to dehydration of clay minerals and possibly adverse effects from any liberated water. Undesirable physical changes to the rock are another concern.

Salt containing gypsum and other hydrated minerals can also be adversely affected by release of water at elevated temperatures. Mineral content of the disposal zone, amounts and rates of dewatering, and the eventual fate of liberated water must all be evaluated.

For full examination beyond dehydration effects, determination of thermal properties such as conductivity and the coefficient of expansion should be combined with testing of rock-mechanics responses under thermal loading.

Radiation Effects. Because of the disorder created in solids by irradiation, energy may be stored there, to be released later upon reordering. It is

known that gamma-radiation energy can be stored in rock salt and possibly other rock types. Also known is that thermal annealing limits the amount of energy which can be so stored. In studies on salt, annealing at temperatures above 150°C has been found to completely suppress this energy-storage phenomenon. Thus, irradiation of rock salt by high-level, high-temperature wastes appears not to pose a problem with regard to disordered energy storage.

The effects of radiation, temperature annealing, and energy storage on other minerals and rocks are not well known. Work to establish these parameters is needed.

Although dry repository environments are preferred, some water-rich argillaceous rocks which exhibit extremely slow rates of water movement might prove acceptable host rocks. Such rocks should possess high ion-exchange capacities so as to inhibit movement of radionuclides. This characteristic would complement the primary physical containment (i.e., stability) of the host rock. The effects of radiation upon ion-exchange capacity within clay-rich rocks is not now known.

Future Geologic Events

Most geologic processes act slowly over millions of years in creating structures and sculpturing the topography of the earth's surface. Certain events, such as faulting, glaciation, and volcanic activity can, however, act more quickly and produce dramatic changes, both regionally and locally. The possibility of such more rapid events must be taken into account in any site selection investigation.

For time periods of ten thousand years or more, it is reasonable to assume that another glacial episode will occur and produce stream modification, sea-level lowering, and crustal adjustment to ice loading and unloading. Pronounced climatic changes can also be inferred as well as the effects these will have on surface and subsurface hydrologic regimes.

Awareness of the impacts of these events and the calculation of acceptable buffer limits for a repository to accommodate these changes must be integrated into the total evaluation of any site.

23.2.2 Site Characterization Techniques

Rationale

It is well known that the earth's surface varies considerably in landform development, drainage routes, soil types and elevations. Geoscientific investigations have also shown that the shallow subsurface portion of the earth's crust accessible for man's use is similarly variable. Rock units are not generally homogeneous but are prone to lateral change and discontinuity. Economically important mineral deposits are not uniformly distributed everywhere.

Some geologic areas are structurally complex, while other display essentially undeformed, horizontal rock units. Granitic bedrock may be exposed at the land surface in one region and be buried beneath 10,000 m of sedimentary material in an adjacent area.

Recent technology has provided capability to investigate areas before drilling or building, and in so doing to evaluate their potential or feasibility. This is basically the mission of exploration, whether what is sought is the most promising subsurface prospect for an oil accumulation or disposal zone for high-level radioactive waste.

The evaluation of many sites and the selection of the one or several most attractive ones is, therefore, the goal of any exploration program. Methods to ultimately select sites from the myriad of potential ones available and the techniques to characterize the necessary geologic and hydrologic parameters that will make a site amenable for a repository are the focal points of this subsection.

Methodology

In planning a program designed to explore or evaluate sites for this purpose, three basic considerations must be taken into account. First, all exploration, whether the site targets are above ground, in the subsurface, or in both domains, involves initial review of generalized or regional information. Each ensuing set of exploratory-evaluative activities generates increasingly more detailed data on smaller and smaller tracts until one or more site-sized areas are ultimately selected. Even at this stage, the characterization or exploration process is not complete. This is because much more information can be acquired about the site by specific testing or technical methods directed at the narrowly defined target area.

Secondly, any site-selection program requires trained specialists, capital and time. Exploration methods have become sophisticated to the extent that a wide range of expertises can now be applied in analyzing data. Rarely is one person or technology able to furnish all the needed knowledge. There are no fast, cheap answers in site evaluation work, especially when the margin for error is small. Site characterization techniques are, therefore, expensive.

They require time to be properly designed into a total program. Interpretation of the results and decisions based upon the technology also require adequate time.

Lastly, no exploration or evaluative program can ever hope to be totally precise. This is especially true for subsurface sites where it is impossible to acquire many closely spaced data points. Furthermore, natural geologic-hydrologic variability in the subsurface combines with this less-than-total data coverage to make human interpretation a necessary ingredient. Despite man's most advanced geophysical and geologic techniques, not every well or ore drillhole based on exploration assessment finds valuable mineral resources. Exploration or site evaluation always involves some measure of risk in that this interpretive component may contain some error.

With regard to site characterization for repositories of high-level radioactive wastes, it is possible to carry the exploration process one step further than is normally the case when mineral resources are involved. No mineral resource exploration program would construct an underground shaft and mine just to test whether enough deposit existed there. But such a measure can be considered for a waste repository in order to acquire the data needed to assure that a particular site is in fact acceptable. Due to the significant containment requirements needed to insure that waste remains isolated from the biosphere, this added procedure is justified even if the additional data eliminate that particular site.

Regional Evaluation Techniques

In beginning at the regional level, it is instructive to acknowledge two additional considerations. First, and particularly for the United States, most large-size regions have undergone some measure of geologic investigation, even if at a reconnaissance level. Thus a logical starting point in an exploration program is a thorough review of all existing literature, maps and reports pertaining to the regions of interest. If, for example, the program seeks to evaluate repository sites in salt sequences, the information already available from the U.S. Geological Survey, state geologic surveys and various geological journals indicates the location of major sedimentary basins expected or known to contain salt. In fact, the geological literature contains numerous detailed articles on the configurations of such basins, their regional structural settings and the rock units present at least in surface outcrop. Therefore, a review of published sources, as found in any of several annotated bibliographies, will focus the direction of the program on well defined regional targets.

Secondly, much information can be obtained from techniques or practices whose chief aims were something other than waste repository site selection. Wells drilled to supply water for domestic or municipal use, for example, can

be used to gain information on the depths to aquifers in a basin, groundwater-flow conditions and thicknesses and types of rock or other geologic materials present. Records about old or current mining activities can pinpoint some of the more favorable areas in a basin where thick salt deposits are known to occur.

One of the most widely used techniques for regional studies is aerial-photograph interpretation, which has been recently combined with imagery from satellites under the designation remote-sensing data. Almost all regions in this country have some sort of coverage from one or both of these approaches. Although many areas lack adequate topographic map coverage, there are few regions of interest which are not now inventoried on either aerial photos, ERTS images, or U.S. Geological Survey topographic maps. Probably the most dramatic application of the increasingly available satellite photography is the delineation of major structural features whose dimensions are commonly difficult to assess using ground-based techniques. Remote sensing data have been used to map faults, locate seismically active faults, detect abnormal geothermal areas and accurately depict regional drainage paths. An excellent review of the value of remote sensing information is given by Alexander et al.⁽²⁾ Proceedings of symposia on more detailed applications are available periodically in several special publications of the National Aeronautics and Space Administration.

Depending on the degree of geologic information already available, some reconnaissance mapping to determine rock types, thicknesses, outcrop patterns, geologic structures, and terrain conditions may be needed. The specific techniques by which field geologic mapping is conducted are contained in two concise field manuals by Lahee⁽³⁾ and Compton.⁽⁴⁾

Geophysical techniques can be used both at the regional scale and the more site-oriented level. Passive geophysical methods are those that record slight variations in either the magnetic or gravity fields of the earth by means of instruments which do not emit some form of energy to accomplish the measurements. Rather, differences in rock types, thicknesses or mineral content cause the variations recorded by either gravimeters or magnetometers. Both types of instruments can be flown over large areas in aircraft, and the results can be used to delineate the exact dimensions of basins or locate subsurface structures such as large mineralized deposits, salt domes or masses of igneous rocks. In fact, gravity and magnetic surveys are best applied in either regional or basin-wide studies.

Nonpassive geophysical techniques employ energy transmitted into the subsurface as seismic waves whose return to the surface is recorded by special listening devices. The two basic methods are known as seismic refraction and reflection, depending on how the energy behaves within the stratified

subsurface. These methods are largely employed in detailed petroleum exploration. Refraction can be used to determine the thickness of sedimentary sequences in large basins. These methods are generally more commonly used for local detailed subsurface mapping and structural determinations than for regional analyses.

All geophysical methods require considerable instrumentation, highly technical personnel and appreciable interpretation. The most recent and comprehensive treatment of geophysical techniques is by Dobrin.⁽⁵⁾

There is need for information on regional seismicity or earthquake activity. Using historical records, the National Oceanographic and Atmospheric Administration periodically publishes nationwide seismic-risk maps, which depict various zones throughout the country, each zone reflecting a different degree of expectance with regard to damage from some future earthquake. Thus, for example, areas lying within Zone 3 of a seismic-risk map would be undesirable as a waste repository because the possibility of major destructive earthquakes occurring there is the greatest. Although these maps do not show the locations of all recorded earthquakes or where future events can be expected, they at least provide some overview on what seismic hazards can be expected in a given region. More detailed seismic evidence for subsequent use is available from the same agency.

At this point, the following accomplishments should have emerged from an integrated evaluation using the aforementioned regional techniques:

1. elimination of totally nonprospective regions
2. selection of one or several more promising regions for further investigation
3. understanding of the basic geologic, structural and tectonic-seismic settings of these selected region(s)
4. knowledge of the regional topography and surface-water hydrology.

Local Evaluation Techniques

The next basic evaluation step is to obtain more detailed basin-wide information so that one or more smaller local areas can be chosen. In addition to published sources, useful information which also can be used here would include oil and water-well drilling records, mining activity reports, and specific seismic-history records. This information would be obtainable from industrial files, various state and Federal agencies, professional geologic associations and mineral-resource firms already active within that area. More detailed geologic mapping and even some specific geophysical survey work could be added at this point, depending on the exact need. More specific mapping could enhance the understanding of the geomorphic development of topography, drainage history and main surface hydrologic features.

Some of the data available from existing wells (this kind of data would also be obtainable from any new test wells) might be in the form of electric or wire-line logs. The latter refer to a variety of paper-print records made by mechanical/electrical/acoustical devices used in wells (generally in petroleum wells, but also in deeper water or coal-exploration wells). By recording the responses received by these devices as they are drawn through the penetrated sequence of rocks, it is possible to learn what rock types are present, their thicknesses, porosities and average permeabilities, formational water contents and other specific features such as structural inclination. A large array of so-called wire-line logging methods can be used. Articles by Pirson⁽⁶⁾ and Schlumberger Limited^(7,8) detail the different types of logs which can be made, the theory behind their application, and the type of geologic-hydrologic information recorded by each or combinations of several used simultaneously.

Previous drilling also furnishes one added source of information, namely small chip samples of the penetrated rock types. It is very likely that some earlier drilling will have also recovered actual cored sections from some of the rocks present. If there exist any active or safely abandoned mines, as developed in salt or limestone, a direct sampling source is also available. Analysis of well cutting samples is described by Low,⁽⁹⁾ while articles by Millison,⁽¹⁰⁾ Lynch⁽¹¹⁾ and Keelan⁽¹²⁾ review various considerations for coring and the analysis of cores.

It is possible at this point in exploration to start depicting subsurface relationships in cross sections, fence diagrams, correlation charts, and a variety of subsurface mapping methods. The structural configuration and spatial thicknesses of individual rock units can be shown in map form, to complement already prepared topographic and geologic maps of surface features. Geophysical data can also be integrated into the preparation of these various subsurface mapping techniques. Badgley⁽¹³⁾ has outlined the several approaches used by exploration geologists to represent subsurface conditions on maps and diagrams.

At this stage of exploration, selection of one or more potential site areas is within the program's capability. Information has been condensed from a regional basis and supplemented with increasing types and amounts of more localized data. From previous discussion we can see that information such as the depth to potential disposal zones, thickness and extent of the disposal zone and the contiguous beds, structural attitude of the rock sequence, presence of mineral resources in the area, and location of water-bearing zones can be known with a large degree of certainty for much of the basin area.

A candidate site may now be chosen at a location somewhat removed from any previous mining, deep water-well or oil-well drilling activity. Experience has shown that the more numerous these manmade penetrations, the greater

the opportunity for artificially breaching the containment of an otherwise favorable disposal zone. If such a decision is made, there are three or four additional evaluative steps which can be taken.

Site-Area Evaluation Techniques

To verify geologic exploration-characterization studies concluded on areas slightly distant from the chosen site, one or several exploratory holes will have to be drilled. Inasmuch as their locations are known, they can be carefully plugged after all testing is concluded in order to insure future containment integrity. Drilling samples, cores and wire-line logs will all gather specific site data to be analyzed as before. Results obtained will be compared to the subsurface maps and diagrams previously prepared.

These same wells can be used to gather data on groundwater conditions, which at this point in site selection procedures is possibly the least well known parameter. Standard aquifer testing as reviewed by Ferris and others and Stallman^(14,15) can be applied, but special methods such as pumped circulation between boreholes combined with an isotopic tracer or pumping boreholes by depth zones isolated between packers may also be involved. These data, combined with existing information on the nature and occurrence of groundwater, can provide needed volume, flow rate and direction-of-flow data for groundwater units above a potential disposal zone.

If old mine openings are available in a potential host rock, such as rock salt, various rock-mechanics data can be acquired by direct field and backup laboratory measurement. Cores taken from the disposal zone at the site area can also be used in gathering rock-mechanics data. In-situ measurements for some parameters can be obtained with special instrumentation at the site area to compare data obtained from the mine area somewhat geographically removed. Obert and Duval⁽¹⁶⁾ discuss most of the salient elements involved in obtaining rock-mechanics measurements. If a test shaft is driven to the potential disposal zone at this selected site-area, additional measurements can be made.

If all the site-area data have continued to mesh favorably throughout the evaluation scheme described thus far, it appears possible to conclude that the characterization process has focused on a potentially repository site. In the absence of any major discrepancy between the newly acquired data and those projected for the site, the next logical step would be to install a shaft and begin excavation of a pilot-plant repository. As noted previously, however, data acquired within the disposal zone itself may ultimately mitigate against final approval of the site. The site-characterization process could continue and overlap with the implementation stage.

23.3 QUALIFICATION OF SYSTEMS

The primary objective for qualification methods and procedures is assessment of environmental and public safety. The assessments are specific for the site and system; they are done for two time frames: near-term and long-term.

The assessments of near-term safety concern operational safety during periods of human activity at a repository. For disposal systems, they would concern procedures and facilities used while the repository is being filled and sealed. For storage systems, they would--in addition to operational safety considerations--be concerned with attainment and maintenance of retrievability.

Assessments of long-term safety are concerned with safety after human activity or control ceases. In a disposal system, control ceases after emplacement; assessment of long-term safety, therefore, precedes implementation of a disposal system. In a storage system, anticipated long-term safety characteristics will be checked during operation because of the retrievability feature of storage. As indicated in Figure 1, the storage systems can be converted to disposal systems if projections of long-term safety are satisfactory.

The focal point for assessment of long-term safety is the potential for, and consequences of, nuclide release from a repository after human control ceases. Generic factors to consider in making such assessments are discussed below in Section 23.3.1. Methods and data for nuclide migration projections are discussed in Sections 23.3.2 and 23.3.3, respectively. Retrievability in storage systems is discussed in Section 23.3.4.

23.3.1 Qualification Factors

The starting point for selection and qualification of geologic storage and disposal systems is the geologic medium. As outlined in Section 23.2 and discussed in detail in Appendix C, geologic media will be selected on the basis of their tectonic, hydrologic, and mechanical stability, i.e., their capability to keep the wastes isolated. There are numerous geologic formations in the continental United States that are expected to have sufficient stability (Appendix C).

After appropriate geologic media have been identified, qualification of geologic storage and disposal systems becomes site-specific. As shown in Figure 23.1, definition of site-specific system concepts involves information concerning the geologic, biologic, logistical, and waste characteristics for the site. Qualification of the site system involves evaluation of geologic, chemical, and mechanical stability, operational reliability, and consequences of any nuclide release to the biosphere.

FIGURE 23.1. Areas of Assessment in Projections of Long-Term Safety of Geologic Systems

Qualification of a geologic storage or disposal system in terms of nuclide escape involves:

1. Assumptions concerning time of escape. The amounts and types of radioactivity in the repository at the time of release will depend on the characteristics and amounts of wastes originally placed in the repository and the time elapsed for decay.
2. Assumptions concerning mode of escape. There are two basic types of escape mechanisms: sudden catastrophic events that disrupt the repository, and slow degradation processes such as diffusion and leaching of waste by water.
3. Data on nuclide movement in the geologic medium. Transport of radionuclides in geologic media is a complex function of the physical and chemical properties of the medium, the physical and chemical form of the radioactivity, and the driving force for migration, which can be water flow, temperature differences, or concentration differences. Details are discussed in Section 23.3.3.
4. Estimates of the effects of radioactivity that enters the biosphere. The effects depend on local ecology and demography. Methods to estimate effects have been developed, as indicated in Section 23.3.2.

These items reflect the need for safety qualifications to be specific for the repository site and system. Dominating factors to consider will differ for the various geologic storage and disposal alternatives, as indicated below.

Near-surface geologic storage: climate, erosion, intrusion by surface waters.

Storage in deep geologic formations: maintenance of retrievability, which depends on geologic, chemical, and mechanical stability.

Disposal in deep geologic formations: geologic stability, retention of nuclides by the geologic medium, potential for water intrusion.

Seabed: geologic stability, interactions with sea water, retention of nuclides by the surrounding medium, dispersion of any released activity by ocean currents and biological organisms in the sea.

Ice sheets: movement and stability of ice sheets, interactions between contained waste and ice, dispersion of radioactivity by melting ice.

Methods for making safety evaluations for storage or disposal in continental geologic formations are discussed in Section 23.3.2; site-specific geologic and other system data (cf. Figure 23.2) are needed to implement them. Safety evaluation methods for seabed disposal are being developed (Section 25.2). There is currently no work on ice sheet qualification.

23.3.2 Nuclide Migration Prediction

Radioactive wastes are placed in geologic media with the expectation that the medium will keep the wastes isolated as long as needed to prevent significant adverse consequences to humanity and the biological environment. One aspect of qualifying a candidate geologic repository for service involves estimating the potential for, and consequences of, nuclide migration from the repository.

As indicated in Section 23.3.1, nuclide release from a repository can occur in two ways: by catastrophic events that disrupt the repository (e.g., meteorite impact), or by degradation processes that persist for long periods. The methods for assessing these types of nuclide release differ. For catastrophic events, emphasis is placed on the type and probability of the initiating event. For degradation processes, phenomena which can release radioactivity are assumed to occur; emphasis is placed on the rates and characteristics of release and migration.

Consequences of nuclide release will depend on the types and quantities of nuclides and their interaction with the environment and its inhabitants (plants, animals, and humans). Methods to estimate consequences to humans have been developed and are described in References 17 and 18.

This discussion focuses on prediction of nuclide migration under conditions of degradation release. An example of release estimation for catastrophic events is given in Section 24.2.5.

Migration of nuclides from a repository under degradation conditions requires a degradation process and a method for transporting the radioactivity through surrounding media. The agent that can initiate and sustain degradation, and transport the nuclides, is water. Evaluation of nuclide migration therefore presumes that water intrudes the repository, acts to release the radioactivity (e.g., by leaching), and transports the radioactivity through surrounding media to the biosphere. Radioactivity that enters the biosphere is then the starting point for consequence estimations.^(17,18)

Evaluation of nuclide release and migration from a repository intruded by water involves analysis of two processes: release of radioactivity from the waste material to the water, and subsequent movement of water and its contained radioactivity through surrounding media.⁽¹⁹⁾

Release of radioactivity into water can be impeded by waste form, containers, and engineered structures used as barriers (cf. Section 23.0). Containers or structures, if present, can prevent or delay contact of water with radioactivity in the waste. Waste forms can be selected to be highly resistant to attack by water. Results of analyses of the effectiveness of such barriers will depend on the design of, and materials selected for, the repository.

The geologic medium surrounding the repository (and through which the contaminated water must flow on its way to the biosphere) can impede nuclide migration. The dissolved nuclides can have complex physicochemical interactions with the geologic medium as they migrate. These interactions, called sorption, cause nuclides to move at lower (usually much lower) velocities than the water and thereby reduce, as a result of radioactive decay during holdup, radioactivity releases to the biosphere.

Sorption includes such phenomena as adsorption, ion exchange, colloid filtration, reversible precipitation, and irreversible mineralization. Some nuclides, such as the isotopes of uranium and thorium, are strongly sorbed on media that have sorption capability; others, such as strontium and neptunium, are moderately sorbed; and a few, such as technetium and iodine, are poorly sorbed by many geologic media.^(1,20) The sorption of the isotopes of a particular element is expressed as a sorption equilibrium constant, K , that is the ratio of the water velocity to the nuclide migration velocity. For particulate media the sorption equilibrium constant is related to a distribution coefficient, K_d , by the equation $K = 1 + K_d \rho / \epsilon$ where ρ is the bulk density of the medium and ϵ is the void fraction. The K_d value is a measure of the moles of nuclide in the sorbed state per unit mass of the geologic medium

divided by the moles of nuclide in the dissolved state per unit volume of groundwater when the groundwater and medium are in equilibrium with one another.⁽²⁰⁾ For faulted media the sorption equilibrium constant is related to a distribution coefficient, K_a , by the equation $K = 1 + K_a R_f$, where R_f is the surface to volume ratio for the fault. The K_a value is a measure of the moles of nuclide in the sorbed state per unit surface area divided by the moles of nuclide in the dissolved state per unit volume of groundwater when the groundwater and medium are in equilibrium with one another.⁽²¹⁾

The K_d and K_a values are dependent on a number of parameters which include the acidity (pH) of the groundwater, the concentration of dissolved salts (such as sodium chloride), the solution temperature, and for some situations the concentrations of the dissolved nuclides themselves. Laboratory experiments to measure the distribution coefficients under conditions which approximate actual conditions in nature are difficult to perform, and as a result considerable uncertainties can be expected in these measurements. Nevertheless, it is possible to obtain predictions of nuclide discharge rates to the biosphere with proper use of the approximate K values in the modeling of nuclide transport through the geosphere.

A considerable amount of modeling of radionuclide transport through particulate geologic media has been accomplished, and recent work suggests that the model for particulate media (Table 23.1) can be directly applied to faulted media as well.^(21,22) The geosphere transport model has been integrated with a biosphere transport model and applied to determine the conditions

TABLE 23.1. Characteristics of the PNL Geosphere Migration Model

1. All of the Year 2000 U.S. nuclear power economy waste is contained in a nonsalt repository surrounded by a western U.S. desert geologic medium.
2. The waste is contacted by groundwater from a typical U.S. desert starting at varying times between the Year 2000 and the Year 10,002,000.
3. The waste is leached by that groundwater at varying rates between 0.00003 and 100%/year.
4. The groundwater moves from the repository through a one-dimensional column of the medium and discharges into a surface water body.
5. The sorption equilibrium constants are based on measurements and estimations for U.S. desert subsoils.
6. The groundwater velocity is 1 ft/day.
7. The path length from the repository to the surface water body varies from 0 to 100 miles.
8. The axial dispersion coefficient is $0.008 \text{ cm}^2/\text{min}$.

under which the partitioning and elimination of selected nuclides from high-level waste could lower the risk compared to that of unpartitioned waste. (17,18,23)

The results of these studies using the model described in Table 23.1 are summarized in Figure 23.2. The axes on this three-dimensional plot are measurement scales for the three barriers which man can select or control when designing a management system for nuclear waste. The time of initial release after the Year 2000 is a measure of canister integrity and site stability from water penetration; the leach time is a measure of the resistance of the waste form to leaching, and the path length is a measure of geosphere isolation, i.e., distance to the biosphere. An increase in any of these parameters

FIGURE 23.2. Waste Management Control Surface for Incremental Background Dose with No Partitioning

results in greater isolation of the waste from man's environment. A surface can be located on this three-dimensional plot which corresponds to any assumed incremental radiation dose level (i.e., background in this case). Combinations of path length, leach time, and time of initial release which define points located inside or under the surface (i.e., between the surface and the origin) have incremental doses above background; combinations located outside or above have doses below background. Figure 23.2 shows that a waste management system for which the migration path length between the repository and the biosphere is 5 miles or more of western U.S. desert subsoil or equivalent and the leach time is 15,000 years or greater (roughly equivalent to a monolithic borosilicate glass waste form with a leach rate of 10^{-5} g/cm²/day) will have incremental doses at or below background even if water contacts the waste immediately after it is placed in the repository.

A second geosphere migration model to describe situations when water contacts the waste but either flows at some very small velocity or does not flow at all has been developed.⁽²⁴⁾ A third model to describe radionuclide migration from a salt formation is under development.^(25,26)

23.3.3. Nuclide Migration Data

Sorption Measurements

As discussed in 23.3.2, measurements of sorption (the nuclide retention capability of geologic media) are needed to predict the migration of nuclides from deep geologic isolation should the waste be contacted, dissolved, and transported by flowing groundwater.

As far as is known, all sorption measurements to date have been made using surface or near-surface soils and rocks. No sorption measurements have been performed using deep geologic media. Sorption in deep media is discussed in Section 25.1.6.

The general applicability of the surface and near-surface sorption data to deep geologic disposal has not been assessed. However, the applicability of near-surface data (taken mostly at the Hanford site) to geologic isolation under a western U.S. desert has been evaluated, and these data have been modified and used to evaluate the incentives for partitioning high-level waste.^(1,23)

Considering first the activation products present in nuclear waste, a limited number of sorption measurements have been made for Be, Mn, Fe, Co, Ni, Zn, Zr, Mo, and W; no measurements have been made for C.⁽²⁷⁻²⁹⁾ The lack of data for ¹⁴C is significant because the results of the previously referenced partitioning study (which used an estimated ¹⁴C sorption value) suggested that ¹⁴C was the most important activation product from the long-term safety standpoint.⁽²³⁾

Many sorption measurements have been made for the fission products Sr and Cs under a wide variety of conditions, and a few measurements have been made for Nb, Y, Ag, Sb, Sn, Ru, Tc, and I. (27-31) Because the existing data for near-surface soils with low organic matter content show that Tc and I are almost unsorbed, ^{99}Tc and ^{129}I are expected to be poorly sorbed by deep geologic media. (30,31) Thus, ^{99}Tc and ^{129}I may be the most mobile fission products. Because these nuclides also have combinations of high fission yield, long half-life, and significant biological uptake, ^{99}Tc and ^{129}I may be the most important fission products for the long-term safety of nuclear wastes in geologic isolation. (23) A limited number of actinide sorption measurements have been made for Th, U, Np, Pu, Am, and Cm; none for Ac and Pa; and a few measurements for the actinide decay products Ra, Po, and Pb. (27-30,32) Based on the previously referenced partitioning study, ^{226}Ra , ^{234}U , ^{238}Pu , and ^{237}Np may be the actinides of greatest concern from the standpoint of long-term geologic isolation safety. (23)

The sorption measurements to date have almost exclusively been made in the laboratory. In the future it may be practical or necessary to augment or replace the laboratory measurements by in-situ measurements at the waste isolation sites. However, in-situ measurements would increase the cost of the measurements by at least an order of magnitude.

The past experiments have been characterized by relatively high solution concentrations of the sorbing nuclides and sorption to near or complete saturation of the geologic medium sample. In geologic media, the nuclides may be at very low concentrations in the contaminated groundwater and their sorption on the medium may be much below saturation. Thus, current experimental methods may be inadequate for geologic isolation application. In addition, because the K value (water velocity/nuclide migration velocity) is used in the migration models, it may be more appropriate to measure K directly rather than the less useful but more fundamental K_d or K_a values (see 23.3.2 for the definitions of K, K_d and K_a). (33) All of these uncertainties are at present unresolved.

Oklo Phenomenon

This term refers to the natural fossil nuclear reactor which existed about 1.7 billion years ago in uranium ore deposits located near Oklo in the Gabon Republic, Africa. As a result of tectonic stability in the area the reactor remained virtually undisturbed until discovered in 1972. (34-36) The phenomenon represents a unique opportunity to evaluate nuclide migration under conditions comparable to those expected to be used for geologic storage and disposal. (37)

Occurrence of the natural nuclear reaction was deduced from measurements on ore body materials which indicated that the ^{235}U concentration in the

uranium is at present much lower than the normal 0.72% and that the concentrations of some stable isotopes formed from the decay of nuclear fission products are abnormally high.⁽³⁴⁾ The fission reactions apparently occurred because of the presence of appreciable amounts of water, which served as a moderator of neutron energy just as the water coolant does in commercial LWRs.⁽³⁸⁾ At the time of the fissioning, the ^{235}U percentage in the uranium was about 3.16%, i.e., comparable to that in LWRs. The ore deposits in which the fission reactions occurred were in clayey sedimentary formations.⁽³⁹⁾

Studies of compositions and structures of ores in the fission reaction zones are in progress. One objective of this work is to provide data indicative of the potential for migration of radioactivity from waste repositories. To date, experiments have concentrated on analyzing, by mass spectrometry, samples taken from drilling cores taken through the reaction zone.⁽³⁹⁾

Results show that the fission product elements Nb, Ru, Rh, Pd, Ag, Te, La, Ce, Pr, Nd, and Sm, and the transuranic element plutonium, have not migrated from the clayey sedimentary formation in which they were formed. Elements such as Rb, Cs, Ba, Sr, Mo, Tc, I, Kr, and Xe have moved from their formation sites. No conclusions have yet been drawn concerning the migration of ^{237}Np . Although ^{90}Sr was mobile, it decayed to unmeasurable levels before migrating from the reaction area.

23.3.4 Retrievability in Storage Systems

Geologic storage and disposal are distinguished by the techniques used for waste emplacement and the effect of these techniques on waste retrievability. Geologic disposal would use emplacement techniques that are not reversible and selected so that retrieval is not anticipated. An example is emplacement so that the waste mixes with surrounding medium and forms a somewhat homogeneous rock-waste mixture.

In contrast, geologic storage repositories would use emplacement techniques designed to permit waste retrieval. Ease of retrievability can range from readily retrievable (e.g., using rates and methods essentially the same as those used for emplacement) to retrievability using excavation and mining techniques that involve removing some of the geologic medium along with the waste.

Retrievability techniques will depend on repository design and operating practice. To achieve the readily retrievable condition, the wastes would be put into the medium so that position and configuration would be maintained. The repository would be in active use and extensively monitored. In this document, this mode of operation of a geologic storage repository is termed "provisional storage."

Provisional storage operations permit an in-service opportunity to evaluate the stability of the repository before conversion to permanent storage. Specifically, provisional storage permits evaluation of the effects of thermal radiation, and chemical interactions on the stability of the repository and maintains opportunity for readily removing the wastes if the effects of these interactions are found to make the use of the repository unacceptable for permanent storage.

A repository that is found to have acceptable stability could be converted from provisional to permanent storage by backfilling and sealing after the repository is full. The permanent storage repository would be considered to be a storage repository rather than a disposal repository because tests and procedures during the provisional storage mode of operation would have verified the basic stability of the emplaced wastes. The wastes would therefore be available for retrieval. The retrieval operations would, however, require techniques different from those used to emplace the wastes. Seals and backfills would have to be removed, and appropriate waste handling methods would have to be used.

Methods needed for retrieval of wastes from permanent storage might depend on the time elapsed since emplacement and the rate of withdrawal desired. If long times have elapsed since emplacement, degradation of the waste form may have occurred. If such degradation has occurred, it would be necessary to remove some of the geologic medium along with the waste. The amount of surrounding material to be removed would depend on the extent and nature of the degradation and the degree of decontamination of the site that is required.

The selection of the storage site would be made on the basis that there is no intention of retrieving the waste once the repository has been designated as permanent. However, a brief analysis of costs of retrieval beyond that point indicates the costs would increase with time. This is represented in Figure 23.3, which assumes a hypothetical geologic medium, emplacement technique, decontamination requirement, and withdrawal rate.

As shown by the curve in Figure 23.3, three characteristic levels of retrieval cost can be identified: Case A, representative of retrieval from provisional storage; Case B, representative of retrieval after backfill but before significant rock-waste mixing and/or degradation of waste form; and Case C, representative of conditions after significant degradation and

dispersing migration of radioactivity from the original waste form and location.

The difference in costs for conditions B and C would depend on the severity of degradation, the time of degradation, and the degree of migration and dispersion of radioactivity. Actual differences may be small, especially if dispersion of radioactivity is slight. If degradation does not occur until long after emplacement, radioactive decay, because it would reduce actions required to achieve a given decontamination level, could reduce costs for Case C to the point where they are similar to or less than those for Case B. In such circumstances the incentives for retrieval might be negligible.

FIGURE 23.3. Retrieval Cost as a Function of Elapsed Time After Emplacement for a Hypothetical Final Storage Alternative

Case C, with costs significantly higher than costs for Case B, could also be interpreted to represent conditions for geologic isolation by disposal.

23.4 DECOMMISSIONING AND CONVERSION OF GEOLOGIC STORAGE REPOSITORIES

Facilities and equipment used at geologic isolation repositories may at some point be decommissioned. A brief discussion of the alternative procedures and related technologies for decommissioning facilities and converting storage repositories from provisional to permanent status follows.

23.4.1 Facilities Decommissioning

Facilities at provisional storage repository sites would be designed, built, and operated to provide for the ready retrieval or recovery of the emplaced radioactive waste containers (see discussion in Section 23.3.4). If the retrievability option is not exercised, the repository would be converted to permanent storage and facilities could be decommissioned as discussed in Section 24.2.2. (The procedures and criteria for changing a provisional storage repository to a permanent storage or disposal repository are discussed in Section 24.2.6.) The same alternative procedures for decommissioning facilities such as reactors, fuel reprocessing plants, etc., can be used to decommission facilities at geologic storage sites. These procedures, protective storage, entombment, and dismantling, are discussed in detail in Section 15, "Decontamination and Decommissioning of Retired Facilities," and need only be modified to accommodate the deep underground workings.

The feasibility of using the dismantling procedures to decommission a geologic repository was demonstrated on a small scale in Project Salt Vault.⁽⁴⁰⁾ In this experiment, 21 canisters containing about 200,000 Ci of activity were emplaced, stored for about 19 months, retrieved, and the mine decontaminated to radiation levels sufficiently low to permit its return to its owners with no restrictions on its use.

23.4.2 Conversion to Permanent Storage

Procedures for converting a provision storage repository to permanent storage will provide for the following operations:

Backfill of underground working. The magnitude of the deformation and other geologic disturbances, i.e., subsidence, room closure, etc., which can occur around the underground workings within the protective geologic formation can be reduced by backfilling the storage rooms, access corridors, etc. While the technology for backfilling exists, the backfill material and the extent of the

backfill are dependent on the type of waste, the emplacement method, and the rock formation.

Sealing the underground workings. The mine shafts, boreholes, etc., which provide access from the surface to the underground workings, penetrate the protective geologic formations, somewhat compromising the containment. It will be necessary to seal these penetrations in such a way that the ability of the seal to preserve the integrity of the repository containment would equal or surpass that of the protective rock formation surrounding the repository. The technique, criteria, and testing methods for placing and monitoring such seals are being developed. (41,42)

Decommissioning surface facilities. After the boreholes, shafts, etc., are sealed, the surface facilities will be dismantled and removed, eliminating any problem with leakage of radioactive material from the surface facilities. Techniques to accomplish this are available and are discussed in Section 15.

Post decommissioning site control. Site control procedures would be based on government ownership of the land, and include appropriate fencing, warning signs, and a low level of surveillance. Surveillance would include the actual but infrequent patrol of the restricted areas plus the monitoring of the level of the land surface over the underground workings, groundwater, and a few fairly deep wells which approach but do not penetrate the geologic formation.

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24.0 GEOLOGIC STORAGE ALTERNATIVES

24.0 GEOLOGIC STORAGE ALTERNATIVES

This section describes alternatives for waste isolation by storage in geologic formations. Section 24.1 describes geologic storage in burial grounds; Section 24.2 describes storage in deep continental formations.

Waste retrievability and placement in stable geologic media are the common features of burial grounds and deep geologic storage. They are distinguished by degree of isolation from the biosphere. Burial grounds are located in proximity to the biosphere, i.e., in near-surface geologic media in which biosphere phenomena occur. Deep geologic storage repositories would be located in media at depths beyond those for which biosphere phenomena occur.

The near-surface depth of burial grounds places them also in proximity to surface physical processes such as erosion and deposition. Deep geologic repositories would be isolated from such processes.

The section on burial grounds describes current practice and alternatives for future practice. The section on deep geologic storage outlines factors involved in design, operation, and evaluation of alternative deep geologic storage systems.

24.1 BURIAL GROUNDS

The term "burial," as used in this section, refers to the placement of waste at relatively shallow depths in earth materials, with no intent or provision for ready retrievability at a later date. Buried wastes are subject, in some degree, to the influence of water infiltration and movement, erosion, plant uptake, animal penetration, and human activity. The consequent migration or dispersal of radioactive and other toxic materials must be kept to some minimum or acceptable level. The physical and chemical form of the waste, engineered containment mechanisms, and the surrounding earth materials can be used to attain the desired level of containment.

Shallow earth burial has been used for the disposal of radioactive waste since the inception of nuclear weapons research in the 1940's. Initially, only the Atomic Energy Commission (AEC) National Laboratories and contractor facilities handled and disposed of radioactive waste. Burial grounds were operated at each facility for the disposal of waste generated by weapons

development and other nuclear research programs. Radioactive wastes generated by private industry were initially disposed of at the AEC-operated burial grounds.

The volume of radioactive wastes generated by private industry increased dramatically during the 1950's. Burial grounds operated by the AEC at Oak Ridge National Laboratory (ORNL) in Tennessee and at the National Reactor Testing Station (NRTS) in Idaho received some of this waste. However, the majority of the radioactive waste generated by private industry was managed by sea disposal services offered by seven private companies.

Growing pressures against sea disposal led to the AEC designating regional land disposal sites which would accept waste for burial from private industry and from contractors without burial facilities. Under this interim policy, suitably packaged wastes were accepted for burial at ORNL and NRTS until such time as commercial burial sites became available.

In 1962, a commercial burial facility was opened in Beatty, Nevada. This burial ground, licensed by the regulatory branch of the AEC, provided an alternative to both sea disposal and the AEC facilities. A similar site was opened shortly thereafter in Morehead, Kentucky, and in May 1963, the AEC discontinued its policy of accepting radioactive waste from private industry. In the following years additional commercial sites were opened in other states and by 1971 six commercial burial grounds were licensed for the handling and disposal of radioactive waste.

Wastes presently buried at both commercial and Energy Research and Development Administration (ERDA) facilities contain a large variety of radioactive materials, ranging from low activity radiopharmaceuticals to high activity fission products. The hazards these wastes pose to man and his environment are primarily dependent on their radioactive properties, rather than on their chemical toxicity. The time frame of the hazard is dependent on the half-life of the specific radionuclide. The bulk of the radioactive materials at existing sites exhibit half-lives on the order of tens of years or less, and at typical concentrations will thus decay to innocuous levels in periods of less than a few hundred years. In the human time frame, the hazard period of other materials, such as ^{239}Pu with a 24,000-year half-life, differs little from that of toxic stable elements like chromium. Present policies have nearly eliminated the burial of extremely long half-life materials but such material buried in prior years remains in place.

Disposal operations are similar to conventional sanitary landfill operations, with the additional handling precautions requisite to radioactive materials. Burial in open, unlined trenches is the common practice, with each

trench containing a mixture of radionuclides and waste forms. Water is the principal medium by which the radioactive materials can migrate away from the burial trench, either by dissolution in subsurface water, or by erosive processes. Thus, site investigation and selection procedures involve extensive studies of the hydrogeology of prospective sites. Similarly, efforts are made to exclude water from the waste following burial. Less than half of the existing burial sites have detected the movement of radioactive materials away from the buried waste and none of these releases have posed any hazard to public health.

The expanding nuclear power industry has relied heavily on land burial for waste disposal in the past, and is expected to continue to do so in the future. The first portion of this section describes the experience and current practices in land burial of radioactive waste at the six commercial facilities and at five major ERDA installations. Only the commercial sites are expected to receive waste from the nuclear power industry, but experience gained at ERDA facilities is essential to a thorough understanding of present burial practices. The second part of this section describes modifications to current practices that would increase the containment capability of existing commercial sites. These modifications apply to three general areas: waste form, containment engineering, and the natural environment. Consideration is also given to possible administrative and economic modifications. The third part provides design objectives for siting and operation of burial sites in the various hydrogeologic environments present in the contiguous United States. There is a general tendency at both the state and national levels, to discourage the proliferation of radioactive waste burial sites. However, continued use of shallow earth burial for disposal of radioactive wastes will require the eventual expansion of existing sites, or development at new locations. The design objectives are provided to give insight into alternative ways to obtain acceptable containment of waste materials in the various burial ground environments.

24.1.1 Current Practices and Experience

The purpose of this discussion is to describe existing commercial and ERDA-operated radioactive waste burial facilities and technology. This understanding is basic to any consideration and evaluation of existing practices and potential modifications.

24.1.1.1 Commercial Technology

Three companies operate the six commercial burial facilities (see Fig. 24.1): Chem-Nuclear Systems, Inc., (CNS) operates a site at Barnwell, South Carolina; Nuclear Fuel Services (NFS) operates a site at West Valley, New York; and Nuclear Engineering Company (NECO) operates the sites at Sheffield, Illinois, Beatty, Nevada, Richland, Washington, and Morehead, Kentucky (also referred to as the Maxey Flats facility).

There is considerable similarity in the overall operations at the six commercial waste disposal sites. Small differences are found in wastes accepted, surface water control provisions, and other site-specific regulations.

FIGURE 24.1. Burial Site Locations

Wastes Accepted for Burial

Wastes buried at the commercial facilities originate not only from the nuclear fuel cycle, but also from sources such as educational institutions, hospitals, government installations, industrial research and production facilities, pharmaceutical manufacturers, and waste disposal and decontamination companies. Waste commonly referred to as *low-level* is characterized as mostly paper trash with varying other items such as packing material, protective clothing, broken glassware, plastic sheeting and tubing, radioactive carcasses of experimental animals, obsolete contaminated equipment, and building rubble. Higher activity wastes typically originate from the nuclear fuel cycle. Common wastes in this category include spent ion exchange resins, filters, filter sludges, evaporator bottoms, shielding, piping, instrumentation, control rods, and equipment and materials with induced radioactivity.^(1,2,3) Radioactive wastes not accepted for burial generally include materials whose inherent chemical toxic or hazardous properties are judged to be greater than their radioactive hazard.⁽⁴⁾ Materials that have pyrophoric or explosive characteristics, free liquids, and compressed gases above 1 atm are excluded from burial at all sites. Some sites accept liquids, but effect solidification prior to burial.^(5,6,7)

Limitations on package external radiation and radionuclide content also exist, but specific values vary.^(5,6,7,8,9) Generally, limitations on the quantities of radioactive materials that can be stored above ground prior to burial are applied to three categories of waste. Byproduct material includes "any radioactive material (except special nuclear materials) yielded in or made radioactive by exposure to the radiation incident to the process of producing or utilizing special nuclear material."⁽¹⁰⁾ Special Nuclear Materials (SNM) are defined as fissionable radionuclides such as ^{235}U , ^{233}U , or ^{239}Pu . Source Materials are those which can be converted in a reactor to SNM, and include materials such as natural uranium and thorium. Typical state-regulated site-specific restrictions for the onsite possession (unburied) of these are 5000 to 50,000 Ci of radioactive materials, 4000 to 60,000 lbs of source material, and 350 g ^{235}U , 200 g ^{233}U , 200 g Pu, or the weighted sum limit. Wastes with transuranic alpha-emitting contamination above 10 nCi/g are only accepted for burial at the Richland site. Specific restrictions on acceptable wastes appear to be the most restrictive at the West Valley, New York, site. In addition to those materials already listed, other excluded materials include infectious materials, poisons, insecticides, rodenticides, herbicides, fungicides, and other such materials. The West Valley facility will only accept spent resins from nuclear power plants "when incorporated in concrete or equivalent concrete containers;"⁽¹¹⁾ as a consequence generators have shipped such wastes to the Morehead and Barnwell facilities, where this special packaging or handling is not required.

Waste Packaging

No Federal or state standards have been developed for the burial of radioactive waste.⁽⁷⁾ In lieu of such standards, waste packagings are determined primarily by Department of Transportation (DOT) regulations for the shipment of radioactive materials. Most waste packaging methods presently in use are not intended to provide containment following burial since the soil is considered to be the primary container.

The bulk of the waste materials are packaged in 210-liter DOT-approved mild-steel drums.^(7,12) Very low-level wastes (generally microcurie quantities) may be shipped and buried in wood or fiberboard boxes. Current practice at all sites is to bury waste materials in the same containers in which they are received.^(6,7,11) The exception to this is material which requires transport in an expensive reusable shielded cask. This waste is packaged in a can or liner which can be safely removed from the cask and placed in a trench or other specially designed burial location.^(7,9,11) A specific provision at the West Valley, New York, facility requires all waste packages to be self-supporting and capable of remaining intact during handling and under all weather conditions.

Waste Treatment

Except for solidification of liquid wastes, little attention is given to the containment potential of waste forms. Liquid wastes commonly are treated at their source in one of three ways: mixing dewatered waste with concrete or a urea formaldehyde polymer with or without an adsorbing agent, dewatering and mixing with an adsorbing agent, or dewatering alone.

Dewatered bead resins and filter sludges are normally packaged in drums or disposable cask liners without incorporation in a matrix, but an adsorbent is sometimes added. Spent filter cartridges at some power plants are imbedded in cement and at one plant they are packaged with an adsorbent material. Liquid wastes received at Richland are stored in tanks and eventually solidified by evaporation for tank burial in place. At the Barnwell and Sheffield sites liquid wastes are stored and then mixed with cement and/or an adsorbent and placed in drums for burial. Contaminated water pumped from open, unfilled trenches is processed through a treatment plant at the West Valley site. The Morehead facility pumps trench liquids into storage tanks. The supernatants are evaporated and the evaporator bottoms and storage tank sludges are solidified and buried.⁽¹³⁾ Compaction of dry, compressible wastes has been used as a means of reducing the volume of waste going to burial.⁽¹⁴⁾

Burial Procedures

Open trenches are used at all sites as the primary burial facility. A summary of site-specific burial trench specifications and procedures is contained in Table 24.1.^(1,4,5,6,7,8,11,15) Trench design is similar at all facilities, ranging in physical dimensions from 60 to 260 m long, 8 to 20 m wide at ground surface, and 5 to 8 m deep. For all the burial sites it has been specified that the bottom of the trench shall be above the maximum ground water elevation.

At the four eastern sites, wastes are routinely covered daily to limit contact of water with the wastes. This covering may be either earth fill or a temporary covering such as a tarp, depending on the extent to which the trench is filled. Because of the arid climate at the Beatty and Richland sites, each pit is filled with waste and then covered with excavated material.

Techniques to cover and seal waste in trenches vary with the local climate, soil, and ground water conditions. Site specific procedures are described in Table 24.1. One to 3 m of soil is mounded and graded over the top of the waste, the mound being 1 to 2 m high over the centerline of the trench. At the more arid western sites no special attempts are made to further compact the fill or seal the trenches. Efforts at the other four sites involve more impermeable soils such as clay in constructing the cover, reseeding the mound, and construction of drainage fields around the mounded trenches.

Special Burial Techniques

Special handling and burial procedures are generally used for high activity beta-gamma wastes.^(7,8,12) Such wastes generally are not buried in the same trenches as the bulk wastes. At four burial sites sealed sources and other high activity beta-gamma wastes are removed from shielded shipping containers by mechanical means and placed in dry wells.⁽⁷⁾ Dry wells are 0.3- to 0.6- m-diam concrete, transite, clay, or steel pipe placed vertically in the bottom of a special burial trench. A shielding plug is used to cover individual pipes. Radiation levels through the plugs are maintained at less than 100 mr/hr by the addition of concrete. Fissionable materials are similarly isolated in special trenches, with criticality control accomplished by limiting the minimum spacing between buried packages containing significant quantities of SNM. Site specific restrictions on SNM wastes require an ability to identify and locate SNM packages authorized for burial.

Special precautions and regulations govern the disposal of gaseous wastes at the commercial sites. These wastes are generally either ^{85}Kr (krypton) or ^3H (tritium), but others have been buried. The wastes, normally received in small containers, are placed in a steel can filled with a desiccant;

TABLE 24.1. Commercial Waste Burial Ground Operations

<u>Operation</u>	<u>West Valley</u>	<u>Morehead</u>	<u>Barnwell</u>	<u>Sheffield</u>	<u>Beatty</u>	<u>Richland</u>
Burial Trench Size (meters)	180-210 x 10 x 6 deep	60-150 x 24 x 6-8 deep	140 x 15 x 5-7 deep	150 x 15-18 x 6-8 deep	260 x 12-15 x 8 deep	90 x 8 x 6 deep
Provisions for Water Collection and Containment	Trench sloped 2° to provide a low end away from waste; sump with riser pipe at low end	Bottom slopes 1° min.; 0.6 x 0.6 x 0.6 m pit at low corner; standpipe; (2 x 0.6 m) clay berm around trench	0.6 x 0.6 m drain field, with standpipes in center bottom; 1° slope to end; 1 m minimum of sand in bottom of trench	Bottom slopes to centerline and one end; 0.6 x 0.6 m ditch, filled with broken brick; sump and stand pipe	None required	None required
Waste Disposal Procedures	Trench filled beginning at one end; filled to original grade level at center	Random dumping beginning one end; filled to 0.6 m of surface	Trench filled to 1 m of surface	Packages randomly emplaced; filled to 0.6 m of surface	Entire trench filled to 1 m of surface	Entire trench filled to 0.6 m of surface
Waste Covering Frequency	Daily	Daily	Daily	Daily	When trench is filled	When trench is filled
Type of Final Cover	Excavated earth fill; compacted; topsoil added;	1 m compacted clay; mounded; reseeded	0.6 m of clay between waste and ground surface; additional mounded cover	Compacted clay cover; surface reseeded	Excavated earth fill, no compacting	Excavated earth fill; no compacting
Depth of Final Cover	Minimum of 3 m cover; mounded 1.5 m min. above grade	1 m cover mounded 0.6 m above grade	3 m cover at centerline, 1.5 m at trench edge	Minimum 1 m final cover; mounded	Minimum 2 m total; mounded 0.6 m above grade	Minimum 2 m total; mounded 1 m above grade
Other	Trench dug as filled. Packages must be self-supporting. Min. 3 m between trenches at surface	Trench pumped dry following final covering	2.5 m dirt separates wastes from different generators. Min. 3 m between trenches at surface.	Min. 3 m at surface between trenches		

this can be sealed with tar or some similar product. The container is then sealed in concrete in a clay, steel, or concrete pipe and placed in a trench. No site will accept gaseous wastes under pressures of more than 1 atm. At some sites, a 100 Ci per package limitation is imposed.^(6,7,8) A special amendment to the Morehead license allows burial in a trench without special containment when a package contains less than 100 Ci. From 100 Ci to a 1000-Ci package limit, encasement in 15 cm of concrete is required.

Disposal Location Markings/Records

Written records of waste disposals are required at all sites. While essentially the same data are maintained at the different sites, specific formats vary slightly.^(11,12) Basic data required includes package type, volume, weight, radionuclide(s) and quantity, chemical and physical form of the waste, and package radiation dose.

Permanent trench markers are generally required within one year of the closing of a trench. These consist of concrete posts located either at the ends or the corners of the trench. Information required on the markers includes the trench number, date opened, date closed, survey data on the trench location, and summary data on the wastes contained. Summary data typically includes total waste volume, total radioactivity, and amounts of source and special nuclear materials.

Burial Site Inventory

Existing buried waste inventories at the six sites, through 1973, are given in Table 24.2.⁽¹⁶⁾ All of this material is low-level solid waste in the sense that it has not originated from the first stage of a nuclear fuel reprocessing operation. The categories of material listed in the Table are defined above.

One difficulty encountered in analysis of existing commercial site inventories arises from the past use of terms like "byproduct material" and "mixed fission product" (MFP) rather than identifying specific radionuclide contents of buried wastes. For example, in preparing a detailed inventory of wastes at the Morehead, Kentucky, burial ground, 43.2% of the byproduct activity in one pit and 48% in another is listed as MFP activity.⁽¹⁷⁾

A second difficulty involves the variation in values from independent inventories at each site. There are several causes, including the lack of specific radionuclide information as discussed above, and differing interpretations of those unspecific data. In addition, the values for radionuclide content of the buried wastes have been determined largely by estimates provided by the waste generator. Sophisticated instrumentation and techniques are

TABLE 24.2. Commercial Site Waste Inventories
(As of 1/1/74)

<u>Site</u>	<u>Waste Byproduct Material</u> (curies, non-decayed)	<u>Special Nuclear Material^a</u> (grams) ^b	<u>Source Material</u> (kg) ^b	<u>Total Waste Volume</u> (meters ³)
West Valley	380000	49000	386000	55500
Morehead	1500000	327000	58900	95000
Barnwell	121000	198000	67000	23000
Sheffield	29500	34000	27000	29000
Beatty	85500	635000	24300	40900
Richland	190000	1700	5000	15500
Total at all sites	2306000	1244700	568200	258900

^a Includes approximately 80 kg ²³⁹Pu

^b Weight of radionuclides in waste

not available to determine the identity and quantity of the contaminants in most solid waste forms.

Burial Charges

The charges levied for burial of waste at most of the commercial sites are regulated by the licensing agreements. The charges were established initially to be comparable to those levied by AEC facilities prior to the existence of commercial sites. They have increased over the years, principally in response to inflation. The charges are assessed primarily on the volume of the waste package, and varied in 1974 from \$1.00 to \$1.75 per cubic foot. These charges include the actual cost of the burial operation, fees passed on to the state for establishment of perpetual care funds, and a profit margin. Special charges are levied for particular waste forms requiring special handling or disposal.

Institutional Aspects

Handling and burial of solid radioactive waste by the private sector is regulated in several ways. At the national level, the Nuclear Regulatory Commission (NRC) has responsibility for licensing waste burial sites except where its authority has been relinquished to Agreement States. The ownership and licensing of commercial sites is summarized in Table 24.3. Illinois, the only nonagreement state with a commercial radwaste burial site is presently in the process of negotiating such an agreement. The NRC also licenses the handling and burial of special nuclear materials at the commercial sites in Nevada and Washington, as the quantities of these materials handled are greater than permitted under Agreement State provisions.

The license for operation of a burial site continuously undergoes modification. This results from special petitions from the site operator to the regulating agency requesting authority to handle waste forms or quantities not previously encountered. As a result, the license is in effect an accumulation of correspondence between the site operator and the various regulating agencies which may occupy more than a meter of shelf space. Licenses are issued for a fixed time period, at the end of which the operator must reapply. The license is automatically extended during the application review period. At present all licenses are in the renewal process except the Barnwell license, which expires in July 1978.

The renewal procedure is greatly complicated by the complexity of the license and its amendments, and by continually changing state and Federal regulations. For example, several states have restricted the burial of plutonium in concentrations greater than 10 nCi/g of waste. This restriction resulted primarily from a similar restriction imposed by ERDA on its own burial

TABLE 24.3. Licensing of Commercial Sites

	<u>West Valley</u>	<u>Morehead</u>	<u>Barnwell</u>	<u>Sheffield</u>	<u>Beatty</u>	<u>Richland</u>
Site Operator	Nuclear Fuel Services (NFS)	Nuclear Engineering Co. (NECO)	Chem-nuclear (CNS) Systems	NECO	NECO	NECO
Site Ownership	State, leased to NFS	State, leased to CNS	State, leased to NECO	State, leased to NECO	State, leased to NECO	Federal, leased to state, leased to NECO
Agreement State	Yes	Yes	Yes	No, (pending)	Yes	Yes
Activities licensed directly by NRC	None	None	Handling and burial of SNM	All Burial	Handling and burial of SNM	Handling and burial of SNM
Activities licensed by State	All	All	All other	None	All other	All other
Burial of plutonium >10 nCi/gm	Restricted since 1974	Restricted since 1974	Never permitted	Permitted, but not practiced	Restricted since 1975	Pending restriction

site operators. A proposed 10 CRF 20⁽¹⁰⁾ rule change would prohibit all shallow land burial of wastes containing more than a specified level of transuranic nuclides. Such waste would be transferred to Federal custody for ultimate disposal. The present non-uniformity of regulations, and pending changes in those regulations produces considerable difficulty in the license renewal.

Hydrologic Characteristics of Existing Sites

Commercial burial grounds have been located on the basis of regional requirements for waste disposal facilities. Site selection involved the survey of several prospective sites, with the final decision based on hydrogeologic and economic factors.⁽¹⁸⁾

The single most important factor affecting the containment capability of a burial ground is the degree to which ground and surface water can contact the waste and subsequently cause migration of the radionuclides. As a result, a hydrologic assessment is required as a portion of the licensing procedure for each site. In effect, these studies provide an estimate, prior to use of the burial ground, of the degree to which ground and surface water will contact the waste following burial, the pathway of the water away from the burial site, the ion-exchange or adsorptive capability of the materials along that path, and the extent to which the radionuclide content of offsite ground and surface waters will be effected by the burial grounds.

The important characteristics related to the containment capability of each of the six commercial sites are presented in Table 24.4. Earth materials tend to be both non-homogeneous and anisotropic. Solid waste burials are universally located at shallow depths, in the transition zone between soil materials and subsurface geology. This zone has not received much attention, and the uncertainties related to subsurface conditions within a given burial ground reflect this lack of attention. The numerical values provided in Table 24.4 are either estimates of a range of values, or approximate mid-range values. Some values vary from point to point and on a seasonal and annual basis at each burial site. Permeabilities are based on a limited number of analyses, usually on disturbed material obtained from drill holes. The determination of the depth to ground water is very difficult in non-homogeneous materials of low permeability. The pits or shafts used for burial of waste are on the order of 10 m or less in depth at all burial sites. In regions where saturated zones are commonly encountered at depths of 15 m or less, some of the buried waste will be saturated with water at least some of the time. In areas where saturated zones are at depths in excess of 50 m, saturated conditions will rarely, if ever, occur within the waste material.

TABLE 24.4. Commercial Burial Site Characteristics

	<u>West Valley⁽³⁾</u>	<u>Morehead⁽⁵⁰⁾</u>	<u>Barnwell</u>	<u>Sheffield⁽¹⁾</u>	<u>Beatty⁽¹⁾</u>	<u>Richland⁽¹⁹⁾</u>
Mean annual precipitation (mm)	1000	1200	1100	900	100	200
Surficial material type	Glacial drift, silty clay/interbedded sand and gravel	Weathered shale; clay and sand	Sand and clay sand	Glacial drift; sand, silt and gravel	Alluvial sand and gravel	Clay, sand and gravel
Thickness (meters)	20-30	3-5	0-10	20-30	Over 200	Over 150
Interstitial permeability to water (cm/day)(a)	Low (0.5)	Very low (0.02)	Very low (0.02)	Variable (0.04-40.)	Variable (0.02-0.1)	Variable
Bedrock material type	Shale	Shale	Clay, sand and sandstone	Shale, claystone and coal	Metamorphic and sedimentary	Volcanic
Structure	Flat-lying	Flat-lying	Flat-lying	Flat-lying	Folded	Flat-lying
Ground water depth to shallowest saturated zone (meters)	Variable, 1-20	1-2	10-20	5-20	80-90	100
Depth of continuous groundwater zone (meters)	20	10-15	10-20	5-20	80-90	100
Depth to regional aquifer (meters)	None present	None present	200	100	80-90	100
Surface water proximity	Onsite	500 m	At site boundaries	At site boundaries	3 km	10 km
Flow characteristics	Small, perennial	Small, perennial	Small, perennial	Lake to north; small, perennial to south	Ephemeral, following storms	Large, perennial (Columbia River)
Absorptive or ion exchange capacity of material surrounding burial(a)	High	High	Moderate	Low	Moderate	Moderate
Principal flow paths away from burial	Shale fractures and lens in drift	Shale fractures	Pore spaces in sand	Pore spaces in till	Unsaturated flow in pores	Unsaturated flow in pores

(a) Interstitial permeability and absorptive capacity can be bypassed by flow along fractures or other high permeability zones.

Due to a lack of data, it is not possible to quantitatively define the adsorptive capacities of soil materials at any of the existent burial sites in the United States. This capacity varies as a function of the particular radionuclides, the soil chemistry during migration, horizontal and vertical position within the subsurface material, and the flow path through that material. Clays usually have high adsorptive capacities, silts moderate, and sands low capacities. From this principle and some measurements, the adsorptive capacities were estimated for Table 24.4.

24.1.1.2 ERDA Burial Technology

Experience with land burial of radioactive waste began with the establishment of weapons development centers under the AEC, now ERDA. Although ERDA burial sites do not currently receive waste from the nuclear fuel cycle, their long history and experience provides an invaluable supplement to the commercial experience.

Major burial grounds for radioactive waste are presently operated at five ERDA sites: Savannah River Laboratory, South Carolina; Idaho National Engineering Laboratory, Idaho (INEL); Oak Ridge National Laboratory, Tennessee (ORNL); Hanford, Washington; and Los Alamos Scientific Laboratory, New Mexico (LASL). Waste burial operations at ERDA sites are similar to those at commercial sites, but differ significantly in some regards. The following paragraphs identify both techniques which are more or less unique to ERDA sites, and aspects of ERDA sites which distinguish them from commercial sites. Operations at the five sites are summarized in Table 24.5.

Wastes Accepted For Burial

ERDA sites must handle and dispose of all onsite generated wastes. Wastes received for burial are similar to those delivered to commercial burial sites, but differ significantly in relative proportions and degree of contamination. ERDA generated wastes include considerable quantities of very large and bulky obsolete or failed equipment and building debris. Generally, attempts at effecting significant size reduction are neither specifically required nor attempted because of the close proximity of the onsite burial ground. Certain ERDA sites place requirements on the waste form acceptable for burial. An example is the requirement at the Hanford site that all contaminated highly reactive metals (e.g., sodium and potassium) be in a reacted form prior to disposal.

Waste Packaging

As at commercial sites, solid wastes buried at the ERDA sites are packaged for containment of the radioactivity during handling, transport, and any

TABLE 24.5. ERDA Waste Burial Ground Operations

	<u>Savannah River</u>	<u>Oak Ridge</u>	<u>Los Alamos</u>	<u>Idaho</u>	<u>Hanford</u>
Burial trench/ pit size	6 wide x 6 deep x variable length	3 wide x 15 long x 3-4.5 deep	8-30 wide x 120- 180 long x 8 deep	Trench: 2-3 wide x 275 long x 4 deep Pit: 30 side x 4 deep, variable length	1.5-5 wide (bottom width) x 4-8 deep x variable length
Provisions for water collec- tion and con- tainment	None; trenches have monitoring wells	Trenches sloped to one end; 15 cm metal casing as monitoring well	None	None	None
Waste disposal procedures	Random placement in trenches	Trench filled to 1 m of surface	Pits filled in layered fashion; final waste lay- er 1 m below sur- face	Pits/trenches filled to 1 m of surface	Trench filled from one end
Waste covering frequency	Covered after disposal for fire, contam- ination, radia- tion control	When trench is filled	Combustibles covered day of de- livery; others as required for con- tamination control and layering	As trench/pit is filled	Daily after deliver- ies
Type of final cover	Excavated fill to ground sur- face; mounded as necessary	Excavated mater- ial, to ground surface; few ex- perimentally sealed; ~0.5 m be- low surface; re- seeded	Excavated tuff fill with compaction by heavy earth moving equipment	Excavated soil fill; reseeded	Excavated fill to surface; mounding as necessary
Depth of final cover	Minimum 1.2 m cover, or that needed to reduce dose to <6 mR/hr at surface	Minimum 1 m to ground surface	Minimum 1.5 m total excavated tuff cover with mounding to 0.5- 1 m above grade	Minimum 1 m to ground surface	Minimum 2.5 m total, or that needed to re- duce dose to <1 mR/hr at surface
Other		Minimum 1.5 m be- tween trenches	Minimum 4.5 m between pits at surface; min- imum 15 cm crushed, compacted tuff in pit bottom prior to waste fill	Minimum 0.6 m soil in pit/trench bot- tom to underlie wastes	

temporary storage prior to burial. Only wastes received from offsite ERDA generators must be packaged to meet specific DOT-transport criteria. Much of the ERDA-generated laboratory wastes (mostly paper, plastic, rubber, glass, small metal items, etc.) are packaged in plastic bags and cardboard boxes. Where burial grounds are onsite these packages are generally collected in large metal dumpster-type containers for storage and eventual transport to site burial grounds.

Waste packaging at ERDA sites is not intended to provide containment following burial except for tritium and some high activity beta-gamma wastes. The use of asphaltic compounds to either line or completely encase tritium waste packages is employed at several ERDA locations. Complete asphalt encasement of a smaller container in a larger container (e.g., a 115-liter drum inside of a 210-liter drum) has been quite successful for containing tritium in wastes.⁽²⁰⁾

Highly active beta-gamma wastes at several sites are packaged for burial inside of concrete containers such as sealed culverts or large boxes. This is done for both radiation protection in handling and for added containment of radionuclides following burial.

Waste Treatment

Unlike commercial sites, ERDA burial grounds do not provide liquid waste treatment facilities, requiring instead that liquid waste be solidified prior to delivery at the burial ground. Compaction facilities for dry compressible wastes are provided at INEL and are planned at LASL and ORNL.

Burial Procedures

The largest volume of ERDA-site generated wastes are disposed of by burial into trenches or pits similar to those at commercial facilities. Specific written and approved guidelines exist at ERDA sites governing the locations, dimensions, and other physical characteristics of disposal pits and trenches. Actual waste burial procedures vary significantly. At sites using small trenches, the trench is generally filled prior to any covering of the waste. At sites where larger pits are employed, all combustible wastes require covering on the day of delivery as a means of contamination control and fire prevention. At INEL, operation of a waste compactor-baler generally precludes the requirement to cover wastes daily, since the 10:1 volume reduction renders the waste less combustible. At LASL, wastes contaminated with transuranics at levels below 10 nCi/g of waste are segregated in separate burial pits.

Final covering of pits and trenches is accomplished by filling to grade and, at several locations, mounding to more than a meter above grade. In

contrast to the commercial sites, no trench or pit dewatering techniques are applied at any of the ERDA sites. Establishment of a vegetative cover over completed burial facilities is accomplished at the ERDA sites to control surface erosions.

Special Burial Techniques

At INEL the presence of basalt at less than 0.3 m below the ground surface over part of the burial ground area precludes any possible excavation there. At this site, a 91 m x 30 m x 5 cm thick asphalt surface over an 8 cm gravel base has been constructed. The pad is sloped both to the center and to one end to provide drainage. Drums and wooden crates of uranium waste, and wastes containing less than 10 nCi of plutonium per gram of waste are stacked horizontally on the asphalt pad. (This procedure is not to be confused with a similar technique used for retrievable storage of transuranic material above 10 nCi/g.) When a section of pad is filled with containers it is covered with a minimum 0.5 m of earth and reseeded for erosion control. It has been found in this operation that the pad provides a good all-weather working surface.

Vertical shafts are used for disposal of several categories of waste at all ERDA sites. Shafts vary in size from 0.2 to 2.4 m in diam and from 6 to 20 m deep. High activity beta-gamma wastes are placed into such shafts by remote means. One technique used involves bringing the waste to the disposal area in a truck-mounted cask, which can be unloaded remotely through the bottom. Disposal shafts are commonly capped using earth fill and/or concrete to a minimum depth of 1 m. Radiation through the cover is usually the governing factor as to the thickness of cover applied. Concrete lined shafts are used at LASL for the most highly contaminated wastes as well as for highly reactive wastes such as contaminated sodium or potassium.

Trenches other than those used for low-level waste are utilized for disposal of high activity beta-gamma wastes. Wastes are placed directly into trenches either with no special packaging or containment, or packaged in concrete boxes or caissons. Remote handling is always required with such wastes. At INEL where these wastes are buried without added packaging, a special removable concrete trench liner and cover is employed for shielding while the trench is being filled. At Hanford, such wastes are loaded via a 0.9-m diam helical fill pipe extending from the ground surface to a concrete caisson located approximately 8 m below grade. During filling operations the caisson is vented through a HEPA filtered exhaust with an attached air sampler. Once filled, or when radiation levels become too high, the caisson is buried in place. At Hanford, large pieces of contaminated equipment are packaged into

large concrete boxes which can be transported by rail car to the burial site. There, the box is remotely lowered via a ramp into a trench for burial.

Segregation and special disposal of particularly hazardous or problem wastes is practiced at several locations. At LASL, shafts are used specifically for disposal of contaminated oils, animal tissue, tritium wastes, and classified radioactively contaminated wastes. The placement of oils in a separate shaft is to prevent fires among other combustible wastes. Shaft disposal of contaminated animal tissue keeps predators from the material. Segregation of tritium wastes provides better isolation and monitoring of this highly mobile contaminant. At several sites, contaminated pyrophoric materials such as metallic zirconium, magnesium, sodium, etc., are placed into separate and somewhat isolated trenches or shafts.

Disposal Location Markings/Records

ERDA sites are required⁽²¹⁾ to maintain permanent records of the types, quantities, and locations of buried wastes. Identification of the locations of burial pits and trenches at the ERDA sites is similar to that at commercial sites.

Inventory

Radioactive waste generated at ERDA sites results from many research projects involving nuclear materials. In addition, activated materials from reactor and accelerator research are also disposed of by earth burial. The bulk of the waste received and buried at ERDA sites is similar to that handled at commercial burial sites, differing primarily in the relative proportions of various waste types and radionuclides. Reporting requirements are different for ERDA burial sites than for commercial sites, with the emphasis placed on identifying uranium and transuranic elements. A summary of waste buried at the five major burial sites is presented in Table 24.6.

Burial Costs

Costs of waste burial at the ERDA sites has increased continuously over recent years at an average of 10% or more per year. Costs for burial ground operations in FY 1975 ranged from \$0.72 to \$2.77 per cubic foot of waste. These values include manpower and overhead costs incurred in the receiving, handling, radiation monitoring and burying of waste. Also included are the costs for construction of burial facilities, materials, site maintenance and environmental monitoring. Generally not included are the costs of packaging, transportation, waste treatment, land, and equipment. Costs also vary considerably within each site depending on the burial technique employed. For example, pit burial of the bulk of the waste generated at one site is estimated at costing about \$0.60 to \$0.65 per cubic foot of waste, while burial in

TABLE 24.6. Summary of Waste Buried at Selected ERDA Sites
(As of 7/75)

<u>Site</u>	<u>Ci</u>	<u>TRU (g)</u>	<u>U(Kg)</u>	<u>Volume (m³)</u>
Savannah River	4,410,000	7,000	79,000	279,000
Oak Ridge	(a)	13,000	100	179,000
Los Alamos	160,000	13,000	251,000	225,000
Idaho	3,579,000	361,000	288,000	140,000
Hanford	810,000	365,000	600,000	199,000

(a) Records incomplete because some were destroyed in a fire.
Estimated as less than 60,000 Ci.

unlined shafts costs from \$1.50 to \$9 per cubic foot of waste buried. Waste disposal into concrete caissons at another site costs in the range of \$60 - \$100 per cubic foot. However, the cost of waste burial operations at the ERDA sites are not clearly defined, and therefore not readily comparable. Prior to 1972, all operations were funded from site overhead making the accounting of such charges difficult. Functions such as health physics monitoring, some waste handling operations, land acquisition and environmental monitoring are still commonly included in overhead charges.

Hydrologic Characteristics of ERDA Burial Grounds

The burial grounds at the five major ERDA sites discussed in this report are located within their laboratory site boundaries. The locations of the laboratory sites were originally selected because of their remoteness from population centers. Burial grounds are typically located in the more remote portions of the site providing for considerable physical isolation. Initially, burial grounds were not located on the basis of detailed geologic assessments. Locations were selected, however, which appeared to offer an acceptable containment ability. In later years, numerous cooperative agreements were negotiated with the U. S. Geological Survey to accomplish site investigations. Presently, all ERDA sites with burial grounds employ qualified geologists or hydrologists to assist in site investigations. A summary of general site characteristics for the five major ERDA burial grounds is presented in Table 24.7. The same comments apply to this table as to the comparable one (Table 24.4) for commercial burial sites.

24.1.1.3 Combined Considerations

In the preceding paragraphs, distinctions are made between procedures at commercial and ERDA burial grounds. A continuous need is seen for ERDA burial grounds, but at about the present scale or operations. Commercial burial grounds, on the other hand, are expected to experience a rapid increase in the quantities of wastes received over the next 20 years, principally from the nuclear fuel cycle. Considerations related to the long term containment ability of burial grounds apply equally well to ERDA and commercial sites. The remainder of the discussion of current practices is devoted primarily to commercial sites; information from ERDA burial grounds is presented only as it pertains to commercial burial ground performance.

Environmental Monitoring

The majority of the commercial burial sites were initially intended to receive only very low-level wastes, sometimes referred to as "chemwipes and coveralls".⁽²²⁾ The bulk of this waste originated outside the nuclear power industry, then in its infancy. Site studies were frequently limited to a few test borings and surface examinations, considered at the time to be adequate

TABLE 24.7. ERDA Burial Site Characteristics

	<u>Savannah River</u>	<u>Oak Ridge</u>	<u>Los Alamos</u>	<u>Idaho</u>	<u>Hanford</u>
Mean annual precipitation (mm)	1100	1300	400	200	200
Surficial material type	Sand and claysand	Weathered shale and fill	Weathered tuff	Alluvial sand and gravel	Clay, sand and gravel
Thickness (meters)	0-10	0-10	0-2	1-10	Over 150
Interstitial permeability to water(a)	Very low	Very low	Moderate	Moderate	Variable
Bedrock material type	Clay, sand and sandstone	Shale	Volcanic tuff	Basalt	Volcanics
Structure	Flat-lying	Folded	Flat-lying	Flat-lying	Flat-lying
Ground water; depth to shallowest saturated zone (meters)	10-20	0-5	200-400	60-300	100
Depth of continuous groundwater zone (meters)	10-20	2-5	200-400	60-300	100
Depth to regional aquifer (meters)	200	None present	200-400	60-300	100
Surface water proximity	On site	On site	1 km	3 km	10 km
Flow characteristics	Small, perennial	Small, perennial	Small, ephemeral	Small, ephemeral	Large, perennial (Columbia River)
Adsorptive or ion exchange capacity of material surrounding burial(a)	Moderate	High	High	Moderate	Moderate
Principal Flow paths away from burial	Pore spaces in sand	Shale fractures and pores in fill	Fractures and pores in sand	Pores in sand	Pores in sand

(a) Interstitial permeability and adsorptive capacity can be bypassed by flow along fractures or other high permeability zones.

for the types of waste being received. Commercial burial sites are now receiving radioactive wastes in every category other than high-level wastes. Potential problems with containment predicted in the original site studies have materialized,⁽¹⁾ commonly involving wastes that were not anticipated during the early site studies.

Each commercial waste burial site is required, through its licensing agreement, to conduct environmental monitoring to determine if any radionuclide migration from waste emplacements is occurring. In addition, each licensing agency conducts independent monitoring programs at the sites it serves. Table 24.8, adapted from a report by the General Accounting Office,⁽²⁾ indicates the extent and frequency of existing licensee monitoring programs. The extent of the programs differs according to site environs and particular operating procedures. The sampling frequencies are in some cases specified by the licensing agreements, but in general the sampling is performed at intervals felt to be sufficient for early detection of migrating radionuclides.

ERDA facilities conduct environmental surveillance programs primarily on a large scale to cover total site operations. Table 24.9 indicates only those programs that are conducted specifically for the burial ground locations. The term *onsite* herein refers to "within the exclusion area" of the burial site, while *offsite* means "outside the exclusion area" although it may still be on the ERDA site.

Monitoring data are expected to reinforce the siting contention that radionuclide movement out of the burial trench is unlikely. In the event of loss of site containment, the data are expected to indicate source, magnitude, and hazard of the migration. Table 24.10 summarizes information from monitoring programs at the commercial burial sites; Table 24.11 gives similar information for the major ERDA sites.

Tritium migration has been separated out as a special case. Unlike other radionuclides, tritium does not require a transport media such as ground water; it is capable of migrating out of a burial emplacement in the vapor phase through unsaturated soil.⁽²³⁾ Thus, it is expected that tritium migration could be detected at all burial sites. Tables 24.10 and 24.11 indicate, however, that this is not the case. Data from West Valley indicate that early tritium movement may not be detected if sampling locations are too distant from burial trenches. Similarly, the tritium sampling program at Savannah River indicates substantial variation with sample location. Only a few of the many sample locations provide sufficient data for quantifying tritium migration. In arid climates, vapor phase migration is only detected by sampling small quantities of soil moisture in properly located borings.

TABLE 24.8. Licensee Monitoring Programs at Commercial Disposal Sites⁽²⁾

<u>Sample Media</u>	<u>West Valley</u>	<u>Morehead</u>	<u>Barnwell</u>	<u>Sheffield</u>	<u>Beatty</u>	<u>Richland</u>
Water	2 onsite streams, quarterly; offsite surface water, weekly	8 onsite wells monthly; offsite water samples, quarterly	10 onsite wells, semi-annually; offsite water supplies, annually	8 onsite wells quarterly; offsite surface water, quarterly	1 onsite well, monthly; offsite ground water, semi-annually	3 onsite wells, quarterly; offsite surface water, semiannually
Soil	Offsite quarterly	4 onsite and 4 offsite quarterly	Offsite, quarterly	Offsite, quarterly	Offsite, semi-annually	Offsite, quarterly
Vegetation	Offsite, annually	4 onsite and 4 offsite, quarterly	Offsite, annually	Offsite, quarterly	Offsite, semi-annually	Offsite, quarterly
Air	Offsite, continuous	Continuous	Offsite, continuous	Continuous	None	None
Other	Offsite animals and fish, semiannually; offsite milk, weekly	None	Offsite animals, annually	None	None	None

TABLE 24.9. Burial Ground-Specific Environmental Surveillance Programs at ERDA Sites

<u>Sample Media</u>	<u>Savannah River</u>	<u>Oak Ridge</u>	<u>Los Alamos</u>	<u>Idaho</u>	<u>Hanford</u>
Water	Onsite surface run-off samples; weekly when water is present; 8 wells on perimeter of site; quarterly or more frequent. 42 wells on an onsite grid; quarterly or more frequent; 22 wells in waste trenches, monthly or more frequently	Offsite surface water samples; continuous; no routine ground water samples specific to burial grounds; special studies in progress	Neutron moisture measurements in pit cover; monthly	2 onsite surface run-off samples when significant rainfall occurs; 1 onsite deep well quarterly. Several onsite shallow wells; annually	None specific to the burial ground
Soil	None specific to burial grounds	None specific to burial grounds; special studies in progress	None specific to burial grounds; special studies are made.	25 onsite; samples at 0, 15 and 30 cm depths; annually	None specific to burial grounds. Special studies planned
Vegetation	13 offsite samples; monthly; several onsite samples; annually	None specific to burial grounds	Special studies only	None specific to burial grounds	No routine sampling; samples are taken if surface radiation survey indicates possible activity uptake
Air	None specific to burial grounds (occasional operational sampling)	None specific to burial grounds (occasional operational sampling)	None specific to burial grounds (occasional operational sampling)	6 onsite stations; continuous	None specific to burial grounds (occasional operational sampling)
Other		Earth surface radiation survey; quarterly, more frequently in wet season	Special tritium diffusion studies	18 onsite TLD stations; semi-annually; special deep well cross-contamination studies	Earth surface radiation survey; semi-annually

TABLE 24.10. Environmental Monitoring Results at Commercial Disposal Sites

	<u>West Valley^(24,25)</u>	<u>Morehead^(27,28)</u>	<u>Barnwell⁽²⁶⁾</u>	<u>Sheffield⁽¹³⁾</u>	<u>Beatty</u>	<u>Richland</u>
Extent of tritium migration observed	On and offsite surface water; onsite ground water	On and offsite surface water; on and offsite ground water	None ^(a) Possible percolation of trench water	None ^(a) Possible percolation of trench water	None ^(a) Possible vapor diffusion in soil	None ^(a) Possible vapor diffusion in soil
Extent of other radionuclide migration observed	On and offsite surface water	On and offsite surface water; on and offsite ground water	None observed	None observed	None observed	None observed
Transport media	1. Surface contamination carried by runoff. 2. Water infiltration of trenches and overflow. 3. Effluent from liquid waste treatment. 4. Possible ground water transport.	1. Surface contamination carried by runoff. 2. Effluent from liquid waste treatment. 3. Ground water transport.	None indicated	None indicated	None indicated	None indicated
Potential causes for future problems	1. Loss of surface water control. 2. Ground water infiltration.	1. Loss of surface water control. 2. Ground water infiltration.	1. Loss of surface water control. 2. Rise of water table.	1. Loss of surface water control. 2. Rise of water table.	Drastic climate change.	Drastic climate change.
Solutions or preventions	1. Surface water control. 2. Possible engineered ground water control.	1. Surface water control. 2. Possible engineered ground water control.	Surface water control	Surface water control	Natural conditions sufficient.	Natural conditions sufficient.

(a) Environmental surveillance program may be insufficient for detection of all modes of migration.

TABLE 24.11. Burial Ground Environmental Monitoring Results for ERDA Facilities

	<u>Savannah River</u>	<u>Oak Ridge⁽²⁹⁾</u>	<u>Los Alamos⁽³⁰⁾</u>	<u>Idaho^(2,31)</u>	<u>Hanford</u>
Extent of tritium migration observed	Onsite ground water	On- and offsite surface water, onsite ground water	Onsite vadose zone	None	None from solid waste burial ground
Extent of other radionuclide migration	None detected	On- and offsite surface water, onsite ground water	None detected	None detected ^(a)	Uptake by deep rooted plants (tumbleweed)
Transport mechanisms	None observed	1. Overflow from water filled trenches 2. Ground water 3. Surface water	None observed	None observed	Tumbling tumbleweeds
Potential causes for future problems	Surface water infiltrations of trenches	Continued uncontrolled surface and ground water	Dramatic climatic changes	Surface water infiltration and movement through fissured formations	Uncontrolled plant growth
Solutions or precautions	Surface water control	1. Engineered surface and ground water control 2. Exhumation	None necessary	Surface water control	Plant species control

(a) Controversial data exist, further studies are being made to determine if migration or cross contamination has occurred.

A comparison of environmental monitoring experience from some commercial and ERDA facilities is of interest in a discussion of radionuclide migration. Particularly, information from Savannah River and Oak Ridge are pertinent since they share the humid climate of the eastern commercial sites.

Radionuclide movement observed at Savannah River has been limited to tritium, but monitoring programs at Oak Ridge, West Valley, and Morehead sites have detected additional radioelements such as cesium, strontium, transuranics, and smaller amounts of other radionuclides. Not all of these are observed at each site, and the specific isotopes and concentrations vary. The radionuclide releases from the latter three sites can be attributed to surface and ground water problems, dispersion of surface contamination, and effluents from liquid waste (trench water) treatment.

At Oak Ridge the observed migration is caused by a shallow ground water system, poor surface drainage, and impoundment of infiltrating water in the porous waste materials. The water problems at the Kentucky and New York sites are somewhat similar in that rain water infiltrates the permeable material over the trenches and is impounded in the trenches by the less permeable original material. Subsurface movement of radionuclides leached by impounded water, along fissured or more permeable zones, is documented at Oak Ridge and Morehead.^(22,29) The same type of movement may have occurred at West Valley, although studies have not yet shown this.^(3,33,34)

Differences in handling the trench water problems are of interest. At the Kentucky site, water is periodically pumped from the trenches and treated. This results in an atmospheric effluent from the treatment process and has not totally prevented subsurface or surface transport of the radioactive leachate.^(27,28,32) At the New York site, water infiltrated the trenches faster than it could flow out through less permeable surrounding material, thus the trenches eventually filled with water and overflowed. The result was potential subsurface migration due to the hydrostatic head of the perched water, and eventual surface flow of overflow seepage.^(24,25)

In both cases the releases can be attributed to water infiltration and possibly the effects of the nonhomogeneous geologic formations of the sites. Actions being taken to control and prevent releases include surface water control, surface sealing of the trenches, and pumping of water from the trenches. At both sites extensive geologic and hydrologic studies are planned to provide more information on the subsurface facets of the problem.

Data from Oak Ridge indicate that additional problems resulted from the presence of organic chelating agents in the buried waste. Such agents are common in decontaminating solutions and can be expected as a component of

wastes delivered to the commercial sites as well. Upon contact with water in the burial emplacement these agents have become mobile and readily carry otherwise immobile radionuclides. In addition, soil absorptive and ion exchange properties are not effective on such complexed radionuclides, negating the natural containment capabilities of the site.

Significance of Changes in Environmental Radioactivity

The major effect of radioactive releases from burial grounds consists of increased levels of radioactivity in the offsite environment. It is generally accepted that these radionuclide releases have resulted in no hazard to the health of man or his environment. (13,27,28,29,33,34) However, specific dose assessments have only been published for the Morehead site. (28) At that site maximum potential doses from aquatic releases were calculated to be 15 mrem/yr from tritium, 100 mrem/yr whole body dose from alpha and beta emitters, and 1500 mrem/yr to the skeleton from alpha and beta emitters. Atmospheric releases could result in whole body doses as high as 5 mrem/yr for tritium and 1.5 mrem/yr for ⁹⁰Sr. This assessment emphasizes that these are the maximum potential doses and are not likely to be approached through actual exposures. At Oak Ridge it has been found that burial ground releases to the, "Clinch River are less than 1% of the amount allowable for unrestricted use of the water by a population," but no specific dose assessment is given. (29)

Vulnerability to Damage from Natural Events

Hazards posed to existing burial sites by natural events are not significant. The main effects of events such as earthquakes, tornadoes, and hurricanes would be to disrupt or spill wastes not yet buried. The primary effects of North Atlantic hurricanes are increased rainfall and winds in eastern states. Tornadoes could cause some spreading of uncovered material. The primary defense against damage by hurricanes and tornadoes is to bury the waste as soon as possible.

In the contiguous United States approximately 90% of the seismic activity occurs in California and western Nevada. (35) Potential seismic risk varies from site to site (see Fig. 24.2). Cracks occur in wet ground and on steep slopes in moderate damage areas. Ejection of sand and water and conspicuous ground cracks are noted in major damage areas. (36,37)

Erosion, plant root uptake, and disturbance by burrowing animals are other natural processes with a potential for dispersing radionuclides in burial grounds. Most commercial and ERDA burial grounds are located in upland areas, where the potential for erosion is minimal. Some sites, such as Morehead and Los Alamos, are established in consolidated bedrock on the tops of plateaus, so that erosion is controlled by the surrounding rock materials.

FIGURE 24.2. Earthquake Damage Zones

All sites employ various forms of surface drainage control to minimize erosion during site operation, and grade or revegetate completed trenches for soil stabilization purposes. Shallow-rooted species are selected that will provide good ground cover and minimize root penetration into the waste material. Animal burrows are commonly limited to the upper meter of the soil horizon, and no specific measures are taken against them at the burial grounds. All of the control measures described above are short-term in nature, frequently requiring continual maintenance to be effective.

Decommissioning

In the context of this discussion, decommissioning of a waste burial ground is defined as those actions taken to make the site suitable for restricted public use after termination of burial operations. The term "restricted public use" is intended to mean that only surface use will be permitted, and that the area will not be otherwise restricted because of radiological hazard. Very little operational experience in burial ground decommissioning is currently available, as none of the commercial sites has yet reached its capacity. At ERDA facilities, burial sites which have been closed still remain within controlled areas, and in most cases no thought has been given to their decommissioning, as it is defined above.

Waste disposal areas in favorable climatic and geohydrologic settings offer only minor problems in decommissioning. For example, at Los Alamos a portion of one waste burial area has been surfaced with asphalt and made available for public storage of trailers, boats, automobiles, etc. While this site is still maintained as a controlled area, it serves as a useful example of productive surface-only use of a closed burial site.

Perpetual Care

Alternatives to decommissioning, as defined above, are perpetual care of the site or relocation of the waste. The former appears to be the approach currently being taken at commercial and ERDA sites. Perpetual care implies a need for monitoring and maintenance, or corrective action to avoid endangering public health and safety after a burial site operator ceases operations.

The extent of present detailed plans for perpetual care of commercial burial sites includes locating the burial sites on state or Federal land, and the collection of fees from the operator (and in some cases the requiring of performance bonds) to provide funds for such care.

The general concepts of perpetual care outlined in current plans include surveillance and maintenance of surfaces and fences existing after site closure. When operations cease at a burial site all trenches will be covered, graded and seeded, and drainage control maintained. In some cases, liquid waste treatment facilities will also be needed to process trench water. These functions will be performed by the operator of the site. If the commercial operator fails to carry out these tasks, the funds for their implementation will come from state general funds, perpetual care accounts, or performance bonds.

The states of Illinois, Nevada, New York, South Carolina and Washington collect fees for waste burial. Kentucky has not collected fees since 1972 in accordance with contractual provisions. The contracts may be renegotiated between the States and licensee changing the amount of the fee.⁽²⁾

The New York perpetual care fund is intended to apply to the adjacent reprocessing plant, high-level disposal/storage area, and low-level burial grounds. South Carolina and Nevada require performance bonds to protect the States from deficiencies in perpetual care funds should the licensees discontinue operations prior to the planned closure dates. Kentucky has three plans under consideration, and South Carolina is reviewing a study on long-term care and its financial implications.

The final alternative to decommissioning, as it has been defined above, is waste relocation. This concept is discussed in Section 24.1.2.2.

Resource Commitments

One of the principal resources consumed by land burial operations is the land itself. Sanitary landfills are the most common means of disposal for industrial and municipal solid waste. A recent survey of the United States⁽³⁸⁾ identified over 18,000 known landfills in existence. Only 225 of these were designated as hazardous waste disposal sites, with nearly two-thirds located east of the Mississippi. This indicates a considerable demand for land suitable for solid waste burial, particularly in the heavily populated East. The requirements imposed on sites for burial of radioactive and other hazardous wastes are, of necessity, more stringent than for normal municipal waste. Competition for the remaining usable sites will increase in years to come.

Various estimates have been made of the total volume of solid radioactive waste to be generated by the nuclear power industry.^(16,39) Irrespective of the total volume generated, the land area required for burial of a specified volume of waste is more or less fixed. It has been estimated that with present burial practices, an acre of land can accommodate approximately 10,000 m³ of waste.⁽¹⁶⁾ Using present estimates of future waste volumes^(16,39) between 100 and 200 acres per year will be required by the year 2000 for burial of solid radioactive waste. Present commercial burial sites vary in area from 22 to 370 acres.⁽¹⁶⁾ Location of new sites large enough to permit efficient utilization will be increasingly difficult.

Current burial practices cannot guarantee that no radionuclides will migrate out of the burial trenches. Efforts have been made at assigning maximum acceptable values to radionuclide concentrations or inventories in burial pits.⁽⁴⁰⁾ However, until such values are universally accepted, any water moving through or below the waste material must be considered to be potentially contaminated by radionuclides. In addition, the entire hydrologic section between the burial pit and the point of ground water discharge could be potentially contaminated. Thus, the actual subsurface area influenced by the burial ground is substantially greater than that required for waste burial.

24.1.2 Potential Modifications at Existing Commercial Sites

There is considerable concern over the containment capability of existing commercial burial sites, particularly when viewed in the long term.^(2,22) In the following discussions of burial ground modifications, the underlying assumption is that a burial ground should be capable of preventing undesirable contact between its contents and man's environment. The time frame of the necessary isolation is dependent on the quantity and half-life of the radionuclides in the waste material, and on the concentration of the radionuclides

considered acceptable in man's environment. Positions taken by various authorities vary from an insistence on zero release to an acceptance of releases at the maximum permissible concentrations in air or water.

Existing burial practices have resulted in migration of radionuclides from some burial sites. Although no hazard to public health has yet resulted from this migration, these releases have been deemed unacceptable.^(2,22,24,27) In keeping with the concept that burial sites should provide acceptable containment, the following modifications to present practices could be made to increase the containment capability of existing burial sites.

24.1.2.1 Engineered Containment

Waste burial containment can be improved through numerous engineered modifications to the physical aspects of wastes and burial sites. A few of these specific engineering applications are considered below.

Placement Engineering

It is estimated that present waste placement techniques result in compactable void spaces of 30 to 50% of the trench volume.⁽¹⁶⁾ Such large void spaces result in settling of the fill, with subsequent disturbances of the surface of the trench cover. This problem can easily be reduced by altering present placement techniques. Many ERDA sites use daily covering of waste as a fire control procedure. Typically, fill material is deposited on top of the waste, and then spread with a bulldozer or front end loader in the waste trench. This reduces the total volume of waste that can be placed in a given trench, but has two beneficial effects in addition to fire control. The waste material is compacted by the weight of the fill and earth moving equipment, reducing the amount of settling that occurs at a later date. This minimizes earth cracking and consequent water infiltration. In addition, the waste is completely surrounded by relatively porous earth material. This increases the contact area available for radionuclide adsorption, retarding potential movement of contaminants out of the burial trench.

One of the more extreme measures suggested for waste placement involves the complete encapsulation of waste containers in concrete. This would reduce the rate at which radionuclides are leached from the waste, and would eliminate possibilities for compaction following burial.

Waste Segregation

Some buried waste contains solidified organic complexing agents, such as EDTA, used for cleanup and decontamination. Water entering the trenches mobilizes these chelating agents which may complex other buried nuclides.⁽²⁵⁾ This has been shown to enhance migration potentials.⁽²⁹⁾ Organic complexing

agents could be encased or incorporated in a matrix so they could not come in contact with water. Alternatively, these agents could be prohibited from those burial trenches with a potential for water accumulation.

Radionuclide segregation by placement in separate trenches could be useful in the event exhumation was required. More soluble nuclides could be placed in specially lined and sealed trenches. Development of these techniques should include economic evaluation to determine if they are justified.

Burial Site Engineering

Engineered control of surface or ground water is one means for improving the containment of a given burial site. Modifications in surface drainage can be effected to reduce the quantity of water entering an open or completed burial trench. However, such modifications require continued maintenance and should not be considered for long-term controls. Similarly, techniques designed to lower the water table within burial grounds rely either on active ground water pumping or drainage systems. These cannot remain operative without continued surveillance and maintenance. Other water control techniques in use serve to impose an additional barrier between the waste and the surrounding burial media. The Barnwell site places sand in the bottom of each burial trench, to a thickness of over 1 m, to provide ready drainage of any water entering the waste. This reduces contact time and resultant leaching of radionuclides. Clay materials have been used experimentally at Savannah River in the final cover to inhibit downward percolation of water.⁽⁴¹⁾

Another procedure is possible in areas characterized by low to moderate precipitation, sandy soils, and a deep water table. If the waste material and fill is completely surrounded by more permeable material, such as a well graded sand, water will be conducted around the waste rather than through it. This procedure has been suggested for use in subsurface storage of high-level waste⁽⁴²⁾ and would function equally well for other waste forms. This technique requires development and testing.

Techniques which seal the bottom or sides of a trench without sealing the surface are detrimental. Water entering the trench will be trapped, producing concentrated leachates, and with sufficient rainfall, an overflow of the trench at the ground surface. This condition is similar to problems resulting from waste burial in relatively impermeable materials. Some ground water control systems have been proposed, but not applied. For example, a diversion ditch backfilled with coarse material could be used to intercept and divert lateral ground water flow before it reaches burial trenches.⁽²⁹⁾ Engineered containment measures can only be assumed effective over a time span of a few hundreds of years, since subsurface degradation can and does occur.

Waste Form and Packaging

Waste form and packaging can be primary safety mechanisms for inhibiting the migration of radionuclides from buried waste. Thus, the ability of a treatment product to retain radionuclides is an important criterion for the assessment of waste treatment processes. This retention ability is dependent on the physical, chemical, and radiological properties of the waste. Some of the desirable properties are: low leach rate, chemical stability, high mechanical strength, and radiation stability. Various methods for waste fixation and stabilization are discussed in Sections 8 and 10.

At present, no credit is taken for radionuclide containment by waste packaging following burial. Although this may be interpreted as a conservative approach, it discourages effective use of engineered containment. For example, migration of tritium can be reduced or eliminated through the use of asphalt materials to encapsulate the waste.⁽²⁰⁾ Research is required into interactions between waste materials, container materials, and the burial environment to identify additional containment capabilities.

24.1.2.2 Administrative Measures

Several measures for improved burial ground operation are related more to administrative actions than to technological changes. These include controls on burial ground inventories, restructuring of burial charges, modifications in institutional structures, and alternatives for site closure.

Inventory Control

Inventory control is practiced as a means for improving the acceptability of shallow earth radioactive waste burial. It may be beneficially expanded to enhance further burial ground usage. Past controls on wastes accepted in burial inventories include restraints on the burial of high-level and transuranic wastes. Future waste inventory controls could include restrictions at selected sites on curie quantities of waste buried per year, acceptance of short half-life radionuclides only, and banning of particularly hazardous radionuclides. In areas with a potential for water borne releases of radionuclides, inventory restrictions may be useful in controlling the magnitude of the release. If transport times from the burial pit to the environment can be calculated, then environmental levels of buried radionuclides can be limited by ensuring that migrating material will decay to innocuous levels during ground water transport. Finally, restraints might be placed on any radionuclides which appear to present particular problems in terms of difficulty of containment or excessive hazard upon release. Cautions must be attached to the possible alternative practices described above; they depend on data and knowledge which currently does not exist in sufficient detail for any of the commercial burial sites.

Cost Restructuring

Charges for burial of solid radioactive waste at most commercial sites are regulated by licensing agencies. However, there is no requirement that the cost of burial be tied to the effectiveness of burial. As an alternative, waste disposal charges could reflect the cost of bringing each site to a level of long-term containment designed to facilitate decommissioning. The costs to some sites would be much greater than to others, and their waste burial charges would increase accordingly. Should the price differential prove great enough, transportation to more suitable sites might become feasible.

Burial charges could also be made on the basis of the desirability of the waste for burial. For instance, an increase in the unit volume burial charges would provide a cost incentive for volume reduction or waste concentration by the waste generator. Additional charges might be imposed to provide for adequate packaging of difficult-to-contain radionuclides (such as tritium), unless special packaging requirements are met by the generator.

The regulation of burial charges and the competition among various sites, in the absence of penalties for not attaining a particular containment level, has resulted in little change in burial technology over the last 15 years. In evaluating the monetary structure of waste burials, the costs of improved operations must be contrasted with the high cost of correcting problems after they have occurred.

Institutional Modifications

There is no national policy regarding the siting, operation, and decommissioning of shallow earth burial sites for radioactive waste. Many agencies at the state and Federal level are concerned about the lack of national direction. With the exception of the control of the burial site in Illinois (soon to become State regulated), Federal regulation of burial sites pertains only to the control of SNM materials prior to burial, with an eye toward security and safeguards. State efforts toward a unified policy are hampered by the difficulties of coordinating with agencies of other states. Many problems are common to several sites, and are best solved through combined effort.

Several conferences have been sponsored jointly by the Radiation Control Program Directors (state officers), the Environmental Protection Agency (EPA), Nuclear Regulatory Commission (NRC), and the U. S. Public Health Service. (43,44) They have identified as areas requiring attention, proliferation of sites, the lack of site-specific studies on radionuclide containment, and the development of criteria for site decommissioning. Programs are being initiated in response to these concerns by the U. S. Geological Survey (USGS), the EPA, NRC, and ERDA.

Decommissioning

Decommissioning, as defined in Section 24.1.1.3, requires that the closed site be acceptable for surface-only public use. Alternatives to this are perpetual care of the site and exhumation of the waste. The most attractive of the three is complete decommissioning, with its attendant release of the land for alternate uses. Some surface-use concepts include paved parking or storage areas, green areas, or parks, that could be associated with future nuclear developments in the vicinity of the burial grounds.

Most of the current waste burial operations are based on the concept of the second alternative, perpetual site care. Control of a burial site may be required over an indefinite period to keep the public from entering a burial site, and to prevent waste products from migrating from the site. The actual planning for perpetual care must lie with the agencies responsible for the sites.

Many state officials believe that the present funding arrangements will not be adequate to meet the needs of perpetual care;⁽⁴⁵⁾ as was the conclusion of a study commissioned by the State of South Carolina.⁽⁴⁶⁾ The study recommended a new fee structure and an increased performance bonding to protect the State if the licensee ceased operations before the planned date. A Task Force of the National Conference of Radiation Control Program Directors on "Bonding and Perpetual Care of Nuclear Licensed Activities" assembled "into a single document the guidance necessary for state officials to develop bonding and perpetual care programs."⁽⁴⁵⁾ The report outlines the steps needed to be taken by the states in order to meet their financial responsibilities. The vehicles suggested for fiscal resources are the performance bond and the trust fund. At the present, no other funding arrangements are as well suited for meeting the short and long term needs for perpetual care. In determining these needs, each responsible agency should consider the recommendations of the Task Force,⁽⁴⁷⁾ establish a plan for the physical care of the site, and conduct an economic analysis at least as detailed as that performed for South Carolina.⁽⁴⁶⁾

The final alternative for a closed burial ground is that of waste exhumation and redisposal elsewhere. This is obviously a counter-productive measure. Experience in Idaho indicates that exhumation operations are difficult, expensive, and potentially hazardous.⁽³¹⁾ There are two conditions which would make exhumation appropriate: radionuclide releases from the site representing a present or future hazard to surrounding populations, or long term care and maintenance costs which approximate the cost of exhumation. Both instances would require a detailed economic analysis of perpetual care versus

waste exhumation with subsequent transfer to an alternate disposal site. Such analysis would need to consider the financial vehicle for exhumation which is not provided under current funding arrangements.

24.1.2.3 Implications of Burial Ground Releases

The significance of radioactivity released from burial grounds, in terms of the impact on the health of man and his environs, is a major factor in the acceptability of the shallow earth burial concept. It is difficult to interpret the significance of changes in environmental radioactivity because of widely divergent opinions on the long term effects of low-level irradiation. Shallow earth burial grounds represent a potential long-term source of radionuclides in the biosphere. Present evidence indicates that burial ground releases have presented no hazard, yet the pressure of public opinion and uncertainty about the effects of prolonged releases have resulted in extensive corrective studies and actions. Environmental monitoring programs can provide the knowledge necessary for evaluating burial ground impacts.

Previously, sampling point selection has been hampered by poor knowledge of site hydrogeologic characteristics. Samples taken from convenient or obvious places, such as existing test wells or around surface seeps and streams, may not reflect the extent of subsurface migration. Sampling points with a better potential for radionuclide detection could be located through more extensive site hydrologic studies. Samples are collected and analyzed at fixed intervals, and occasionally show anomalously high radionuclide concentrations. The uncertain detection potential of the sampling points, and the infrequency of the sampling, renders these anomalies difficult to interpret. Detailed site investigations are presently being conducted by the USGS at the West Valley, Morehead, and Barnwell sites, and are planned for the other commercial sites. It is expected that these studies will improve the effectiveness of environmental surveillance programs and reduce the uncertainties in interpreting the data.

Recent reports from the EPA concerning municipal and nonradioactive hazardous waste burial grounds serve to emphasize the importance of effective environmental monitoring systems for radioactive waste burial sites.^(47,48) The need for effective monitoring over prolonged time periods is demonstrated by recent incidents of ground water contamination and human poisoning resulting from arsenic compounds buried in the 1930's. In addition, the need for action to prevent further release, once evidence of migration is obtained, is illustrated by the ever expanding areas of subsurface contamination from such sources.

Few criteria are available to allow the balancing of the cost of corrective actions against their benefits. The present policy is to continue operations in the usual fashion until an observed release motivates a corrective action to improve containment. At least two general alternatives to this policy are available: numerical standards for permissible releases to man's environment, or a goal of isolating the wastes from his environment. In principle, man's environment comprises that portion of the earth's biosphere which provides probable pathways for radionuclide transfer to man. To illustrate, radionuclides migrating in subsurface hydrologic systems are isolated from man's environment until they enter a usable ground or surface water supply.

Numerical standards for permissible releases would permit burial sites to operate on much the same basis as other nuclear facilities, which are required to operate at or below stated limits for given effluent streams. However, knowledge of the hydrogeologic characteristics of existing burial sites is not sufficient to permit specification of the effluent streams with the same certainty as for other nuclear facilities. Thus, at present there is no means to ensure compliance with such a requirement.

The goal of isolating radionuclides from man's environment can in part be accomplished by improving the containment of existing burial sites. Although it is not possible to accomplish total containment for all waste already emplaced, containment can be improved by engineering means.

Future waste emplacements can be appropriately managed to reduce the need for corrective actions. The benefits of the isolation goal could include the reduction of the long-term cost of burial: "An ounce of prevention is worth a pound of cure."⁽⁴⁹⁾ Similarly, actions to minimize the release potential will reduce the public burden of perpetual surveillance and maintenance.

24.1.2.4 Summary

The modifications suggested previously could all be used, in various combinations, to increase the containment capability of existing burial sites, and to generally improve the effectiveness of shallow earth burial practices. The benefits and disadvantages of the modifications are summarized in Table 24.12. All of the modifications will likely result in cost increases beyond the present rates. However, the cost savings associated with increased containment, on both the long and short term, may outweigh these increased costs.

TABLE 24.12. Benefits and Disadvantages of Potential Modifications

<u>Modification</u>	<u>Benefits</u>	<u>Disadvantages Beyond Cost Increase</u>
<u>Engineered Containment</u>		
Waste compaction with fill in trench, or encapsulation	Reduced surface subsidence and water infiltration. Increased adsorption area	Reduction of waste/trench volume ratio
Surface drainage or ground water control	Reduced water in trenches	Requires maintenance
Special trench construction	Reduced water contact time with waste	Requires maintenance or trench pumping
Internal drains	Reduced water infiltration	Requires pre-compacted waste below cover to avoid cracking.
Clay in final cover	Reduced water contact time	Suitable only for areas with good subsurface drainage
Coarse material "bypass"	Reduces subsurface migration	Water accumulation in trenches
Sealing only sides and bottoms of trenches		
<u>Waste Form</u>		
Immobilization of complexing agents	Reduced leaching of radionuclides	Volume increase
Solidification:	Reduced leaching, mechanical stability	Increased volume
Cement or equivalent		
Urea formaldehyde	Same as above, but cheaper	Uncertain
Asphalt or bitumen	Reduced leaching, mechanical stability, with volume reduction for liquids	Combustibility
Incineration	Radionuclide concentration, volume reduction, final product inert	Need for ash fixation
Packaging for containment	Reduced migration from container, provides for retrievability	Short-term measure
<u>Inventory Control</u>		
Radionuclide segregation by trench or site	Aids selective exhumation. Provides better migration control	Increased handling and/or transportation
Restriction of specific radionuclides or waste forms	Reduced hazard, reduced migration potential	Need for alternate disposal site
Limitation on quantities of radionuclides	Controls release rate, or eliminates hazard	Requires alternate disposal site for excluded materials. Hydrologic data not yet available.
Cost restructuring	Provides incentives for increased burial effectiveness or waste treatment	May make some sites non-competitive
Improved environmental monitoring	Increased knowledge of migration potential	Requires hydrologic data not yet available
Modified institutional structure	Improved national coordination of waste burial	

24.1.3 Burial Site Design Objectives

The containment capability of a burial ground is dependent on the physical and chemical form of the emplaced waste, the retaining ability of the natural system surrounding the waste, and any engineering modifications to the system. In general, migration of radionuclides from buried waste to man's environment involves two separate phenomena: dissolution of radionuclides in the waste by water in contact with it, and movement of the liquid containing the radionuclides away from the burial emplacement at a sufficient velocity that they enter man's environment during their hazardous lifetime. Most physical and biological processes that mobilize and transport radionuclides can be described in these general terms. Thus, measures that impede either of these phenomena will increase the containment ability of a burial site.

Criteria for burial ground selection have been suggested by numerous investigators.^(50,51,52) These various criteria differ regarding the degree of containment required. The fundamental consideration in site selection is the potential for radionuclide transport by ground and surface waters. Maximum containment is obtained if there is no transfer of water into or out of the waste materials, by physical or biological means. Certain areas in the arid west might be located where this could be realized, but throughout most of the United States some compromise on this position is required. The extent of the compromise can only be determined by the collection of site-specific data regarding all mechanisms for water and radionuclide migration.

The purpose of site investigations is to determine the extent to which the natural system aids burial ground containment. Site investigations should define the nature of the hydrologic system within and around the site, i.e., the rate and quantities of water movement into the waste, as well as rates and pathways from the waste to man's environment. The specific types of data required, and the relative difficulty of obtaining them, have been discussed elsewhere.⁽⁵²⁾

Natural materials exhibit varying degrees of ion exchange or adsorptive capacity. This capacity results in a retarding of the rate at which radionuclides move relative to the fluid in which they are contained. Most adsorptive phenomena are reversible; the amount of a particular radionuclide adsorbed on the solid material is proportional to the amount in the contacting fluid. Further, the proportionality is dependent on the nature of the solid media, as well as the composition of the fluid. Some materials, such as organic complexing agents, tend to reduce or negate the adsorptive ability of solid materials. If flow is occurring through highly porous zones, or fractures, the contact area available for adsorption may be quickly saturated.

This will result in an initial reduction of the migration rate, but subsequent to saturating the adsorption capacity the radionuclides will move at the rate of the migrating fluid.

Hydrologic investigations typically rely on measurements at selected points and infer the behavior in the intervening areas. To be effective, this requires that the points selected be representative of the entire system and that no significant heterogeneities exist. This principle works well in homogeneous porous materials where the flow is uniform throughout. However, many systems are characterized by extreme heterogeneity. For example, the bulk of a shale may exhibit extremely low permeabilities to water movement. But widely spaced fractures or bedding plane separations commonly exist in shale, with permeabilities orders of magnitude higher than the bulk permeability. Thus, the majority of the water flow may occur along these pathways, and such flow may not be detectable by sampling points located in the bulk material. The important mechanism retarding radionuclide migration is the adsorptive capacity along the flow path. Measurements made of the adsorptive capacity of the bulk material do not reflect that of the fracture surfaces.

Laboratory measurements of permeability and adsorptive capacity are useful for inferring the behavior of hydrologic systems only to the extent that these measurements characterize the principal flow paths. In situ measurements are preferable, but difficult to obtain in heterogeneous material, or in materials of low permeability.

Simulation modeling, either analog or digital, is useful for interpreting the data obtained from hydrologic studies. Hydrologic transport models can be developed, with present techniques, which will adequately describe the movement of radionuclides (or any other hazardous material) through some hydrologic systems.⁽²³⁾ Those systems which are too complex to be adequately modeled with a reasonable level of effort will likely prove unacceptable as burial ground locations. Once developed, a simulation model may be used to a limited extent to predict future behavior of the system. The reliability of this predictive ability is strongly dependent on the degree to which data must be extrapolated.

The principal impediment to adequately describing the hydrologic system within and around a burial site is the natural heterogeneities of earth materials. The simpler the system, the easier it can be quantitatively described. Introduction of foreign materials, as in the burial of waste, adds additional components to the natural complexity. Thus, the degree to which the containment ability of a particular burial site can be defined depends primarily on the resources that can be committed to a study of the natural system. The

location of new burial sites, to replace or augment existing ones, will involve the consideration of many alternate sites. State and Federal officials have pointed out the need for minimizing the total number of radioactive waste burial sites.⁽⁴⁴⁾ Realization that a site is not providing acceptable containment could lead to its closure and a further proliferation of sites. Thus, sites selected for burial grounds should provide optimum conditions for containment of radionuclides during their hazardous lifetimes.

There is considerable variability in the ability of hydrologic environments to provide radionuclide containment. In describing these various environments, it is useful to divide the United States into several geographic regions, as illustrated in Fig. 24.3. Each region is characterized by a particular combination of precipitation, evaporation, surficial geology, ground water and surface water conditions, as presented in Table 24.13 (Ref. 53). Within each of the regions delineated, a wide variety of conditions prevail, dependent on local soils, geology, topography, and precipitation patterns. The characteristics listed in Table 24.13 describe the region as a whole, rather than any specific point within it.

Shallow earth burial grounds differ in two significant ways from other facilities with a potential for releasing radionuclides. First, no well

FIGURE 24.3. Physiographic Regions

TABLE 24.13. Characteristics of Generic Types

<u>Region</u>	<u>Mean Annual Precipitation (millimeters)</u>	<u>Approximate Precipitation/ Evaporation Ratio</u>	<u>Nature of Surficial Material</u>	<u>Probability of Saturated Zones Shallower than Fifty Feet</u>	<u>Nature of Surface Water Occurrence</u>
1. Northeast	1000-1300	1.5	Glacial deposits	High	Perennial in most drainages
2. Appalachian	1000-1300	1.0	Thin residual soils	High	Perennial in most drainages
3. Southeast Coastal	1000-1500	1.0	Alluvial sand and clay	Moderate	Perennial in most drainages
4. Midwest	500-1000	0.8	Glacial deposits	Moderate	Perennial in most drainages
5. Mid-Continent	400-1000	0.3	Glacial deposits in north, alluvium in south	Moderate	Perennial in major drainages
6. Rocky Mountains	200-800	0.5	Thin residual soils and alluvium	Low	Perennial in major drainages
7. Intermontaine	100-500	0.1	Alluvial deposits	Low	Ephemeral in most drainages
8. Pacific Mountains and Coast	200-2500	0.1-2	Glacial deposits in north, alluvium in south	Low	Perennial in major drainages

defined discharge points exist. Effluents from a burial site must be determined indirectly through environmental surveillance and inference. Second, it is extremely difficult to effect a change in effluent magnitudes should the need arise. In some cases, it may be easier to establish a new burial facility. Further, the migration mechanisms, biological uptake, and long term effects of low concentrations of most radionuclides are not sufficiently known to permit setting of limits for environmental releases from burial sites. Such deficits in knowledge, and difficulties in taking remedial actions, dictate that future burial sites be designed with the goal of isolating the wastes from man's environment.

Deficiencies in the natural containment system which compromise isolation can be offset by various combinations of waste form and site engineering. The relative importance of these three factors will vary from site to site, as well as the degree to which they can be applied to obtain acceptable isolation of the waste. The remaining portions of this section describe the natural containment capabilities of various burial ground environments, and possible means of augmenting that capability.

24.1.3.1 Generic Classification

The present capacity of commercial radioactive waste burial sites is expected to be exhausted about 1995.⁽¹⁶⁾ The site investigations required to allow expansion of the existent sites, or addition of new ones, will require substantial lead time. Some primary considerations for site selection follow.

Northeast Region

In the Northeast Region, nearly all municipal/industrial water supply is derived from surface water sources. The surface material is either glacial deposits of silty clay and sand, or thin residual soils. The precipitation is high in comparison with potential evaporation, and the soils are wet to moist all year. The subsurface materials in which burial sites could be located are generally of low permeability to water movement, and have saturated zones within a few meters of the surface. Drainage of precipitation generally occurs as surface water runoff.

Burial grounds will collect moisture to the point of saturating the waste material, unless special precautions are taken to avoid such circumstances. The flow distance to surface water will be measured in tens to hundreds of meters at most, and a significant portion of the flow will be on or near the surface. Subsurface flow is likely to occur through fractures or sand lenses, providing only limited opportunity for ion-exchange or adsorption mechanisms to retard radionuclides.

The subsurface and hydrologic conditions characterizing this region are particularly difficult to define. Design emphasis should be placed on waste form and engineered containment to offset possible inadequacies in the natural containment mechanisms. Environmental monitoring under these hydrologic conditions can be extremely uncertain and requires extensive monitoring networks.

Appalachian Region

This region differs from the Northeast in two principal regards: soils are almost exclusively thin residual developments, with bedrock at or near the surface, and some major water supplies are derived from ground water sources. Burial sites must be located outside the recharge zones for these aquifers, in relatively impermeable bedrock materials such as shales or siltstones.

Design considerations for burial sites in this region are similar to those in the Northeast Region. Location of burials within fractured bedrock materials necessitates maximum attention to containing radionuclides within the burial site. Opportunities for radionuclide adsorption on the earth materials will be limited by flow along fractures or bedding planes. Elimination of water from the burial trenches is of principal importance in ensuring radionuclide containment. Again, monitoring is exceptionally difficult in this environment due to ground water flow through fractures.

Southeast Coastal Region

The relatively flat-lying sedimentary materials underlying this region reach thicknesses of thousands of meters in some areas. Surficial materials are generally unconsolidated sands and silts, with a high clay content due to rapid weathering. Most of the region is underlain by ground-water aquifers which are used extensively for municipal or industrial water supply. Frequently these aquifers are overlain by relatively impermeable shales or clay, with recharge occurring primarily in upland areas. Subsurface materials in the region are relatively homogeneous. Saturated zones are not common near the surface, and subsurface flow occurs through interstitial spaces.

Description and understanding of the hydrologic regime is a reasonable goal. The development of saturated zones within the burial trenches can be avoided through attention to surface drainage and condition of the final cover. This, in combination with acceptable waste forms, reduces the importance of unusual engineered containment. The nature of the subsurface material provides considerable opportunity for radionuclide adsorption. In this environment, sufficient knowledge could be obtained to allow design of an effective environmental monitoring system.

Midwest Region

This region is characterized by surface accumulations of glacial deposits, generally in excess of 10 m. The deposits are similar to those in the Northeast, with significant amounts of sand and gravel. Precipitation is somewhat lower than potential evaporation. Saturated zones are commonly at depths of 5 m or more, but water movement may occur preferentially through sand or gravel lenses interbedded with relatively impermeable clay or silt. Substantial use is made of ground-water aquifers which are frequently recharged by the saturated zones in the overlying glacial material. Thus, care must be taken to locate burial grounds where aquifers are overlain by substantial thicknesses of relatively impermeable material.

The Midwest Region is somewhat intermediate regarding the reliance that can be placed on natural containment mechanisms. Preferential movement of water through sand or gravel lenses may bypass much of the adsorptive capacity of the material surrounding the burial site. Emphasis placed on waste form will reduce the need for engineered containment if appropriate water control measures are exercised. Due to the heterogeneous nature of the surficial materials, effective monitoring will be difficult.

Midcontinent Region

The Midcontinent Region is characterized by relatively thick accumulations of glacial deposits or alluvial materials. Evaporation is higher than precipitation throughout, and soil materials may be dry much of the year. The combination of relatively permeable surficial materials with a low precipitation/evaporation ratio results in saturated zones not being common near the surface. Substantial use is made of ground-water aquifers, and burial sites should be located away from recharge zones.

This region is similar to the Southeast Coastal Region regarding the importance of the various containment mechanisms. However, due to lower precipitation, natural systems offer a somewhat greater containment potential. With careful site selection, sufficient knowledge can be available to permit effective environmental monitoring.

Rocky Mountain Region

Here, thick deposits of alluvium alternate with bedrock plateau or mountainous areas. Precipitation is concentrated at higher elevations, where it equals or exceeds potential evaporation. The opposite is true at lower elevations, so that most surface runoff originates in the upland areas. Ground water occurs in the alluvium and in porous bedrock materials, commonly at depths of tens to hundreds of meters, with recharge occurring over wide areas. Hydrologic and climatic regimes may vary considerably due to significant variations in land elevations.

The most suitable locations for burial grounds in this region are at lower, drier elevations. The unconsolidated surficial materials are generally sand and silt, providing good opportunity for adsorption of radionuclides. Effective environmental monitoring in areas with ground water near the surface may require selection of a site underlain by relative homogeneous material.

Intermontaine Region

This physiographic province consists of widely separated mountain ranges with very thick accumulations of alluvial material between. Potential evaporation is greatly in excess of precipitation. Nearly all surface runoff originates in the mountainous areas and is absorbed rapidly as it flows onto alluvial areas. Short term, intense runoff events are associated with thunderstorm activity. Ground-water zones are recharged primarily along mountain fronts, and where surface runoff reaches into the alluvial valleys. As a result, depth to regional aquifers may vary considerably, reaching depths of hundreds of meters in some areas, but near the surface in recharge zones. These aquifers are used extensively for water supply throughout the region.

In this region, more than any other, reliance can be placed on the natural system to provide containment. The unconsolidated materials are typically quite sandy, with relatively low adsorption capability, but water is not likely to accumulate in the burial trenches. More latitude is possible in waste form, and reliance on engineered containment can be minimal. Environmental monitoring may be difficult due to the general absence of moving water near the surface.

Pacific Coast Region

Throughout most of the Pacific Coast Region, precipitation is significantly less than potential evaporation, except in the far northwest and at higher elevations. In addition, precipitation occurs predominantly in the winter months at all elevations. This region is similar to the Rocky Mountain Region in the nature and occurrence of alluvial material within and adjacent to the mountainous areas. Nearly all surface runoff originates in the mountains, but depths to regional aquifers are less than in the Intermontaine Region. Extensive use is made of these aquifers, but most of the water supply is derived from surface water sources.

Development of saturated conditions within buried waste is not likely, unless burial sites were located in higher valleys where subsurface materials are thin. Similar to the Intermontaine Region, considerable latitude in waste form is possible. However, some additional engineered containment may be required to offset a reduced reliance on the natural system. Adequate environmental monitoring depends on detailed knowledge of specific site characteristics.

24.1.3.2 Role of Shallow Earth Burial in Waste Disposal

Radioactive solid waste in all categories other than high-level and transuranic materials is currently being disposed of by shallow earth burial. Modifications to present practices, as suggested above, could increase the containment capability of burial sites, with the goal of maximum containment of radionuclides. Burial is a viable means of disposal for radioactive wastes with a hazard* measured in hundreds of years. Effective utilization of this disposal mode requires emphasis on stable waste forms, hydrologically oriented containment engineering and the natural materials surrounding a burial site. Decisions regarding the relative importance of these mechanisms must be based, in part, on economic considerations. However, in no event should economy override the importance of controlling or restricting radionuclide releases to the environment.

24.2 STORAGE IN DEEP CONTINENTAL GEOLOGIC FORMATIONS

Design and qualification of deep geologic storage systems will be specific to the site, geologic medium, and wastes emplaced. After a geologic medium and site have been selected, alternatives are derived from design options that can satisfy requirements for operational safety, repository stability, and retrievability. Factors involved in designing to meet these requirements are described in this section.

24.2.1 Waste Types

The three general categories of waste considered for final waste storage are described in detail in this section.

1. Solidified High-Level Wastes

- a) Origin--The origins of these wastes were described in Section 2 and elsewhere.⁽⁵⁴⁾ The following description is reproduced from Reference 54.

High-level wastes are defined in federal regulations as "those aqueous wastes resulting from the operation of the first cycle solvent extraction system, or equivalent, and the concentrated wastes from subsequent extraction cycles, or equivalent, in a facility for reprocessing irradiated reactor fuels." These wastes contain virtually all of the nonvolatile fission products, several tenths of one percent of the uranium and plutonium originally in the spent fuels, and all the other actinides formed by transmutation of the uranium and plutonium in the reactors. They can be generally characterized by their very intense, penetrating

* *Hazard* is not identical with the material's radioactive half-life. It is also related to the specific toxicity of the nuclide and migration conditions.

radiation and their high heat-generation rates. Regulations call for these wastes to be solidified within 5 years after they are generated and for the resultant stable solids to be shipped to a federal repository within 10 years after the liquids are generated.

- b) Physical and Chemical Compositions--Solid waste forms and solidification processes are described in Section 6. Calcines and borosilicate glasses are among the forms being considered. The physical and chemical properties of these and other forms are under investigation (see Section 6).⁽⁵⁵⁾
- c) Radioactivity and Thermal Power in Solidified Wastes--Information on these characteristics of solidified wastes for UO_2 -fueled light water reactors is presented in Table 24.14. The values for the power per unit volume of solid waste (final column) are based on an assumed value of 0.05 m^3 of high-level solid waste per metric ton of heavy metal fuel charged to the reactor. Selection of this value was based on an analysis of reprocessing and solidification processes which showed the volumes of solid wastes to be in the range 0.03 to 0.06 m^3 per ton of LWR fuel (0.03 to 0.1 m^3 per ton of LMFBR fuel), and indicated that the higher volumes are the most likely.⁽⁵⁴⁾

TABLE 24.14. Radiation Characteristics of High-Level Waste from Reprocessing PWR-U Fuel (Enriched uranium oxide, power = 30 MW, burnup = 33,000 MWd/MTU)⁽⁵⁴⁾

Time Since Reprocessing, (a) Years	Power from Waste from 1 MT of Uranium, kW			Total Power Per Unit Volume of Solidified Waste, kW/m ³ (d)
	Fission Products (b)	Actinides (c)	Total	
0	19.2	0.72	19.9	392
3	2.93	0.091	3.02	60
5	---	---	1.78	35
10	1.01	0.067	1.08	21
100	0.105	0.0097	0.115	2.1
1000	0.0225	0.0022	0.025	0.35

- a. Fuel is reprocessed 150 days after its discharge from the reactor.
- b. Assumes that all of the tritium and noble gases and 99.9% of the halogen fission products in the spent fuel are separated from the high-level waste during reprocessing.
- c. Consists of all the actinides remaining after removal of 99.5% of the uranium and plutonium during reprocessing.
- d. Assumes 0.05 m^3 of high-level solid waste per metric ton of uranium.

Table 24.15 shows similar type information for mixtures of solidified from UO_2 fueled LWRs (2/3) and from Pu-recycle fueled LWRs (1/3).

The radiation characteristics and volumes of solidified high-level waste from reprocessing LMFBR fuels will probably be similar to those of waste from UO_2 fueled LWRs.⁽⁵⁴⁾

- d) Packaging and Acceptance Criteria--High-level wastes will be encased in sealed cylindrical metal canisters, probably constructed of 304L stainless steel (Section 6). The design and acceptance criteria have not been established in detail, but some of the important considerations which will affect these criteria can be set forth:

Maximum length--Lengths of 3 to 4.5 m have been suggested (Section 6). The head room available in a given underground storage/disposal facility may limit the acceptable length. A typical canister used in many conceptual design studies is 3 m in length.

Maximum diameter--The maximum acceptable diameter of a waste canister will depend upon whether geologic storage or waste disposal is contemplated. An initial period of retrievability exists during waste storage. During this period, the holes in the mine floor would be specially prepared, possibly by installation of a liner or casing, and the room itself would be left open for easy access. Also, during this time it may be desirable to doubly contain the canister by overpacking with a second canister (Section 6).

TABLE 24.15. Radiation Characteristics of Mixture of High-Level Wastes from Reprocessing PWR- UO_2 (2/3) and PWR-Pu(a) (1/3) Fuel (Power = 30 MW, burnup = 33,000 MWd/MTM charged to reactor)⁽⁵⁴⁾

Time Since Reprocessing,(b) Years	Power from Combined Waste from 1 MT of Heavy Metals, kW			Total Power Per Unit Volume of Solidified Waste, kW/m ³ (e)
	Fission Products ^(c)	Actinides ^(d)	Total	
0	19.8	5.04	24.8	487
3	2.94	1.54	4.48	88
10	0.96	1.16	2.50	49
100	0.099	0.076	0.176	3.5
1000	0.025	0.016	0.041	0.8

- a. PWR with self-sustaining Pu recycle.
- b. Fuel is reprocessed 150 days after its discharge from the reactor.
- c. Assumes that all of the tritium and noble gases and 99.9% of the halogen fission products in the spent fuel are separated from the high-level waste during reprocessing.
- d. Consists of all the actinides remaining after removal of 99.5% of the uranium and plutonium during reprocessing.
- e. Assumes 0.05 m³ of high-level solid waste per metric ton of heavy metal.

These provisions for retrievability will reduce the acceptable diameter of a waste canister for final storage. A typical canister used in many conceptual design studies for a salt mine demonstration waste repository is 30 m in diameter. No estimates have been made for repositories in other geologic formations.

Maximum heat--The acceptable maximum radioactive decay heat in a canister will depend upon several factors including the properties of the geologic formation, the diameter and length of the canister, and whether canister overpacking and/or a hole liner is used. An analysis⁽⁵⁶⁾ of a salt mine waste repository indicated that on the basis of the single consideration of acceptable average salt temperatures up to about 7 kW in 3-m-long canisters containing 10-year aged wastes (at burial) would be acceptable. Other considerations such as the temperature of salt immediately adjacent to the canister, the length of time the room must stay open, the effects of temperature on the canister material and other hardware, and limitations on the temperature of waste within the canister (Section 6) would tend to reduce the acceptable amount of heat in the canister to some extent.

Transferable surface contamination--Limits on acceptable amounts of surface contamination have not been established. They will probably be about the same as those shown in Table 24.16 for the amounts above which the contamination is considered significant.⁽⁵⁷⁾

TABLE 24.16. Significant Amounts of Surface Contamination⁽⁵⁷⁾

Contaminant	Maximum Permissible Level ^(a)	
	$\mu\text{Ci}/\text{cm}^2$	$\text{dis}/\text{min}/\text{cm}^2$
Natural or depleted uranium and natural thorium:		
beta-gamma	10^{-3}	2200
alpha	10^{-4}	220
All other beta-gamma emitting radionuclides	10^{-4}	220
All other alpha emitting radionuclides	10^{-5}	22

a. Averaged over any area of 300 cm^2 of any part of the package surface.

2. High-Gamma Transuranic Wastes

These wastes are comprised of cladding wastes and other alpha-beta-gamma solid wastes.⁽⁵⁴⁾ The specific activities will be much below those in high-level wastes. For example, it was estimated⁽⁵⁴⁾ that the induced radioactivities of the light elements within the cladding from LWRs will generate about 27 W/MTHM fuel at 10 years after discharge from the reactor. At the

same time the fission products in the solidified high-level wastes from the LWR will generate about 1000 W, Table 24.14. It was also estimated that the volumes of compacted wastes and of solidified high-level wastes from LWRs will be about the same.⁽⁵⁴⁾

High-gamma transuranic wastes will probably be encased in sealed cylindrical metal canisters. The design and acceptance criteria of these canisters have not been established. They will probably be similar to those for high-level wastes described above for high-level wastes.

3. Low-Gamma Transuranic Wastes

These wastes are defined as those solid materials that contain very low levels of gamma emitting nuclides so that they can be handled without supplementary shielding and which also contain long-lived alpha emitting nuclides in concentrations greater than 10 $\mu\text{Ci/kg}$.⁽⁵⁴⁾ Examples are expendable material such as absorbent tissues, clothing, gloves, plastic bags, glass, ceramic, solidified wastes, and equipment originating in plutonium processing and fabrication facilities.⁽⁵⁴⁾

Packaging and acceptance criteria for these wastes have not been firmly established.⁽⁵⁸⁾ It is probable that large volume packages will be employed, e.g., 55-gal drums, and that the quantities of transuranics will be limited to small amounts, e.g., about 0.45 W/package for weapons grade plutonium waste.⁽⁵⁸⁾ Also, the limits on contamination will be probably about the same as those listed in Table 24.16.

Terms and Their Interpretations

<u>Term</u>	<u>Interpretation</u>
Waste storage	Always intent to remove waste for other storage at some time in the future.
Interim storage	Same as above but the period of storage is assumed to be limited and relatively short.
Geologic final storage	Geologic storage concepts in which an initial period of retrievability exists.
Geologic disposal	Geologic disposal in which no retrievability is contemplated or provided for.
Pilot plant	Real radioactive waste would be handled as in a "geologic final storage" but only in limited quantities (all readily retrievable) and all aspects of the operations and design would be closely monitored; design confirmation information would be sought.
Demonstration repository	Means the same as "geologic final storage" as defined above.

24.2.2 Repository System Concepts

24.2.2.1 Required Features.

Any facility constructed by conventional room and pillar mining methods for the storage of all types of solid nuclear waste will have essentially the same features. (For details see Sections 4.1 and 4.2 of Reference 6.) A typical facility would include:

1. receiving facilities for both truck and rail shipments
2. surface facility for inspection, monitoring, and decontamination of incoming shipping casks
3. equipment for repair and overpacking of damaged waste canisters
4. waste hoisting equipment and shaft
5. underground receiving, transport, and storage equipment
6. underground retrieval and repackaging facility
7. surface holding area for shipment surges
8. men and materials hoisting equipment and shaft
9. ventilation and filter system
10. instrumentation and monitor system
11. weather station
12. emergency power system
13. emergency communication system
14. emergency medical facility
15. computerized storage record system
16. excavated material handling and disposal
17. support facilities:
 - administration
 - warehouse
 - communications
 - utilities

The specific design of items 1 through 6 are waste type-dependent, while items 7 through 17 are essentially independent of waste type. For this discussion the three categories of waste defined in Section 24.2.1, namely TRU, ILW and HLW, will be used.

24.2.2.2 Items Independent of Waste Type

The most cost effective surface holding area to handle surges in waste shipment and minor equipment breakdown is the shipping container or cask itself. Sufficient rail siding, fences, lighting, etc., will be provided to

allow a few days' waste shipments to backlog. The criterion used to design the hoisting equipment and the shaft for transporting men and materials is essentially the estimate of facility size and growth rate. The ventilation and filter system will always direct the airflow from normally uncontaminated areas to areas of progressively higher contamination. The instrumentation and monitoring system will monitor radiation levels, in the mine and on the surface, atmospheric contamination, radioactive and noxious gases in the mine, mine closure rates, etc. The weather station will cover the standard temperature, pressure, wind direction and velocity, and other conditions as required by licensing requirements. The weather data in conjunction with the instrumentation system will allow fallout, dose rate and dose predictions, should an accidental release occur.

Emergency power, emergency communications, and emergency medical facilities are necessary in the event of standard power failure, accident conditions, etc. Excavated material handling and disposal will have to consider the impact on the environment of the salt removed from the mine. Support facilities are those that any plant or mine would require, bookkeeping, payroll, spare parts, telephone, electricity, gas, roads, etc.

24.2.2.3 TRU-Specific Items

The facilities required for receiving, inspecting, monitoring, and decontaminating of incoming TRU waste are rather straightforward bulk handling, washdown facilities. Where appropriate, repair and overpacking will consist of larger drums and boxes and fiberglass patches. Waste hoisting equipment will differ little from the shaft for men and materials other than having the capability for easy decontamination. The receiving, transport, and storage underground will be a fork-truck-warehouse type operation, backfilling the stacked waste with salt. Retrieval would be a reversal of emplacement--removing the salt, inspecting overpacking if necessary, and hoisting to the surface for shipment elsewhere.

24.2.2.4 HLW-Specific Items

The facilities required for receiving HLW waste are much more complex than those for TRU waste. The waste canisters will arrive at the site in large shielded shipping casks which will be unloaded from the truck or railcar and moved into a monitor area. There the casks will be checked for internal temperature and contamination, cooled if necessary, and then moved into or adjacent to a hot cell and prepared for removing the waste canisters. By use of remotely operated equipment and manipulators, the shipping cask will be opened, the waste canisters removed, radiation and temperature monitored, inspected for shipping damage, repaired and decontaminated if necessary, all inside the biological shield hot cell. The waste canisters will next be lowered from the hot cell on the surface to the storage level, transferred to

a shielded transporter, transported through the mine, placed into a previously prepared cavity and backfilled for radiation shielding. Retrieval will have to be accomplished by specially designed equipment. Two methods under consideration for salt are 1) the use of sleeved holes to protect the waste can from direct contact with the salt and 2) overcoring the waste canister which was placed directly into the salt. In either case, after the waste is recovered, it will have to be returned to the underground hot cell, inspected for damage, monitored for contamination, overpacked if necessary, then hoisted to the surface and placed into a shipping container for transfer to another facility.

24.2.2.5 ILW-Specific Items

The ILW facilities are essentially the same as those for the HLW. The differences arise from the fact that ILW has a lower thermal power density and is packaged in a different--probably larger--container. The lower thermal power of the ILW will permit denser storage arrays; the different canister configuration may dictate the use of a special ILW transporter, which would be less sophisticated than the HLW transporter.

24.2.2.6 Typical Configurations

Figures 24.4 and 24.5 are two examples of how a waste repository might be configured. Figure 24.4 shows a single-level facility with the surface facilities somewhat separated. Figure 24.5 shows a bi-level storage with tightly grouped surface facilities. Certainly other combinations are possible, and several are being explored to attempt to achieve near optimum configuration, efficiency and cost.

24.2.3 Occupational Safety

The objective of any occupational safety program is to protect operations personnel, the general public, and the environment, as well as to minimize damage to, or the loss of, property from the hazards resulting from facility operations. Since all Geologic Waste Storage Facilities will be designed, constructed, and operated by/for ERDA, these facilities will conform with appropriate ERDA standards, guides, etc. Those areas of concern due to the active operation of a waste storage facility include operational and radiation safety plus chronic release of radioactive and other material to the environment and the accidental release of radioactivity.

24.2.3.1 Operational Safety

Geologic Waste Storage facilities combine the operational safety problems of the nuclear industry with those of the mining industry, and both industries have rather rigorous safety standards. For the nuclear industry, these are spelled out in ERDA Manual, Chapter 0550, "Operational Safety Standards" and the applicable standards referred therein. For the mining industry, the

FIGURE 24.4. Single-Level Terminal Storage

regulations, standards, criteria, etc., of the Mine Enforcement Safety Administration (MESA) apply. Other state and Federal regulations governing such items as motor vehicle and traffic safety, explosives safety, fire protection, etc., will also be factored into repository safety requirements. Areas of conflict between the various standards will usually be resolved, in favor of the highest safety level. In addition, these facilities will be licensed by the Nuclear Regulatory Commission, probably under Title 10, CFR 70, and the Safety Analyses Report, which is submitted as part of the license application, will present the entire operational safety program.

24.2.3.2 Radiation Safety

All waste handling operations are to be carried out in such a manner that personnel exposure from packaged waste or contamination are kept to the lowest levels that can be reasonably achieved. Radiation control and monitoring systems will conform to ERDA Manual, Chapter 0524, "Standards for Radiation Protection" and the applicable codes and standards referred therein. Note that

FIGURE 24.5. Bi-Level Terminal Storage

the operational standards will need to be fully compatible with the radiation requirements as an example, while the use of diesel powered equipment may be allowable under MESA rules. With the resulting large volume of particulate laden exhaust air, the need to meet radiation release requirements will affect the design, size, and operation of the exhaust air filter. In addition, the

Safety Analysis Report and the Environmental Impact Report which will be required by NRC as part of the licensing procedure will discuss the entire radiation safety program in detail.

24.2.3.3 Chronic Releases to the Atmosphere

Very small quantities of radioactive and other materials will be released to the atmosphere as a result of the normal operations of any repository. The concentrations of these materials in normal working areas and offsite will be limited to well below concentrations that are considered acceptable for occupational exposure and for exposure of the general public, respectively.

The materials that will become airborne in small concentrations include solid waste particles resulting from resuspension of surface contamination; gases (e.g., ^{85}Kr , ^3H , He) that result from decay of the waste; radon isotopes that occur naturally in any mine; nonradioactive gases (H_2 and HCl) that are formed by corrosion, electrolysis, radiolysis, and decomposition of any brine present; exhaust gases from the diesel-powered equipment and salt or other particles resulting from mining operations.

The effect of chronic releases of radioactivity from a repository is site dependent and an evaluation must be made for each location and design.

24.2.3.4 Upper Limit Releases from Operational Accidents

The types of accidents and the consequences of the release of radioactivity to the surroundings from upper limit accidents in the operation of a waste repository is an obvious function of the design and the location. However, because of the similarities in the design and operation of all repositories, the previously discussed design of the repository located at Lyons, Kansas, can be considered a representative case. In the safety analysis for the Lyons repository, certain highly improbable dispersive events that could result from operation of the repository were considered. This was done to provide some relative appreciation for the maximum consequences that could result from unplanned or unforeseen circumstances. It was planned that all significant quantities of radioactive materials would be handled as tightly sealed packages of solid materials within confinement zones of the repository. Thus the only significant mechanism for sudden and acute escape of radioactive materials from the repository must involve the rupture of a package and the passage of airborne particulates through filter systems that would be physically isolated from operating areas and tested routinely to provide protection against pressure surges, plugging, and deterioration.

The criteria being developed for waste forms in any future terminal storage facility stipulate that no combustible material be present unless it is fixed in a nonflammable matrix. Suppose that through some unpredictable circumstances or sabotage one of these containers would be ruptured and involved

in a fire. This condition was examined for the Alpha-Operations Facility, and it was postulated as having the potential of generating more radioactive aerosol than any other type of accident. The release to the atmosphere following such a fire would be maximized if this accident were to take place in the alpha waste receiving building, since a large fraction of the contaminated smoke would reach the filters.

Consequences of a fire in the alpha waste receiving building were estimated by assuming there is a fire of 30 min duration involving the contents of many packages of alpha waste. Because of the reaction of the sprinkler system this time period may be a conservative overestimate for complete combustion of the contents of a package. The resulting fire was assumed to rupture the package and spread to other packages within the waste package and to any adjacent waste packages.

Based upon reasonable assumptions, the estimated release of plutonium from the stack of the Alpha-Operations Facility would be about 0.005 g. The maximum quantity of plutonium inhaled by a hypothetical person standing at the site boundary during the entire period of release would be about 0.01 μg of Pu, an intake that corresponds to 20% of the maximum permissible body burden.

The upper-limit accident in the High-Level Facility is assumed to result in complete dispersal of the contents of a package of high-level waste in the air of the high-level waste transfer cell. A maximum release of 0.009 Ci of mixed fission products to the atmosphere through the stack was estimated. The estimated maximum quantity of radioactive material that could be inhaled by a hypothetical person standing at the site boundary was 0.02 μCi , an intake by inhalation that corresponds to 1% of the maximum acceptable body burden for occupational exposure.

24.2.4 Repository Performance Factors

24.2.4.1 Structural Stability

Regardless of the geologic formation involved or the type of waste being placed in final storage, the basic mine design will be the same; that is, a system of rooms and rib-pillars connected by corridors and main entries. The first objective of the underground mine design is to assure that the openings will be stable throughout both the operating and retrieval periods. For most of the rock types, the detailed design (primarily the size of the rooms and pillars) is a straightforward process involving consideration of the rock strength, overburden pressure (depth), residual tectonic stresses, width-height ratio of the pillars, and extraction ratio. In these cases, optimization will be performed on a combination of the cost of extraction and efficiency of space utilization.

For some rock types, the detailed design of the mine openings becomes more elaborate because of their peculiar properties. For example, salt is one of the most plastic of rocks. For waste disposal, this property is a distinct advantage (see below), but it does complicate the mine design, especially for retrievability where access to the waste must be maintained for extended periods. Another rock type posing peculiar problems in mine design is shale. Most shales are very sensitive to changes in moisture content and, when exposed to mine ventilation air, either dehydrate, causing a complete loss of integrity (unravelling), or absorb moisture, with resultant loss of cohesion. For waste repositories in shale, it may be necessary to control the mine atmosphere or perhaps to seal the rock by applying shotcrete or other surface membrane.

24.2.4.2 Long-Term Containment

Since the intent of geologic storage (i.e., with retrievability) is to convert the facility to disposal configuration at some undefined future time, the second objective of the mine design is to permit an easy progression to geologic confinement of the waste. Geologic confinement, in this context, means that over the very long periods involved, the various engineered barriers cannot be depended upon to maintain their integrity and that the continued isolation of the wastes is provided by the surrounding geologic materials. This long-term containment and isolation is most easily visualized for salt. In this case, the plasticity of the formation permits a gradual closure of excavated openings, with the compaction, reconsolidation, and recrystallization of the crushed salt backfill in the rooms. When this process is completed, the wastes are contained in a massive and solid salt formation.

For salt, the design problem becomes one of designing the mine openings to accomplish that complete closure at the desired time. For other rocks which are not as plastic, the long-term closure will take place by mechanisms other than plastic deformation. In this case, the method of backfilling the rooms becomes much more critical, and special analysis will be made to assure that any rock fracturing around and over the openings does not breach the formation's long-term integrity.

24.2.4.3 Temperature Limitations and the Influence of Physical and Thermal Properties

When nuclear waste is buried in geologic formations, the heat released increases the temperature of the formation, with subsequent changes in physical and chemical properties that could lead to abnormal behavior. Ultimately the heat will be transferred to the atmosphere, and the formation and surroundings will return to essentially their initial state, provided that the repository design precludes excessive temperatures and temperature gradients that would adversely affect the operational safety, containment integrity, and the environment. In the case of high-level waste, the return to the

initial state will require thousands of years. This dictates the need for long-term analysis of the formation behavior, a thorough thermal analysis, and a subsequent determination of any thermal effects.

Thermal effects that influence the allowable temperature rises and heat release rates include: 1) thermal stability of the waste; 2) thermal stability of the formation; 3) migration of water contained in the pores or small cavities in the formation; 4) structural integrity of the entire formation; 5) temperature rise in any nearby fresh water aquifers; 6) heating of the earth's surface; and 7) temperature increases beyond the boundaries of the disposal area.

Establishing thermal criteria that limit adverse thermal effects is a function of the particular formation and is not a straightforward and simple procedure. Thermal criteria in the form of maximum permissible temperatures were developed by Cheverton and Turner⁽⁵⁶⁾ for bedded salt and applied to a proposed repository to be located at Lyons, Kansas.

In their analysis, the temperature of the solidified high-level waste was limited to the maximum value that existed during the waste solidification process, thus minimizing further volatilization of the waste. This reduces the potential for migration of the waste into working areas during operation of the repository. The maximum permissible temperature at the centerline of a waste canister then becomes limited to the range of 650°C to 1100°C, with the specific value depending on the solidification process.

All rock mechanics-oriented requirements involving mine stability and long-term integrity of the containment appear to have been satisfied when the following restrictions were placed on the amount of salt near the burial horizon that exceeds specified temperatures. Between horizontal planes through the ends of the waste packages and within the radial boundaries of a unit cell associated with each waste package, 1% or less of the salt is allowed to experience a temperature rise in excess of 210°C, and 25% or less is allowed a rise in excess of 160°C. These temperature rises will correspond to limiting temperatures of 200°C and 250°C, respectively, for the proposed depths where the ambient temperature is about 22°C.

To prevent possible adverse effects on the environment, temperature increases were limited to 0.6°C at the earth's surface, 28°C in fresh water aquifers above the repository (assumed to be stagnant), and 0.6°C in geologic formations at the outer edge of the repository buffer zone. Because of the very low heat fluxes [maximum of about 0.2 Btu/(hr ft²) or 0.7 W/m²] in the vicinity of the aquifers, only a very small flow rate in an aquifer is required to limit the temperature rise well below that achieved in a stagnant aquifer.

Figure 3.1⁽⁵⁶⁾ shows the application of these thermal criteria by Cheverton and Turner to disposal in salt burial in rows in excavated rooms. Based on the 25% salt criterion discussed above, the gross loading surface density (which is proportional to the thermal power density based on the gross dimensions of the disposal area) is nearly independent of pitch; based on the 1% criterion, it decreases substantially with increasing pitch. These relative trends were generally valid for all cases considered in the parametric studies regardless of room size or number of rows per room.

The significant trends found by Cheverton and Turner in their studies, which included other waste ages and room sizes (including more than one row per room), can be summarized as follows:

1. The younger the waste, the smaller the pitch for which the 1% salt and the waste criteria become limiting.
2. For a particular waste age, the difference between maximum permissible gross surface densities for room widths between 4.6 m and 15 m is small. However, there are significant differences between net surface densities. For instance, in the case of a 10-year-old waste, the maximum permissible net surface density for a 4.6-m room is about 37% greater than that for the other two room sizes. This constitutes a substantial saving in mining and salt disposal costs for the 4.6-m room.
3. Increasing the pitch permits more waste per package, except where limited by the waste temperature.
4. For any reasonable emplacement geometry, the maximum heat load is about 150 kW/acre (37 W/m^2).

Except for the limitation on the maximum waste temperature, the previously discussed thermal criteria and all of these conclusions cannot be applied to other geologic formations whose thermal characteristics are quite different. Development of thermal criteria for other rock types will require scouting calculations, both thermal and rock mechanics types, as well as experimental work.

The high-gamma transuranic waste and low-level transuranic waste present much less a thermal problem than high-level wastes, particularly the low-level type. The cladding waste or fuel hulls from spent reactor fuel elements (which is intermediate with respect to heat generation) when compacted into waste canisters will produce surface heat fluxes that are about a factor of 100 less than that for high-level waste. Cost considerations, however, dictate large canisters with as close packing as possible in a disposal horizon. A recent study⁽⁵⁷⁾ for a pilot plant for the disposal of fuel hulls in salt was made with the objective of relating the maximum floor temperatures in a room and the maximum salt temperatures to parameters such as age of waste, canister

spacing or pitch, burial depth, and vertical stacking of the two canisters in one burial hole. These results were then used to relate the maximum floor temperature to the required excavation of salt per canister, which can in turn be used in estimating excavation costs for selected maximum floor temperatures.

The temperatures during storage have not been firmly established for particular repositories, burial times, heating rates, etc. A calculation of Cheverton and Turner⁽⁵⁶⁾ for the following particular system in a salt mine repository indicated a maximum temperature rise in adjacent solid salt of 20°C at 1000 years and a maximum temperature difference between the solid salt and the waste of about 10°C.

1. Package and radiation heating--55-gal (0.21 m^3) drum with 0.30 W/drum uniformly distributed within drums at burial.
2. Stacking and backfill--3.7-m-high room with 4 drums stacked one upon the other from floor to ceiling. Stacks are close packed, and remaining spaces are backfilled with crushed salt.
3. Thermal conductivity-- $0.1 \text{ Btu hr}^{-1} \text{ ft}^{-1} \text{ }^{\circ}\text{F}^{-1}$ ($0.2 \text{ W/m}^{\circ}\text{C}$) for waste and backfill.

It is obvious that handling and disposal of these wastes will be a lesser problem than for high-level wastes. Some geologic formations unsuited for high-level waste could be quite suitable for these wastes with lower heat generation rates.

The most important single variable affecting the temperature distribution is the thermal conductivity of the geologic medium. The heat capacity and the density have little importance at times when limiting temperatures are attained since a near steady-state condition obtains (at steady state, the temperature distribution is independent of density and heat capacity). Generally, the thermal conductivity of a salt formation is about a factor of two higher than for geologic formations composed of other materials. Hence, as a rough rule of thumb, twice the temperature rise can be expected in other geologic formations for the same power output by the waste.

The pore water content of rocks has a strong influence on the thermal conductivity and can create problems due to thermal effects such as hydrofracture and enhanced water migration.

Generally, the thermal conductivity decreases with porosity and the pores are usually occupied by brine at the depths under consideration for a repository. If the gas or brine is driven out of the pores by heating of the formation, the thermal conductivity will be much lower (reduction by factors of two to three is possible). This could make the rock unacceptable as a disposal medium for high-level waste.

The behavior of pore water when a geologic formation is heated is not completely understood. In the case of salt, the water occurs as brine occlusions (about 1/2%), and it has been demonstrated that the brine migrates up the thermal gradient by a mechanism that involves solutioning on the hot side and precipitating on the cold side.^(60,61) The important point is that the small amount of brine present and its migration under a thermal gradient has little effect on the thermal conductivity. In nonsalt media, the effect of pore water could be important, particularly in argillaceous geologic materials such as shales, clays, mudstones, etc., and to a lesser extent the limestones and dolomites. The crystalline rocks, such as granite and basalt, will probably contain too little pore water to significantly change the thermal conductivity. However, existing fractures and cracking under thermal gradients will cause a reduction of the effective conductivity.

Project Salt Vault⁽⁶¹⁾ demonstrated that large amounts of water collected at the heaters when they were emplaced in the shale at Lyons, Kansas. This apparent migration up the thermal gradient cannot be explained on the same basis as that for salt. It could be that fractures developed due to the heating and the pore pressure gradient caused the inflow of water. As pointed out previously, a large decrease in the water content of argillaceous materials will cause a significant decrease in the thermal conductivity.

It is clear that water content and migration effects on the thermal conductivity of argillaceous materials must be determined for a proper evaluation as a disposal medium. Some laboratory studies will be useful, but in-situ experiments are a necessity.

24.2.4.4 Thermal Effects on Repository Operations

The effects of the heat released from nuclear waste on repository operations can be divided into two classes. These are short-term effects on the active emplacement operations, and longer-term ones which mainly affect retrievability. Most of the short-term thermal concerns are associated with high-level waste with its high heat generation rate of about 30 kW/m^3 for 10-year-old waste.

High-level waste canisters are designed so that the maximum centerline temperature of a waste canister standing in air will not exceed the incipient melting temperature of waste, typically between 600 and 1100°C. Normally the time cycle involved in removing a canister from a shipping cask, transporting it to the burial location, and emplacing it in a rock formation will be short enough to preclude the canister's temperatures from exceeding design limits.

However, all canister transporters are to be provided with a cooling capability in case of unforeseen delays. Sufficient ventilation of air will be provided to remove heat emitted by the canisters in the corridors, burial rooms, and in any temporary storage rooms. It will also be necessary to provide biological shielding in handling and transporting high-level waste and high-gamma transuranic waste.

The long-term thermal and radiation effects associated with the operation of a repository result chiefly from the retrievability criterion. After a burial room is filled with high-level waste (or high-gamma transuranic waste), the room could conceivably be sealed (but not backfilled) and no longer will be cooled by the ventilating air. The temperature of the room will rise and peak in 10 to 50 years after closure. Although this case has not been analyzed thermally, it can be expected that the behavior will be similar to that for the case of the crushed salt backfilled room with a layer of air over the top of the salt as shown in Figure 3.3.⁽⁵⁶⁾ Based on these results, a temperature rise of 90 to 148°C in the floor and wall temperatures can be anticipated for burial times greater than 5 years or so.

Retrieving waste after such burial times could require unsealing the rooms and providing cooling air to reduce the temperature to working levels. Obviously, the major problem will arise in removing a high-temperature radiation shield from the top of the hole and moving it to a safe storage area, then removing the very hot (up to 1100°C) canister from the hole, transporting it to the shaft, hoisting it out of the repository and placing it in a shipping cask for transfer to another storage facility or reprocessing plant.

These problems are similar to those associated with the burial of the canister except that the canister may have less structural integrity and require more careful handling. The retrievability concerns are currently being examined to determine the need (if any) of any required special equipment and handling techniques.

24.2.4.5 Storage of Radiation Energy in Solidified Wastes

Stored radiation energy refers to that energy associated with lattice damage (displaced atoms) in a solid caused by radiation and released as heat when the damage is annealed (i.e., when the undamaged structure is restored). Most types of radiation damage persisting at the irradiation temperature are annealed when the temperature is increased sufficiently. Damage resulting in chemical decomposition with loss of a decomposition product from the solid is, of course, not reversed by heating the remaining solid at elevated temperatures.

High-Level Wastes. The main concerns with stored energy in solidified high-level wastes are 1) that its rapid release may cause temperature rises which could significantly affect the physical and chemical properties of the

solid wastes and 2) that the explosive release of mechanical energy accompanying a sudden temperature rise in the wastes might be sufficient to damage the container's integrity.

The principal nuclear radiations released within high-level wastes are alpha-RNs (recoiling product-nuclei resulting from alpha disintegration), alphas, betas, and gammas.⁽⁶⁴⁾ There are also relatively small numbers of fissions and neutrons formed within the wastes.⁽⁶⁵⁾ Radiation damage resulting in stored energy is expected to be produced mainly by collision processes between alpha-RNs and the atoms within the waste solids.

The buildup and release behavior of radiation-induced stored energy has been investigated⁽⁶⁵⁾ for several synthetic, solidified high-level waste forms. The objectives have been to evaluate the possibilities of sudden release of the stored energy and the hazards in the handling and containment of the waste which might result from a sudden energy release. The waste forms studied included two borosilicate glass formulations, calcine, calcine on Al_2O_3 , and a hot press compact of 50% waste oxide-50% Quartz. Fused silica and Al_2O_3 , without waste oxides, were also investigated. The materials were irradiated either by internal alpha radiation (by incorporating ^{244}Cm in the solid) or by neutron irradiation in the Oak Ridge Research Reactor.

The results of this study showed⁽⁶⁵⁾ that the amounts of stored energy which may accumulate within about 10 years at relatively low storage temperatures following reprocessing are such that only moderate temperature increases ($<200^\circ\text{C}$) in the wastes would occur in the event of sudden release of the stored energy. Extrapolation of the results to longer storage times in a geologic repository together with available theoretical and experimental information indicated that saturation of energy storage would occur at 50 cal/g or less for each waste type; the temperature rise corresponding to 50 cal/g would be about 250°C for most of the waste types.

It was estimated that sudden but transient increases in waste temperatures of up to 200°C following low-temperature storage or of up to 250°C during long-term repository storage would have no serious adverse effects on the waste container or on the physical and chemical properties of the wastes. The maximum linear expansion of wastes which would result from a temperature rise of 250°C would be about 0.1%. The steel walls of the waste canister would easily accommodate the stress from this expansion on the diameter and volume of the wastes. Thus, in the unlikely event that stored energy is released suddenly for some unforeseen reason, no serious adverse effects are expected.

High-Gamma Transuranic Wastes. These wastes are comprised of cladding wastes and other alpha-beta-gamma solid wastes. The specific activities will be much below those in high-level wastes. For example, it was estimated that for light water reactors the volumes of cladding wastes and of solidified

high-level wastes will be about the same.⁽⁶⁶⁾ However, the amounts of actinides and fission products associated with the cladding will be less than 0.1% of those in the high-level wastes. It was also estimated⁽⁶⁶⁾ that the induced radioactivities of the light elements within compacted LWR wastes will generate about 0.5 kW of heat per m^3 at 10 years after discharge from the reactor. As previously indicated, the fission product activities within 10-year-aged high-level wastes will generate about 100 times this much decay heat.

The cladding wastes are metallic; it is expected that negligible amounts of stored energy will accumulate in these. Other intermediate-level wastes which are in the form of oxides will accumulate stored energy at rates proportional to the rates of formation of α -RNs within the wastes. Accordingly, the rates of formation of stored energy in these oxide wastes will be less than 0.1% of the rates of accumulation with high-level wastes. It was concluded that stored energy in high-gamma transuranic wastes will be of negligible concern.

Low-Gamma Transuranic Wastes. These wastes, also called alpha-wastes, are defined as those solid materials that contain very low levels of gamma-emitting nuclides so that they can be handled without supplementary shielding. They also contain long-lived alpha-emitting nuclides in concentrations greater than 10 $\mu\text{Ci/kg}$.⁽⁵⁴⁾ Examples are expendable material such as absorbent tissues, clothing, gloves, plastic bags, glass, ceramic, solidified wastes, and equipment originating in plutonium processing and fabrication facilities.^(54,66)

The glasses and other metal oxides in these wastes which are subjected to alpha and alpha-RN irradiation will accumulate some stored energy. However, the rate of accumulation per package will be much lower than that in high-level waste. For example, for the situation in which the alpha emitters are uniformly distributed throughout 0.21 m^3 of borosilicate glass it can be estimated that the alpha decays occurring during 200 watt-years would produce about 4 cal/g of stored energy in the glass. This is an innocuous concentration of stored energy.

However, additional consideration should be given the situation in which all the alpha emitters in a package are concentrated in a few grams of material and thus generate a relatively high dose rate and dose within those few grams.

24.2.4.6 Storage of Radiation Energy in Rock Adjacent to Buried Wastes

The rock just adjacent to a waste canister in a geologic repository will be exposed to gamma rays emanating from the wastes. There will also be a few neutrons. Information on accumulation and annealing of radiation damage (storage and release of radiation energy) in the gamma-irradiated rock is pertinent for reasons analogous to those for high-level wastes stated above.

The gamma ray dose in the rock next to a canister will depend on the type of fission product activity, age of the waste within the canister, canister material and thickness, and the proximity of the rock to the canister. For canister diameters greater than about 15 cm there is only a slight dependence upon the volume of waste placed within the rock cavity.⁽⁶⁷⁾ The maximum specific fission product activity in solidified high-level wastes is expected to occur in wastes from light water reactor fuels and from high-temperature gas reactor fuels.⁽⁵⁴⁾ The expected maximum for light water reactor fuels will be that which corresponds to about 17 to 35 kW of fission product decay heat per m^3 in 10-year-aged wastes. This is equivalent to about 350 to 700 kW per m^3 in the waste immediately following reprocessing, assumed to be 150 days after fuel is removed from the reactor.

For 10-year-aged light water reactor wastes in which the specific fission product activity is 28 kW/m^3 (corresponding to 5.0 kW in 0.3-m ID, 2.4-m high canisters), the maximum gamma ray dose in rock at the canister surface will be about 1.8×10^{10} rad at 100 years after burial of the 10-year-aged wastes, about 2.0×10^{10} rad at 1000 years after burial, and about 3.0×10^{10} rad at 10 million years.⁽⁶⁷⁾ The gamma dose rate at burial will be 0.06 to 0.07 megarad/hr. The dose and dose rate decrease exponentially with increasing distance from the surface of a waste canister; with a 0.3-m-diam canister, for example, the dose at 0.15 m from the surface is 1/10 that at the surface.⁽⁶⁷⁾

The contribution of neutrons to the dose in the rock will be negligible.⁽⁶⁷⁾

The gamma ray intensity adjacent to canisters containing high-gamma transuranic wastes will be less than 10% of the maximum around canisters of high-level wastes. That adjacent to canisters containing low-gamma transuranic wastes will be negligible.

Halite. Gamma irradiation of NaCl crystals produces lattice damage (stored energy) in the crystals.⁽⁶⁷⁾ Under certain conditions of temperature, dose rate, dose, etc., the stored energy accumulated is appreciable. Experimental studies were conducted to ascertain both the amounts of gamma ray energy stored in salt under exposure conditions which will prevail in a salt mine repository and the release characteristics of this stored energy.

The experimental data and correlations showed that the maximum efficiency of formation of stored energy in salt in a waste repository will be $1.4 \times 10^{-9} \text{ cal g}^{-1} \text{ rad}^{-1}$. It will be independent of temperature; dose rate (at $<10^8 \text{ rad/hr}$) dose, and type of salt (dome, bedded, strained, etc.). Then the maximum amount of stored energy which would accumulate in the salt next to a waste canister is given by the product of the maximum formation rate, $1.4 \times 10^{-9} \text{ cal g}^{-1} \text{ rad}^{-1}$, and the dose in the salt. As stated above, the maximum gamma dose

in salt immediately adjacent to a waste canister would be 1.8×10^{10} rad, 2.0×10^{10} rad, and 3.0×10^{10} rad at 90, 1000, 10^7 years after burial. The corresponding maximum values of stored energy are 24, 28, and 41 cal/g.

Some annealing of the stored energy will occur during exposure in the repository, depending upon temperature and other factors. At 150°C and above, thermally activated annealing will be the dominant process and will limit accumulation of stored energy to negligible amounts. The rates of this thermally activated annealing increase with temperature but are independent of the amounts of stored energy in the irradiated salt. Illustrative values for the maximum annealing rates indicated by the experimental data are 9×10^{-4} , 0.12, and $34 \text{ cal g}^{-1} \text{ hr}^{-1}$ at 150, 200, and 275°C , respectively.

The rates of annealing by thermally activated processes or others at temperatures $<150^\circ\text{C}$ in a repository could not be predicted reliably from the results of the experimental studies. For assessing consequences of stored energy accumulations in the salt, it was assumed that the maximum amounts of stored energy at temperature $<150^\circ\text{C}$ would be those calculated from the maximum formation rates and the maximum gamma doses (see above).

Salt which is located within a distance of 0.15 to 0.18 m of a waste canister and receives significant gamma ray doses will normally have maximum temperatures greater than 200°C . The maxima will occur at 10 to 20 years following burial of 10-year-aged wastes. The temperatures will exceed 150°C at 1 to 2 years following burial and will remain above 150°C for more than 50 years. Accordingly, energy storage in salt in the normal burial area will be negligible for more than 50 years following burial of 10-year-aged wastes. Energy storage after the salt has cooled to $<150^\circ\text{C}$ will presumably accrue with the efficiency, $1.4 \times 10^{-9} \text{ cal g}^{-1} \text{ rad}^{-1}$.

Some canisters on the periphery of the repository will have salt temperatures lower than those within the normal burial area. This is due to their semi-isolation from thermal effects of other canisters. A rare canister may be isolated from the thermal effects of other canisters, and, accordingly, have still lower temperatures. Due to lack of annealing, significant concentrations of stored energy might accumulate in the salt around these canisters at times less than 50 years following burial.

As implied above, the main concern with stored energy in salt around a waste canister is that rapid release of stored energy could create excessive temperatures and/or release large amounts of mechanical energy. No conditions are apparent that would promote a sudden release of stored energy in repository salt (or in the contained waste) other than rapid heating of the salt (or waste) to a ΔT of a few hundred degrees by a source other than the contained stored energy. No such source of external heat is apparent. However,

assuming a sudden, unforeseen release of the entire amount of stored energy in the waste within a canister and within the surrounding salt, it was estimated that no serious adverse effects on waste containment and safe operation of the repository would result.⁽⁶⁷⁾

Finally, as implied above, H_2 is generated upon aqueous dissolution of irradiated salt.⁽⁶⁸⁾ The experimental results showed that the amount of H_2 is 5.2×10^{-6} mole/cal. At 40 cal/g the salt that dissolves to saturation in 1 ml of water would generate and release 1.7 cm^3 of H_2 . It was also found that the irradiated salt may dissolve much more rapidly than unirradiated salt. Unless the integrity of the repository containment is breached, fresh water will not enter the spaces around a waste can and there will be no occurrence of H_2 -evolution.

Silica Minerals, Feldspars, Clay Minerals, Limestones. A literature search was made⁽⁶⁹⁾ to compile data on the effects of radiations on geologic formations other than halite. No reported direct experimental measurements of energy storage in either silicate minerals or limestone under gamma irradiation were found.

However, a review of the theory of radiation damage and energy storage in solids and of pertinent physical and chemical properties of these rocks indicated no apparent reasons to expect any significant amounts of stored radiation energy in any important rock-forming minerals at the maximum gamma dose and fast neutron fluences which will prevail in rock immediately next to buried waste.

24.2.4.7 Radiochemical Effects in Solidified Wastes and in Rock Adjacent to Buried Wastes

Nuclear radiations bring about decomposition of water and some other liquids.^(60,69) They also decompose some crystalline solids, such as nitrates and chlorates, and some organic solids, and they can affect the chemical reactivity between solids and fluids.^(60,69) In general, considerations are given to the possibilities of such effects in each waste form and disposal system under investigation and to providing for their consequences should they occur.

Solidified Wastes. Radiation and thermal effects on the stability and chemical reactivity of high- and low-gamma transuranic waste forms are under investigation.⁽⁷⁰⁻⁸¹⁾

Halite. Salt deposits of the type which may be used for radioactive waste disposal may contain a small volume-fraction of brine inclusions (up to about 0.5%). Some brine from these inclusions will enter the open and unconfined spaces around a waste canister as a result of the migration of the inclusions. This migration takes place in the presence of thermal gradients in the salt and as a result of salt fracture, which occurs at high temperatures (starting at about 250°C). The brine, vapors, and salts around a can will be

exposed to high gamma-ray intensities and doses and to high temperatures. Accordingly, they will be subject to radiolysis and other changes in composition brought about by thermally induced reactions.

Detailed review and analyses⁽⁶⁰⁾ of information on the rates of inflow of brine and on the radiation and thermal chemistries of the salt and brines near a buried waste can show that the radiolysis products will include, importantly, H_2 , O_2 , and possibly ClO_3^- and BrO_3^- . Most of the ClO_3^- and BrO_3^- will decompose to halides and O_2 at the high temperatures around a can; $Mg(BrO_3)_2$, if present, may give rise to some Br_2 . Nearly all of this H_2 , and the accompanying oxidized species, are formed within the migrating brine inclusions by the radiations absorbed within the brine and by dissolution of the trapped electrons and holes from the irradiated solid salt. Some brine is rich in $MgCl_2$ (about 2.3 molar), and hydrolysis of the $MgCl_2$ around a canister will give rise to HCl . Corrosion reactions between the metal canister and water vapor are possibly large sources of H_2 .

It is believed⁽⁶⁰⁾ that none of these effects will jeopardize use of dome or bedded salt as a high-level waste repository. However, firm information on the identities and amounts of radiolytic and thermal reaction products around a waste canister are needed to help establish suitable burial and operating procedures. In-situ experiments in salt deposits are planned which will help furnish this information.

Silica Minerals, Feldspars, Clay Minerals, Limestones. A literature review⁽⁶⁹⁾ of pertinent experimental and theoretical information showed no apparent reasons to expect any significant radiation-induced chemical effects in any of these minerals. But the information is so sparse that the possibility of significant effects due to radiation cannot be completely ruled out. Thus certain radiation experiments are planned to verify the absence of significant effects as time and rock samples become available.

24.2.4.8 Chemical Effects

Corrosion. The metal canisters, packages, and sleeves which may be employed in geologic final storage will be subject to oxidation and/or corrosion in amounts dependent upon the metal (or metal alloy) used, the compositions of the gases and/or liquids in contact with the metal, and temperature. The waste canisters or packages may also be subject to corrosion on the inner surfaces by the contained wastes and/or by products of decomposition of the wastes. Experimental work is under way or planned to establish the corrosion characteristics of the metal which may be used with given waste types and with given external environments in geologic final storage.^(70-74, 76-83)

Brief comments on materials and exposure environments are given below:

High-Level Wastes. These wastes will be sealed within cylindrical metal canisters. The canister metal will probably be 304L stainless steel although

other candidate materials are being investigated (see Section 24.2.1). During the retrieval period of geologic final storage, these canisters may be located within protective liners or sleeves and thus isolated from contact with the rock and with any liquids in the rock. The canister will thus be in contact only with air during this period, and the air can be maintained dry and free of acid vapors if this appears worthwhile. Sleeve material will probably be carbon steel, but other materials are being considered. This maximum surface temperatures of canisters and sleeves may exceed 300°C and 250°C, respectively, in a salt mine waste repository.^(56,84) Those in other geologic formations have not been established.

In-situ tests within geologic formations are among the planned experiments. These are needed because the exact exposure conditions (amounts and compositions of fluids and distributions of fluids and temperatures) which will prevail in a geologic repository cannot be predicted and/or duplicated by other means.

Low-Gamma Transuranic Wastes. The currently used packages for these wastes are either 55-gal drums or fiberglass-coated wooden boxes. The drums and boxes are protected inside by polyethylene liners. The outer surfaces are protected by coatings of corrosion inhibitors or corrosion-resistant materials as required to ensure corrosion protection during handling, transportation, and retrievable storage. The surface temperatures of the packages during storage have not been firmly established for particular repositories, burial times, heating rates, etc. A calculation of Cheverton and Turner⁽⁸⁴⁾ for the following particular system in a salt mine repository indicated a maximum temperature rise in adjacent solid salt of 20°C at 1000 years, and a maximum ΔT between the solid salt and the waste of about 10°C.

1. Package and radiation heating--55-gal drums with 0.30 W/drum uniformly distributed within drums at burial.
2. Stacking and backfill--4-m high room with 4 drums stacked one upon the other from floor to ceiling. Stacks are close packed, and remaining spaces are backfilled with crushed salt.
3. Thermal conductivity-- $0.1 \text{ W}\cdot\text{m}^{-1}\cdot^\circ\text{C}^{-1}$ for waste and backfill.

Rock-Waste Interactions. Evaluating the feasibility of any waste disposal system will include a review of available information, and performance of additional experimental work as required, to insure that rock-waste interactions will not compromise the integrity of the repository during its operating lifetime. Some subjects in this category are listed below.

Most of the ongoing experimental work and the available information on each subject was mentioned and referenced in Section 24.2.3.

1. Chemical interactions between waste and rock.
2. Effects of radiation on physical and chemical properties of the rock.
3. Effects of high temperature on both waste and rock.
4. Methods of effectively isolating waste packages from rock and from circulating groundwater.
5. Migration of radionuclides from the waste through the formation.

24.2.5 Loss of Geologic Containment

Included in this section is a brief discussion of the effect of the maximum credible accident, that is, the loss of geologic containment by whatever means.

The potential causes of containment failure for a nuclear waste repository contained in geologic formations can be divided into two classes: anthropogenic and natural.

Potential anthropogenic causes of serious breaching of the containment of a nuclear waste repository include sabotage, nuclear warfare, and drilling.

Some natural events, capable of breaching the salt formation and causing dispersal of radionuclides to the environment are sudden and catastrophic, while others are slow and gradual. It is possible to combine the sudden and the slow processes in various hypothetical mechanisms of containment failure. Groundwater is the best candidate for removing radioactive material from the disposal area. Mobilization of such material by groundwater is usually a slow process controlled by the dissolution rate of the salt and the leaching rate of radionuclides. However, the water could gain access to the disposal horizon as a consequence of a sudden event such as faulting or the impact of a giant meteorite.

The most complete study of geologic containment failure mechanisms and their consequences was made for a waste repository in bedded salt of the Delaware Basin in Southeast New Mexico. (85)

The main conclusion of that study was that a serious breach of containment of a waste repository, either by natural events or human action, is an extremely remote possibility that is a much smaller risk than many others acceptable to society and of such small magnitude to be beyond the limit of human experience. Once the waste has been placed in such a configuration and the mine sealed, only the most extreme of natural events have any potential for release of radioactivity from the disposal zone. In addition, a sealed repository would be sabotage-proof if there were no armed intervention and temporary occupation of the area. Even the surface burst of a 50-MT nuclear weapon could not breach the containment.

Exposure of the waste horizon to the action of groundwater by a large fault with a vertical displacement of at least 350 m is highly improbable but cannot be proved impossible. However, in the event that New Mexico were to become affected by intense diastrophism, at least a few hundred thousand years would be required for an offset of 350 m to take place. A more sudden breach of containment could be caused by a large fault that would cut through the disposal zone and hydraulically connect the over- and underlying aquifers. Any water flow along the fracture zone would be downward into the deep aquifers. The geologic evidence indicates that such a fracture could eventually close by plastic flow of the salt beds. In the event that the deep aquifers became contaminated, it is extremely unlikely that activity could be brought to the surface except by drilling in or near the buffer zone since the normal flow velocities in these aquifers are a few kilometers per million years.

In view of the relative potentiometric heads, it is difficult to visualize circumstances that could cause an upward flow and contamination of the fresh water aquifer above the disposal horizon. The only possibility for the release of activity to this aquifer seems to be on the basis of the previously mentioned model that permits faulting and progressive displacement of the disposal horizon until waste is raised and brought into contact with circulating groundwater. Contamination of surface aquifers due to this mechanism of releasing radioactive material could be relatively serious, despite the very long time necessary to produce such a displacement. Based on the tectonic activity of the Delaware Basin in the last 200 million years, the probability of faulting through the repository has been estimated at 4×10^{-11} /year;⁽⁸⁵⁾ however, the probability that faulting would cause failure of waste containment is only a small but unknown fraction of this value.

The mechanism of containment failure with the most serious potential consequences seems to be the impact of a giant meteorite that produces a crater extending to the disposal horizon. The probability of such a catastrophic meteorite impact was estimated at 1.6×10^{-13} /year. The calculated exposure from the resulting global fallout would be on the same order of magnitude as that from the past nuclear bomb tests, provided ^{90}Sr had decayed for at least a few hundred years. The ejecta falling back into the nearby area would cause a serious and protracted contamination problem.

The evaluations and conclusions presented in that report are subject to uncertainty due to the scarcity of data on the behavior of thick salt formations if extensive faulting occurred and on the hydrologic properties of the various aquifers present in the area of the repository. However, it does demonstrate that the risks involved fall into the category of least likely occurrences which affect society. Most of the conclusions will apply to other geologic formations, but similar studies are required for each candidate rock and site for a waste repository.

The possible intrusion of man by drilling at some future time does require further discussion. The first point here is that a waste repository area will be dedicated to that purpose alone, with remote monitoring operations in progress and all possible intrusions by man controlled for as long as our present civilization endures. The problem arises with the possibility of future civilizations to which all records and knowledge of the repository have been lost.

The next point is that one criterion for selecting a suitable disposal formation is that it have no intrinsic value and be so ubiquitous and abundant that the chance of its becoming valuable to future civilizations is remote. Furthermore, the site of a repository will be so selected that there are no known or potentially valuable natural resources associated (especially underlying) with it, further reducing the chances of any future drilling.

Despite these precautions, there is still the possibility that future generations may wish unknowingly to drill at a repository site. If a TRU waste repository is drilled into, the chances are still better than 50-50 that the drill will miss the actual waste and instead penetrate one of the support pillars or unfilled entry corridors. Even if the drill should hit waste, the specific activity is so low, and the volume of material is so limited which can be brought to the surface through the hole that no real harm is done. Of course, this situation would be a local contamination incident. For intermediate-level and high-level waste repositories, the chances that a randomly located drill hole would hit a waste package are very small (for HLW about 2 in 1,000) because the waste was emplaced in a very dispersed configuration. In this case, the consequences would also be much greater, depending upon the length of decay of the waste. For a core drilling operation which directly penetrated a container of high-level waste after less than 600 years, both appreciable local contamination and significant radiation exposure of the drill crew could be expected.

An accidental penetration of a waste repository of this type probably would not result in large-scale breaching of the geologic containment. It seems logical to assume that any civilization having the technical capability to drill hundreds of meters into the earth in search of natural resources would also have the sophistication to recognize a waste repository, accurately diagnose the situation, and take whatever measures were indicated in order to prevent a large-scale uncontrolled release.

24.2.6 Repository Qualification Procedures

The geologic site selection factors and criteria are discussed generically in Section 23.2. Also discussed are some of the techniques and procedures of field investigations used to obtain the information required for geologic

evaluation of the suitability of a particular formation or rock type at a particular location. In a similar way, the general rationale for qualifying any waste storage or disposal facility is discussed in Section 23.3. For the various geologic storage alternatives, it is advantageous to broaden and extend those discussions so as to illustrate both the current approach to establishing actual waste facilities and the types of procedures expected to be required to qualify a final storage facility as a disposal repository.

First, it should be recognized that the specific objective of the geologic investigations is to identify a location for a pilot plant. One requirement is that the location would also be suitable geologically for a disposal repository pending successful completion of design confirmation experiments and waste handling demonstrations. The second and concurrent part of any development program is the completion of a field testing program whose specific objectives are the establishment of a data base for the design of a pilot plant. This field test program would be tailored to the requirements of the particular formation, but would involve electrical heaters simulating the thermal characteristics of waste and possibly radioactive source material to simulate radiation fields. Wherever possible, this sequence of field tests would culminate in an underground vault test simulating the salient features of pilot plant operations which would be carried out in some existing excavation, if such is available, or possibly in a newly excavated test facility created for that purpose.

The objective for both of these programs is the design and construction of a pilot plant storage facility. This pilot plant would handle limited quantities of real waste (in a fully retrievable configuration) and simulated waste or perhaps very small (experimental) quantities of real waste in disposal configuration. Associated with these pilot plant operations would be a broad range of in-situ experiments designed to confirm that the system evaluation is correct and that all performance factors have been considered. Operation of the pilot plant would continue until the objectives of these experiments had been met. Depending upon the geologic formation and the experimental requirements, this period may be either fairly short (2 to 5 years) or may be more extended (10 to 20 years).

After successful completion of the pilot plant experimental objectives, the pilot plant facilities would qualify for expansion into a demonstration plant. This expansion would consist mainly of an increase in the capacity of the facilities so that significant quantities of commercial waste could be handled. The objectives of this demonstration plant phase are first to demonstrate the full-scale waste handling operations and second, to continue those confirmation experiments which require long durations. Because of this second objective, all the waste handled during demonstration phase operations will be

placed into retrievable configuration (except for small experimental quantities of radioactive materials and simulated wastes). The objective of this demonstration plant phase operation is to qualify conversion of the facility from waste storage to waste disposal, that is, to modify the waste configuration so that it is no longer readily retrievable.

The mechanisms for the conversion from pilot plant to demonstration plant to waste repository have not been defined, and, of course, the details of those mechanisms would depend upon both the rock type and the location in question. However, since the pilot plant would be handling small quantities of commercial wastes and the intent of the larger range program is that the pilot plant evolves into a repository without removal of that waste (although it could be retrieved if necessary), it is expected that pilot plants will have to be licensed by NRC. It is anticipated that part of the licenses will define the criteria on which the decisions to proceed with subsequent demonstration plant and waste repository phases would be made.

During the pilot plant phase of operation, retrieval of emplaced wastes is expected to be demonstrated periodically. Retrievability technology will depend on the geologic media and waste emplacement method. Possible retrieval methods include mining techniques (solution mining, remote mechanical mining, overcoring, etc.) in addition to engineering the original emplacement so as to simplify later retrieval.

The recovery concepts being considered for the southwestern New Mexico bedded salt pilot plant illustrate some of the technical approaches to retrievability. Experiments proposed for this pilot plant include an array of high-level waste canisters to simulate the thermal and structural effects in an operational repository. One way to ensure recovery of these canisters is to preserve the canisters intact for the life of the pilot plant. A current concept for ensuring this utilizes sleeves or liners in the salt which will isolate the steel waste canister from the salt/brine environment. The sleeved emplacement mode does not adequately represent some aspects of the eventual operational disposal procedure, which is conceived to be the emplacement of waste canisters in direct contact with salt. To evaluate the consequences of this eventual emplacement concept, a limited number of waste canisters will be placed directly into salt. The corrosion of the canister and the subsequent phenomena significant to redistribution of radionuclides will be examined. The present retrievability concept envisions overcoring to obtain a "salt canister" containing the radioactivity above the decontamination level. This "salt canister" would probably be encased in a new container if it had to be removed from the pilot plant.

The time scale for pilot plant retrievability is limited to the maximum life of the pilot plant and is expected to be 10 to at most 20 years. This

phase of the repository could be described as ready retrieval in that engineered techniques will be developed and demonstrated for the retrieval of any of the waste emplaced during this period. Assuming that the decision is then made to turn the pilot plant into an operational disposal facility, there would not be an intent to retrieve waste from the disposal operation. While recovery of wastes from the disposal facility could be conducted, it would probably require different and more expensive techniques than those which were specifically developed for the 10- to 20-year pilot plant period.

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25.0 GEOLOGIC DISPOSAL ALTERNATIVES

25.0 GEOLOGIC DISPOSAL ALTERNATIVES

Geologic disposal is distinguished from geologic storage by the expectation that disposal will irreversibly rely on the stability and containment characteristics of the geology to keep the wastes in isolation. This irreversibility stems from anticipation that costs of retrievability would be exorbitant (cf. Section 23.3.4).

Numerous geologic disposal alternatives have been conceived. All are variations on the two basic variables available: the waste form at the time of emplacement (liquid or solid) and the geologic medium. Emplacement techniques are selected as necessary to get the waste into the medium.

The disposal concepts described here illustrate the range of alternatives that have been considered. Each is technically feasible with existing emplacement technology or extensions thereof. The major current technical gap is lack of a priori evaluations of the postemplacement stability and the containment characteristics of the geology and its contained waste.

25.1 DEEP CONTINENTAL GEOLOGIC FORMATIONS

The alternatives for disposal in deep continental formations differ in methods for waste emplacement and the way they rely on the geologic medium to keep the wastes isolated. Their common features are emplacement in formations beyond the biosphere, reliance on the geologic medium as the only isolation barrier, and emplacement without intent for retrieval.

Retrieval from geologic disposal is not necessarily impossible; it would, however, be expected to be difficult and expensive (cf. Section 23.3.4). The alternatives are discussed here in order of estimated increasing difficulty of retrievability.

25.1.1 Waste Disposal in Solution Mined Cavities

Salt, whether dome, bedded, or anticlinal, is the only rock type in which solution mining techniques can and are being used to construct large caverns, mainly for the storage of petroleum products, i.e., LPG, crude oil, etc. Basically, the technique consists of washing out the salt, utilizing fresh water. By manipulating the flow of fresh water, the position of the inlet, the location of the brine out flow pipe, the inert blanket, etc., it is possible to control both the size and the shape of the cavern.

The availability of this technique may make it possible to construct and operate a waste repository using only surface operated facilities. These might be considerably less expensive than similarly sized facilities constructed with conventional underground techniques. However, a final storage repository utilizing a solution mined cavern would be capable of handling only high-gamma transuranic and low-gamma transuranic wastes. This is because salt does not dissipate decay heat fast enough to prevent melting.

A brief description of a generalized final storage facility utilizing a solution mined cavity follows (see Figure 25.1). The facility will consist of:

1. waste receiving building
2. waste hoisting equipment and shaft
3. solution mined storage cavern
4. support facilities
 - a. administration building
 - b. warehouse
 - c. utilities
 - d. solution mining equipment suite (includes brine disposal).

FIGURE 25.1. Generalized Concept Solution Mining Final Storage Facility

The characteristics of high-gamma transuranic waste along with a description of a possible container (canister) can be found in Section 24.2.1. The high radiation level will require the same biological shielding-remote handling system used for receiving and handling high-level waste as described in Section 24.2.2.4. After the canister is placed in the hoisting device, it will be lowered into the cavern until it is near the bottom, then allowed to fall onto a random pile of canisters. (This technique has been demonstrated by the Federal Republic of Germany in its Asse waste disposal facility for high-gamma transuranic waste.)^(1,2) No special underground transporters, mine ventilation system, or access corridors, etc. are required.

The type of waste does determine the design of the receiving facility. For low-gamma transuranic waste which can be handled directly, standard industrial bulk moving equipment will be used to unload and move the waste, perhaps to the hoisting facility. Special handling will be necessary for many damaged or punctured packages before repair or overpacking is required.

High-gamma transuranic waste, however, will require a more complex receiving facility. Waste canisters, which will arrive in large shielded casks, will be unloaded by remotely operated equipment, moved into hot cells for inspection, monitoring, decontamination, repair (patching or overpacking), and finally, still using remote equipment, loaded into the hoisting facility. These low heat-generating wastes will not require special cooling facilities.

There are problems, however. Little is known about the stability of solution mined cavities when they are dried out. How big can they be, what shape is best suited for stability, what depths (below the land surface) can be utilized, and methods for drying the cavern are some of the unanswered questions. Also, the question of retrievability has not been adequately addressed. "Fishing" for canisters with a grapple on a long cable is not a demonstrated retrieval method. Low-level wastes do allow men to be lowered into the cavern and directly load canisters onto the hoist. Alternatively, a shaft could be drilled beside the cavern and an opening made into the cavern to allow direct loading of the canisters. The remote handling scheme for high-gamma transuranic has not yet been designed.

Disposal of salt should not be a problem in areas where brine disposal wells can be installed. It becomes a problem if the brine must be solidified and stored. This may limit the sites where a solution mineral storage facility could be built.

25.1.2 Waste Disposal in a Matrix of Drilled Holes

The matrix hole final storage concept would place waste canisters in holes drilled into a suitable geologic formation. A generalized concept of such a facility is shown in Figure 25.2. A matrix of holes, about 1 m in diam would be drilled into a thick, tight geologic formation with no cracks,

FIGURE 25.2. Solid Waste Emplacement in a Matrix of Drilled Holes

fractures, etc., to permit water to circulate. Waste would be received and, after inspection, monitoring, decontamination, etc., placed in a combination transporter-hoist vehicle, moved to the hole and lowered into position. The advantage of such a system is that vertical space in holes drilled from the surface is used for underground storage instead of horizontal space provided by mined caverns.

There are a number of problem areas in this concept, some of which are discussed. It is assumed that a thick (1000 to 3000 m), hydrologically tight, stable geologic formation can be found and qualified, that the spacing of both holes and canister to allow heat from high-level waste to dissipate without melting either waste or rock would not require an excessive amount of surface area as well as a large number of lightly loaded holes, and that the retrievability requirement can be met (cased holes?). Another question involves the problem of how to dispose of the drilling mud, particularly if

salt is the geologic formation used. Also, it would be necessary to carefully seal all the boreholes when retrievability is no longer required. The multitudes of boreholes penetrating the disposal formation, even though they are plugged, would drastically increase the probability that the integrity of the containment provided by the geologic formation could be compromised, with the results that it would be difficult to satisfy the long-term containment needed. Although these problems are formidable, there is enough merit in the concept to justify at least a paper feasibility study.

25.1.3 Waste Disposal in Superdeep Holes

The rationale for using superdeep holes (ranging from 10 to 20 thousand meters in depth) is to take advantage of this distance to assure that no conceivable climatic or surface change would expose primarily high-level waste to the biosphere. A final storage facility using this concept would consist of a large number of large-diameter holes drilled into a thick sequence of rocks having a low permeability. Waste canisters would then be lowered into the hole, filling the bottom one or several thousand meters of hole, after which the hole would be sealed.

There are a number of problems inherent to this concept. First, the technology does not exist today to drill a large-diameter (6 m or longer) hole to these depths. The deepest hole to date, 9,500 m started as a 0.76-m-diam hole and was 20 cm in diam at the bottom. This operation was expensive and drilling took a long time (greater than 500 days). Even if this concept was used only for high-level waste where the depth might be needed, the number of holes needed each year would require an enormous investment in drilling rigs, not to mention casing, etc. Further, the heat generating levels of high-level waste would probably severely limit the number of canisters per hole if melting of waste and rock is not permitted. Obviously, these problems along with the many others not discussed make the engineering feasibility of this concept doubtful. Thus, it is not considered a viable alternative disposal concept at this time.

25.1.4 Hydrofracture

Hydrofracturing (known commercially as "hydrafrac") involves fracturing geologic media around a borehole in order to stimulate either oil and gas production or disposal of wastes. The technology is routinely available as a commercial service; Hubbert and Willis estimated that at least 100,000 wells had been hydrofractured between 1947 and 1957.⁽³⁾ As of January 1975, 209 industrial waste disposal wells were being used, and a small unknown number of these were hydraulically fractured.⁽⁴⁾ Several surveys have been made of the use of hydrofracture for waste disposal.^(5,6)

Use of hydrofracture for waste disposal involves a three-step process:

1. Breakdown of the geological formation. A viscous fluid, to which has been added a gelling agent and a propping agent such as silica sand, is pumped under pressure into the well until the geologic structure fractures.
2. Preparation for waste injection. A fluid containing a gel-breaking agent is pumped in; the fluids are drained out, leaving the propping agent behind to keep the fractures open.
3. Waste injection. The waste fluids, which contain grouting agents such as cement and diatomaceous earth, are pumped into the fractures. The grouting agents subsequently harden, leaving the wastes fixed in position in the fractured geologic structure.

Investigation of the use of hydrofracture for disposal of radioactive wastes began in 1959 at Oak Ridge National Laboratory.⁽⁷⁾ Use of the technique has subsequently become a routine operation at ORNL for disposal of intermediate-level radioactive wastes.⁽⁸⁾ The wastes have all been disposed of into the Conasauga shale between the depths of 700 and 1000 ft through one injection well. Through 1975, a total of 1,823,000 gal of ILW waste grout containing more than 560,000 Ci of activity have been injected. Site proof studies have been completed for a new location about 1000 ft south of the present injection well, but this new site will not be used for waste disposal before the 1980s.⁽⁹⁾ A similar site proof study was completed for a potential hydrofracture site at West Valley, New York, by de Laguna⁽¹⁰⁾ and Sun⁽¹¹⁾ (1974). As of 1975, this site has not been used for hydrofracture injection of radioactive waste.

The primary safety concern associated with use of hydrofracture for radioactive waste disposal is the integrity of long-term containment. There are no readily available data on long-term containment of industrial waste disposed of by hydraulic fracturing. Investigations at two industrial hydrofracture disposal sites showed that containment problems arose from induced seismic activity resulting from the increased pressures at depth as a result of the liquid injections.^(12,13)

Rock-cover wells have been used to monitor the long-term containment of the intermediate level waste injected at Oak Ridge National Laboratory.⁽⁷⁾ As of 1975, there has not been any detectable increase in the permeability of the Conasauga shale at the site. There is no indication of any post-injection movement of radioactivity away from the grout sheets.

25.1.5 Deep-Well Injection

Disposal of liquid wastes by injection into deep wells is well established industry practice.⁽¹⁴⁻¹⁸⁾ The basic concept is simple: the liquid is forced

through a pipe into the geologic formation. Pumping pressures required range from zero (gravity flow) to about 10^6 kg/m², depending on the properties of the formation.

Recommended⁽¹⁶⁾ properties of the geologic formation include a porosity of 10 to 30%, a permeability of at least 25 millidarcies, and a depth of about 1000 m. The formation must be bounded above and below by impermeable strata. Formations possessing these characteristics can be found in the sedimentary basins of the U.S.⁽¹⁷⁾

The major safety consideration for this alternative is to prevent leakage into aquifers containing potable water. This is accomplished by selecting sites where the formation that will contain the waste is not connected by water-transmitting faults to aquifers. Other important safety factors are proper casing of the well and monitoring and maintenance of integrity of all pipes and casing.

The technology needed to use this alternative is currently available, as evidenced by over 40 years of use in recharge of brine accompanying crude oil production. Over 40,000 such recharge wells have been operated, and over 300 industrial waste disposal wells have been used.

Deep-well injection has been investigated as a method for disposal of tritiated water.⁽¹⁹⁾ Its potential for use with liquids containing long-lived or high levels of radioactivity has not been evaluated.

25.1.6 Rock Melting Concepts

Because of the large thermal powers of high-level nuclear wastes (as high as 200 kW/m³ for 1-year-old solidified waste), methods for its disposal are sensitive to the problem of heat management. Depending on the thermal properties of the waste and surrounding geology, the power output of the waste and its configuration, and the arrangement of the disposal site, temperatures in the waste and surroundings can range from only slightly above ambient to well above the melting temperatures of rocks. In this section, disposal schemes which achieve melting in the surroundings are discussed.

No disposal technique involving melting has yet been extensively investigated. Consequently, the discussion in this section is based on preliminary calculations and experiments and, in some cases, conjecture.

Several rock melting disposal schemes have been formally proposed. One suggests direct emplacement of liquid waste into a cavity formed in silicate rock.⁽²⁰⁾ Liquid wastes are injected into the cavity where, because of self-heating, the water from the waste is boiled off. In order to avoid premature melting of the cavity, external cooling is supplied during the charging period of the site. A second proposal suggests drilling holes about 1 m in diameter several kilometers deep into competent bed rock and emplacing solidified waste

in the last several hundred to one thousand meters.⁽²¹⁾ In both methods, the thermal power of the waste would be sufficiently high to cause the surrounding rock to melt. A third proposal suggests the containment of wastes in a capsule which is sufficiently refractory so that while the surrounding rock melts the capsule does not. With a design of sufficient density, the capsule descends slowly through the host rock.

Other emplacement methods which cause melting include variations developed from other types and shapes of cavities, the use of liquid or solid wastes, and the application or not of interim cooling. An important set of variations is likewise obtained from considering alternatives among geologic media.

The essential requirement for waste to be disposed of in a rock melting method is a sufficiently high thermal power to bring the host geologic medium to its melting range. As an example, waste having a thermal power of about 10 kW/m^3 will cause melting if deposited in a 1-m-diam cylindrical geometry in typical basalt. Media having low thermal diffusivities and low beginning of melting temperatures, or wastes having long decay times and small surface-to-volume ratios will tend to enhance the melting process.

Rock melting methods are intended mainly for disposal of high-level wastes. Dilution of HLW with low-level wastes, or old wastes with low thermal power, or cladding hulls is possible. Both liquid and solid wastes can be disposed of by this technique.

Containers for most rock melting techniques are not intended to serve any purpose beyond necessary handling on the surface and during emplacement. If emplaced with the waste, the container will be rapidly breached by melting and hence could not serve as a significant barrier to radionuclide transport. An exception is the self-burial concept wherein the container completely or substantially contains the waste for an appropriate descent time while melting through the underlying rock.

A generalized rock melting disposal process involves:

1. Emplacement of the waste at some depth in a suitable host rock.
2. Increase in the temperature of the rock in the immediate vicinity of emplacement to a value at which melting begins.
3. Migration of the boundary between crystalline and partially liquified rock away from the emplacement region.
4. Generation of a completely molten zone in the immediate vicinity of the emplacement region.
5. Possible mixing, in a chemical sense, of the container and waste material through dissolution in the rock melt, presumably aided by convective transport.

6. Crystallization of the liquified rock, possibly with admixed waste and container material, as the thermal power decays with time, resulting in a dilute waste/rock matrix.

The stages in the generalized rock melting process relative to the time and distance parameters are illustrated schematically in Figure 25.3.

The following discusses generically the behavior of rock-container-waste systems, with emphasis on high-temperature chemical interactions. Such interactions are fundamental to these disposal concepts.

Availability of Host Materials

Important chemical, mineralogical, mass, and volume data pertinent to the Earth's crust are given in References 22 and 23. The crust is the outer skin of the Earth averaging about 40-km thick in the continental regions and less than 10-km thick in the oceanic regions. The most abundant rocks of the Earth's continental crust are the granitic rocks (including granites, granodiorites, quartz monzonites, quartz diorites, and quartzofeldspathic

FIGURE 25.3. Schematic Time/Temperature Relations for an In Situ Melting Event. Curves separate molten from partially molten and partially molten from unmelted zones.

gneisses), while the basaltic rocks are the most abundant in the oceanic crust. Sediments and sedimentary rocks of highly variable thickness rest on the underlying granitic and basaltic rocks. They rarely exceed a few kilometers in thickness on the continents and are less than a kilometer thick in the oceans. In view of their relative abundance and accessibility near the Earth's surface, shales (argillaceous rocks) and carbonates (limestone and dolomite) are potentially particularly important for disposal of radioactive waste. Crystalline schists and gneisses, though frequently buried beneath the sedimentary rocks, represent volumetrically abundant potential hosts for waste disposal. In the western United States, there are large volumes of extrusive volcanic rocks, locally exceeding 2 km in thickness, which represent potential hosts for the disposal of radioactive waste. Types include flows, ash-fall, ash-flows, and welded and nonwelded tuffs.

Other than the carbonate rocks and salt deposits, the principal rocks for possible disposal of high-level radioactive waste are composed of silicate minerals. All are in many ways similar in their high-temperature and high-pressure behavior. These rocks are multiphase, polycomponent aggregates composed principally of quartz, the alkali and alkaline earth aluminosilicates (feldspars), sheet structure alkali aluminosilicates (micas), chain structure ferro-magnesian silicates and alumino-silicates (pyroxenes and amphiboles), and the ferro-magnesian orthosilicates (olivines). The proportions of these phases varies widely in the crustal rocks and provide the basis for the rock name classification used in the igneous rocks. The crystal chemistry of these mineral groups is dominated by the tetrahedral $(Al, Si)O_4$ ionic group. For the most part these minerals are crystalline solutions composed of the light elements: O, Na, Mg, Al, Si, K, Ca, and Fe. Consequently these are the most abundant elements of the rocks forming the Earth's crust.

Melting Relations

Mineralogical and textural changes are produced by increasing temperature and pressure in rocks. Each rock behaves somewhat differently, and it is a reasonably straightforward experimental task to determine the mineralogical, chemical, and textural changes which take place. A sequence of changes in mineral assemblage and texture take place with heat before the onset of melting. These metamorphic reactions are observed both in the laboratory and in the field.⁽²⁴⁾ If heating takes place with sufficient rapidity the "equilibrium" state may not be attained⁽²⁵⁾ and the subsequent melting history will not necessarily be the same as for equilibrium.

Major problems are found in scaling from small samples (<1 g) in the laboratory to the natural waste disposal environment, because of the possible importance of convective mass and energy transport in the large, but not the small-scale, domain.

Rocks are characterized by a melting interval rather than a melting point. The magnitude of the melting interval ranges from a few degrees to several hundred degrees celsius. The melting interval is defined by the solidus temperature, that temperature at which a liquid first forms on heating, and the liquidus temperature, that temperature above which crystals no longer exist stably. The solidus and liquidus temperatures, and hence the melting interval, depend on pressure, chemical composition of the rock, amount of H_2O and other "volatile" components, oxidizing or reducing environment, the degree to which the rock system remains chemically closed during the melting process, and the state of aggregation of the rock. In general the effects of these variables differs for the solidus temperature and the liquidus temperature.

Silicate liquids, formed by the melting of crustal rocks, are capable of dissolving significant amounts (to about 20 wt%) of H_2O and other volatile components at pressures corresponding to lithostatic pressures in the Earth's crust.^(26,27) In general, the pressure/temperature curves defining the solidus and liquidus have positive slopes for anhydrous rock systems. This reflects the lower density of the liquid relative to the chemically equivalent crystalline assemblage in the case of the solidus temperature. However, in the presence of sufficient H_2O to saturate the silicate liquid, the solidus P/T curve has negative slope, reflecting that the liquid has a higher density than the chemically equivalent crystal plus aqueous vapor (fluid) assemblage. In this case the liquidus curve may have either a positive or negative slope. The two--hydrous and anhydrous--cases are shown in Figure 25.4 for a typical granite rock. A silicate melt containing dissolved H_2O is capable of crystallizing hydrous minerals such as micas and amphiboles at high pressure and over a range of temperatures. These minerals do not stably crystallize at low pressures from the hydrous liquids.⁽²⁸⁾ The micas and amphiboles typically have the capability for tolerating more varied crystalline solution with minor elements than do the feldspars or quartz. Potentially at least, high pressure and the presence of H_2O may favor fixation of potentially hazardous radionuclides in these silicate structures.

During the melting of crustal rocks the amount of available H_2O , both as a pore fluid and as the product of dehydration of hydrous phases, is a particularly important variable. If the amount of H_2O is not sufficient to saturate the liquids formed during the melting process, then the liquid (being undersaturated with respect to H_2O) may act as a sink for H_2O from the surrounding rocks.⁽²⁹⁾ This could establish a net flow of H_2O from the surrounding rocks to the location of the liquid. Flow of H_2O in the opposite sense could occur during the subsequent crystallization history. During the crystallization process, the principal silicate phases are either anhydrous or contain little H_2O . Consequently the liquid may become saturated with an

FIGURE 25.4. Schematic Illustration of Hydrous (~~~~) and Anhydrous (—) Melting Intervals for an Average Granite

aqueous fluid phase of low density and viscosity. The fluid phase may then escape from the site and interact with the surrounding rocks. In rocks surrounding normal intrusive igneous rocks and effects of such fluids are often quite evident.⁽³⁰⁾ They are responsible for chemical transport of W, Mo, and Sn leading to the formation of economically important ore deposits.⁽³¹⁾ They also cause alteration of primary silicate minerals to clay minerals and micas.⁽³²⁾ Though dominantly H_2O , other volatile components such as Cl, F, S, and CO/CO_2 are also present in the fluid phase.⁽³³⁾

In general the composition of the liquid formed by melting of crustal rocks changes during the melting process. At the onset of melting the composition of the liquid is most different from that of the bulk rock composition and as the melting process goes to completion the composition of the liquid approaches that of the rock. The proportion of liquid, relative to residual crystalline material, is a distinctly nonlinear function of temperature over the melting interval. Both positive and negative deviations from linearity are observed.⁽³⁴⁾ Many important physical properties, such as thermal conductivity and diffusivity, heat capacity, viscosity, density, compressibility, and thermal expansion are dependent on chemical composition. Consequently, the existence, magnitude, and nature of the melting interval are of importance in the development of analytical models for the time-temperature-stress behavior of the radioactive waste disposal scheme. Details are discussed in References 35-43.

Another factor affecting rock melting is the state of aggregation of the rock material. This factor will be of particular importance to melting relations for basaltic rocks and the silica-rich volcanic ash flows, tuffs, and welded tuffs. These rocks characteristically have relatively high porosity and may contain large amounts of glass or devitrified glass. The changes in mineral composition and texture of such rocks may be much greater during the heating cycle than for completely crystalline chemical equivalents, so their melting history may be dependent to a large degree on heating rate as well as the variables discussed previously.

Another factor of some importance in the analysis of the melting process is the physical migration of the liquid from its site of origin. It is possible that the liquid may migrate from the region of melting, with or without entrained crystals, as a consequence of density contrasts or under the influence of natural tectonic forces. If this takes place the residual unmelted material will be more refractory than before. In the melting of natural material the first-formed liquid is found as an intergranular film⁽⁴⁴⁾ rather than as discrete and nonconnected pockets. It is to be expected that the mechanical properties of the crystal-plus-intergranular liquid will be markedly different from either the crystalline rock or the ultimate liquid and this factor should be included in analytical modeling pertinent to any in situ melting scheme for radioactive waste disposal.

Rock/Waste Interactions and Crystallization

If the high-level waste, its host, and container interact or dissolve in the rock melt, it is necessary to consider the detailed effects of this chemical interaction. Even if a fixed high-level waste composition is assumed, there are wide variations in the possible waste/host ratio and considerable variability in the chemical composition of the host. The effects of mixing on solidus/liquidus relations and on melting and crystallization sequences would require experimental study for each case. It is important to note that the variables discussed previously in connection with the melting process are of similar importance in the analysis and treatment of the crystallization and gravitative settling of dense crystals in the slowing crystallizing natural silicate liquids.⁽⁴⁵⁾ If high-density U and Pu silicates or oxides crystallize early from the liquid, it is possible that these phases could be locally concentrated during the crystallization process. Gravitative settling of such phases could pose two potential problems: concentration of long-lived actinide thermal sources in a restricted spatial domain, and the development of local concentrations of fissionable material.

During the cooling period it is possible that the molten rock (plus waste and container) material may either crystallize or "quench" to a glassy form.

The data on rates of nucleation and crystal growth in natural and synthetic systems^(46,47) suggest that crystallization is to be expected for most geometrical configurations of emplacement.

Knowledge of the partition coefficients for radionuclides (and their decay products) between the various possible crystalline phases, the liquid, and any aqueous fluid phase are of major importance in predicting the spatial location of these elements. It is possible that trace-element data obtained from geologic systems may be useful in treating this problem.^(48,49) The concentrations of potentially toxic or hazardous nuclides in the rock waste melt system may be orders of magnitude greater than their analogues in natural systems, suggesting possible inapplicability of trace-element partition coefficient models derived from dilute solution theory.

Summary on Geologic Materials

The above discussion illustrates the problems which would have to be resolved before implementing in situ melting. The large number of variables involved in predicting the melting and crystallization behavior of the rock/waste system indicates that extensive site and emplacement mode specific studies would be required to evaluate the feasibility of high-level waste disposal involving in situ melting. The technology exists for many aspects of these studies, particularly in terms of small-scale laboratory level experiments. These experiments could provide information as to whether larger scale experiments or even pilot plant operations are warranted. Major problems exist in scaling the results of laboratory experiments, conducted on sample sizes of a few grams or less, to actual disposal. These scaling problems reflect inadequacies in our understanding of transport (mass and energy) processes. Both additional specific experimental studies and advances in numerical-analytical modeling techniques are required before in situ melting can be evaluated for disposal of high-level radioactive waste. It also appears that advances in theoretical understanding of the basic processes are required.

Transport in Rock Melt Systems

There are two forms of transport pertinent to the disposal of high-level waste: mass migration of radionuclides and heat transfer. These are discussed below as they relate to rock melting disposal methods. Since only limited work has been done on rock melting disposal, these discussions are anticipatory rather than definitive.

Radionuclide Transport in Rock Melt Systems

While the problem of radionuclide transport from a rock melting disposal site has not been directly addressed in the literature, it is possible to indicate qualitatively how migration might proceed. Of the various migration processes which can be identified, it is not completely clear which are the more important.

The first factor to consider is the nuclide source which feeds the migration processes in other geologic zones. If the emplaced waste form is liquid, leakage into joints in the surrounding rock may initially be an important migration process. However, as long as the liquid waste remains concentrated, radioactive heating will cause the water to be boiled off and calcining to occur. Thereafter, while details of the chemistry may differ considerably, the general processes are similar for both liquid and solid wastes. Self-heating occurs, leading to melting, first of the waste itself and then of the surrounding rock. Ensuing processes include convective and diffusive mixing which promote chemical reactions between the waste and the rock. At this stage, it may be presumed that the source is sufficiently hot so that liquid groundwater will not directly contact it and, as a result, water leaching will not be an issue. This does not imply that chemical reaction with water vapor may not be important. Diffusion and vaporization, on the other hand, may be very important release mechanisms since both are promoted by high temperatures. Later stages in the life of the rock melt process will be characterized by reduced temperatures, freezing of the melt and, potentially, material segregation. While temperatures are lower than peak values at this time, they may yet be high enough so that, when coupled with large gradients in both temperature and concentration, diffusion and vapor release remain important. During the last stage of the source lifetime (the longest time interval), the source is relatively cool and inactive. At this stage, water leaching is probably the dominant potential release mechanism.

Transport in the geosphere near the source may proceed by different processes at different times. Diffusion and vapor transport may initially be important. Later, as the system cools, groundwater transport would probably dominate the migration process and, at distances far from the source, this same mechanism would likely be the only important one throughout the life of the disposal site.

Transport mechanisms involving water have been the most extensively studied. Leaching has been studied for rock/waste melt mixtures only in a very limited set of experiments.⁽⁵⁰⁾ However, there have been extensive studies of the process with various waste forms.⁽⁵¹⁻⁵⁶⁾ While not completely applicable, these studies do provide a basis for interpreting leaching in waste/rock mixtures.

Studies of transport of a variety of fission products and transuranics in ground water have been conducted for several soils.⁽⁵⁶⁻⁶⁵⁾ This process is closely akin to chromatography with adsorption and ion exchange figuring prominently, but with the additional complexity of varying chemical behavior due to radioactive decay. Since for rock melting disposal much of the transport would presumably take place through the narrow joints of deep competent rock, the migration process will be different from that considered in these studies.

The subject of mass diffusion under concentration and thermal gradients has been extensively studied in simple systems. These processes are poorly understood in systems as complex as the geologic environment. In addition, the required experimental work has not been undertaken and it may be an extensive task in view of the chemical and physical complexity of the system. Data would be needed to understand leaching in the source and, potentially, to explain some aspects of transport in the media adjacent to the source.

The process of vapor transport is potentially complex. Not only is the rate of vaporization of particular radionuclides difficult to predict because of the complexity of the source chemistry, but it is also possible that vapors, particularly water and carbon dioxide, will be expelled due to decomposition of the rock. Consequently, the vapor transport process may involve the coupled effects of vaporization, condensation both on solid surfaces and in the gas phase, re-evaporation due to self-heating, rapid motion of the ambient gas due to gas production at the source, and chemical reactions in the gas phase and at surfaces.

Heat Transfer

In general, a complete heat-transfer analysis for a waste disposal option would consider at least the following factors: the waste power; its decay characteristics; the waste emplacement schedule; convection in the molten pool; the range of radiation levels in the waste and rock; stored energy due to radiation effects; the distribution of radioactive materials in the molten pool; the changing geometry of the heat source due to thermal expansion; mixing, vaporization, and other forms of material migration; gross migration of the source; appropriate boundary conditions such as the Earth's surface or sufficiently isothermal aquifers; appropriate geologic material properties allowing for stratigraphic variations, rock joints, the local hydrology, and alterations in hydrologic flow patterns due to heating from the disposal site; and interaction with nearby disposal sites, if any.

No such analysis has been carried out for a rock melting disposal scheme; preliminary analyses which display the general features of the heat transfer evaluation have been done for several cases, however.⁽⁶⁶⁻⁷⁰⁾ These analyses have necessarily made greatly simplifying assumptions but show the following features. Following emplacement of the waste, there is a period of increasing temperature and enlarging molten zone. Eventually, because of the decay in the heating rate, a maximum melt radius is achieved and resolidification occurs, proceeding from the outside toward the center. The time to reach a maximum melt radius may vary from a few years to 100 years depending on, among other factors, the waste quantity, decay characteristics, age at emplacement time, emplacement schedule, and the geometry of the disposal site. Maximum melt radii range from the initial emplacement radius to approximately nine times that radius, again depending on the same factors.

Convection in the molten pool has been the subject of only the most preliminary study.⁽⁷¹⁻⁷³⁾ It has been shown experimentally that it does occur in a particular rock/waste (simulant) mixture and geometry.⁽⁶⁶⁾ The capacity of present analytical methods to predict the magnitude and nature of convection is limited at best and at worst may be entirely inadequate.

Bulk migration of the molten pool is a possibility which needs further analysis. The presence of convection implies the possibility of upward motion of the molten pool and spreading at the top. One calculation for the case of a spherical pool indicates upward motion of only about one-half the final melt diameter.⁽⁷³⁾ Downward motion might also occur due to appropriate segregation of heat-producing isotopes or gravitational effects.

The thermal expansion of the molten waste/rock mixture may give rise to significant volume increases and profoundly affects the heat transfer behavior. The lower power density of the expanded waste/rock mixture provides negative thermal feedback to the melting process and tends to self-limit the maximum melt size.

An effect which has formerly not been considered in the rock melting context, but which may be highly important, is thermal expansion of the surrounding solid rock. If it is of low porosity, the expansion volume would tend to be taken up in the emplacement cavity, causing displacement of the molten material and modification of the systems heat transfer behavior. Likewise, the solid thermal expansion would induce thermal stresses in the rock. A preliminary estimate of these stresses indicated that they might exceed the fracture strengths of rocks.⁽⁷⁴⁾

Example Rock Melt Systems

Rock melting disposal concepts include a broad array of specific options differing by initial waste form, emplacement geometry and technique, extent of waste/rock mixing, etc. The four specific descriptions that follow were chosen to illustrate the diversity of options that exist and, as seen in Table 25.1, span wide ranges in parameter variation. Many other rock melting disposal concepts are possible.

Concept 1: Mined Cavity/Liquid Waste/Interim Cooling. As presently envisioned, in this concept, a cavity would be mined in an appropriate, isolated geologic medium deep under a fuel reprocessing site.^(20,75) A cavity volume of about 6000 m³ (a sphere of about 12-m radius) could dispose of 25 years' waste from a 5 ton/day plant.

High-level liquid waste would be injected directly from the plant into the cavity. Decay energy from the waste would quickly bring the liquid in the cavity to boiling, and cooling water would have to be added to maintain the cavity at relatively low temperatures and steam pressures. When the

TABLE 25.1. Parameter Variations in Rock Melting Concepts

<u>Concept Number</u>	<u>Emplacement Geometry</u>	<u>Initial Waste</u>	<u>Waste/Rock Mixture During Melt</u>	<u>Required Container Life After Emplacement</u>	<u>Time from Initial Emplacement to First Melt</u>	<u>Time from Melt to Resolidification</u>
1	Cavity L/D ~ 1	Liquid	Complete	No container required	Up to tens of years	Up to hundreds of years
2	Cavity L/D ~ 1	Solid	Complete	Tens of years (for as long as retrievability is desired)	Up to tens of years	Up to hundreds of years
3	Cavity L/D $\gg 1$	Solid	Complete	Days	Down to hours	Tens of years
4	Self-Burial	Solid	None to complete	Up to tens of years	Down to hours	Locally days; total system tens of years

cavity was filled with nuclear waste, cooling would be stopped and access holes to the cavity sealed. Melting of the waste and the surrounding rock would commence almost immediately.

For this example (25 years' waste from a 5 ton/day plant) calculations show that melting would continue for about 65 years, by which time the maximum melt radius would be 96 m.

The advantages of this concept include immediate removal of the waste from the biosphere, minimal handling or processing of the waste between process stream exit and final disposal, and the avoidance of transportation of the waste from the processing site to the disposal site.

The technical problems raised by introducing liquid waste with long-term boiling into a cavity include need to control steam transport of radionuclides, need to prevent leakage of the waste into a zone containing mobile water, and potential need for special design features to mitigate buildup of silica scale in the steam exhaust line.

Concept 2: Mined Cavity/Solid Waste/Interim Cooling. This concept for rock melting disposal of high-level nuclear waste offers the option of retrievable storage of contained, solidified waste in mined cavities followed by eventual disposal in a melted and resolidified rock matrix. A typical storage/disposal scenario can be summarized as follows:

High-level nuclear waste from a nuclear fuel reprocessing plant is converted to a stable solid and sealed in appropriate storage canisters. These canisters are then delivered to a central storage/disposal facility where a cavity has been mined in an appropriate rock formation deep underground. There cooling water is circulated around the canisters to dissipate decay heat and prevent their melting. The cooling water is circulated to the surface of the ground and passed through a heat exchanger to maintain a low enough temperature. The rock cavity containing the waste canisters would be kept filled with water, and canisters would be handled when necessary by remotely controlled equipment. Any or all canisters could be retrieved at any time.

After the cavity had been filled to capacity with nuclear waste canisters it would only be necessary to stop the flow of cooling water to dispose of the nuclear waste in an irretrievable way. As soon as the water in the cavity had boiled away, the canisters, together with their contents and the surrounding rock, would fuse into a sphere of molten rock in which the waste would be dissolved. For final disposal to work properly, enough rock must be melted to dilute the nuclear waste to a low concentration.

This storage/disposal concept may not be as attractive economically as the preceding one, but it does have some potential advantages:

- Ready retrievability of the waste while in storage.
- Possibility for one or a few centrally located facilities which could handle waste from several or all of the nation's fuel reprocessing plants.
- A sensitive detection system to monitor for possible failure of a waste canister can be installed in the cooling water.
- Deferral of waste/rock chemical reactions until the melting phase of the ultimate disposal procedure is initiated. The interim period could be used for testing.
- Less likelihood of radionuclide transport to the surface with steam return.

However, there are some disadvantages to this concept (in addition to economic penalties), in comparison with liquid waste disposal:

- More complex from the standpoint of engineering design, construction, and operation.
- The requirement for additional processing and handling of high-level nuclear waste.
- Less diversity in kinds of waste handled. This concept is not as readily applicable to low- or intermediate-level radioactive waste.

Concept 3: Deep Drilled Hole/Solid Waste/No Interim Cooling. This concept, termed deep rock disposal (DRD), would be used for high-level or combined high- and low-level radioactive wastes. Figure 25.5 shows the stages of a typical DRD process. Holes about 0.3 to 3 m in diam are drilled several km into competent bedrock or basement rock. Solidified radioactive waste, either uncontained or in expendable containers, is then placed in the bottom section of the drill hole. Radioactive decay heat produced by the waste melts the waste, the surrounding rock, and containers if they are used. Natural convection currents mix the waste and rock. As the size of melt surface increases and the heating rate of the waste decreases, a time is reached when conduction losses into the surrounding rock exceed the heat generated and the rock/waste matrix begins to resolidify. After complete resolidification (typically 5 to 20 years) the waste has been substantially diluted in the rock. The result is a product which may have low leachability. It has a low total surface area, and is emplaced several km below the Earth's surface. After resolidification of the matrix, the top of the melt cavity may be sealed with a low-melting-temperature glass which can be melted by the heat in the matrix and surrounding rock. After further cooling, the glass will resolidify and the remainder of the access tube can be filled with concrete and/or other suitable fill material.

A typical disposal configuration (use of 300 m of a 1-m-diam hole in bedrock, 10 holes per year), would dispose of the predicted Year 2000 inventory of commercial high-level waste.

FIGURE 25.5. Stages of Deep Rock Nuclear Waste Disposal

The deep rock disposal method includes features which may be advantageous. Isolation of the waste does not depend on long-term integrity of man-made containers. The drilled hole has a large length-to-diameter ratio which would tend to mitigate possible upward migration of the melt zone. The configuration also constrains the time the matrix is liquid as well as the possible dilution. The relatively low total heat generation per unit length of a cylinder coupled with the "thermal expansion quenching" effect constrains heat-induced stresses and upheaval problems in the unmelted rock. The small melt radius of the cylindrical configuration limits temperatures at the center of the melt and thereby limits volatilization. Active cooling may only be required during loading.

Significant technical uncertainties not now resolved include the long-term radionuclide migration and transport in the geologic media; the geologic conditions of deep bedrock; the details of heat transfer in this conducting, convecting, and reacting system; and the transport of volatile and gaseous products from the waste.

In modeling studies of such a disposal system, a waste with an "effective" half-life of 4 years was assumed along with typical thermal properties for crystalline rocks. The waste, the waste/rock mixtures, and molten phases were further assumed to have the same thermal properties as the surrounding rock. The initial hole diameter was 1 m. The analysis was done in a nondimensional form using a finite-difference code, CINDA.⁽⁵⁸⁾ Some results of

these thermal calculations are shown in Figure 25.6. Two curves of maximum melt radius are shown. The upper curve corresponds to the assumption that no thermal expansion occurs. For an initial axial power density of 7 kW/m, a maximum melt radius of 2.0 m is found. This occurs at approximately 2.35 years. If it is assumed that the melt is expanded by 10%, the result is quite different (lower curve). Here the maximum melt radius is 1.0 m occurring at about 1.65 years.

FIGURE 25.6. Maximum Melt Radius as a Function of Axial Power Density (from Reference 57)

An experiment to test the DRD concept has been conducted using simulated waste, electrical heating, and a block of dolerite 3 ft x 3 ft x 4.5 ft high. (21,50,78) The block was placed in the earth in order to improve the heat-transfer simulation. The heater and surrounding tubing were 3 inches in diameter and 2 feet long. The power to the heater was ramped upward over a 45-day period to a maximum of 4.1 kW, held for several days, and reduced to a very low value for several more days. The maximum temperature reached in the melt was somewhat over 1300°C. Observations included verification of heat transfer and thermal stress calculations, the existence in the melt zone of substantial convection, the sealing of cracks in the rock by the molten material, and substantial mixing and reaction of the waste and rock. Post-test analyses also showed substantial segregation and stratification of the rock waste product.

Concept 4: Solid Waste/Capsule/Deep Descent. The rock-melting-capsule concept^(79,80) uses decay heat from high-level radioactive wastes in a container to melt rock. Descent by gravity achieves self-burial. Molten rock resolidifies in the wake of the capsule, providing isolation from the environment.

A drilled and partially or fully cased hole up to 2-km deep has been proposed⁽⁷⁹⁾ to permit waste capsule release and the start of rock melt at a safe depth. The waste-filled container, with a temporary separable cooling system attached, is lowered to the bottom of the hole. Remote preparation for release includes adding a startup thermal contact medium such as granular aluminum, which flows into the space between the capsule and rock walls. The cooling system is then shut down and retrieved for reuse. In the absence of cooling, the waste contents heat up and melt, the capsule surface temperature rises, and the thermal contact medium melts, followed by the start of rock melt. With the capsule density greater than the rock density, the capsule settles and displaces the melt in a continuing process (Figure 25.7). The velocity of descent is determined by the rate of energy exchange with the rock underneath and by hydrodynamic limitations associated with rock-melt displacement. Following passage of the capsule, the molten rock wake solidifies, forming a permanent seal. After a suitable delay, the bottom of the hole may be prepared to receive the next capsule, thus permitting many capsules to be released from the same hole.

Calculated results for descent in basalt rock, Figure 25.8 and Table 25.2, are for a blend of 80% waste from enriched U fuel and 20% from Pu recycle fuel, with 40 wt% calcine incorporated in borosilicate glass. Each capsule has a capacity of waste from 45.5 metric tons of fuel, representing 1.66 reference reactor years. Early failure of the capsule would reduce the depth attained, but still permit self-burial to be accomplished. Lower concentrations of waste, such as the nominal 25 wt% calcine content proposed for disposal in bedded salt, may also be used; significant descent is attainable for waste aged to 20 years or more before release. Capsule capacity might be augmented by use of elongated rather than spherical containers, but capsule size is a trade-off among several factors: handling convenience and safety during loading and emplacement; diameter of required emplacement hole; and thermal properties of waste to be accommodated. Descent in granite would require temperatures above 1500° C for adequate reduction of rock melt viscosity.

Use of the concept in salt domes is less difficult than for granite. Alloys such as Hastelloy N have good corrosion life in molten salt at approximately 800°C. Salt domes extend from at or near the ground surface to depths as great as 10 km. Self-burial is the only method which can achieve

FIGURE 25.7. Descending Waste Capsule

FIGURE 25.8. Depth Attained Versus Time of Descent in Basalt

TABLE 25.2. Calculation Results for 75-cm-Radium Waste Capsule Descent in Basalt Rock

Waste Age, yr	Surface Temp, T_c °C	Heat Generation, q''' cal/(cm ³ s)	Velocity v m/day	Surface Temp, T_c °C	Descent Time, yr	Depth, z km
≤ 1.1	> 1400	q''' too high				
1.2	1390	0.073	3.04	1150	66	9.77
2.0	1290	0.048	2.05	1150	64	8.89
3.0	1200	0.034	1.48	1150	64	8.20
5.0	1150	0.022	0.93	1150	60	7.18
7.0	1150	0.017	0.66	1150	58	6.55
10.0	1150	0.012	0.45	1150	54	5.88
15.0	1150	0.010	0.36	1150	48	5.06
20.0	1150	0.009	0.30	1150	43	4.43

Container: 3-cm wall thickness, 8 g/cm³ density, 10% voids

Contents: 90% waste, 3.6 g/cm³ density, 5% void,
5% module containers & ballast, 7.2 g/cm³ density

Average density of capsule: 4.01 g/cm³.

emplacement near the bottom of such formations. This potential for deep emplacement in salt domes may avoid effects of diapirism. In this application, the waste modules need not necessarily melt; a convective circulation of a liquid metal coolant may be feasible for heat transport to the container wall.

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25.2 SEABED DISPOSAL

About 72% of the surface of the Earth is covered with sea water. Most of the seabed is, for mankind, among the least accessible and least usable areas on our planet. Moreover, the oceans may cover geological formations which are more suitable for the emplacement of radioactive wastes than any site within the borders of some nations.

Seabed geologic disposal thus provides an international approach to the problem of high-level nuclear waste. Informal working arrangements presently exist among scientists of Japan, Italy, and the United States to develop this approach. At a February 1976 International Workshop, cooperative research and development activities were proposed for the Nuclear Energy Agency of the Organization for Economic Cooperation and Development.

25.2.1 Potential for Disposal Within Seabed

The potential of submarine geologic media for disposal of high-level radioactive waste is addressed here. This concept is not one of "dumping" as has been the practice with some wastes, which may or may not be detoxified in time by chemical or biological processes,⁽¹⁾ but one of controlled emplacement in a geological formation whose coordinates happen to be in the ocean instead of on land.

As in land disposal of radioactive waste, seabed emplacement requires careful assessment of the processes by which the radionuclides could migrate from the emplacement site back through containers and barriers to man's environment. Actual seabed emplacement of any quantity of such waste will occur, if ever, only after the completion of a program to demonstrate, by calculation and experiment, the retention capabilities of each of the natural and manmade barriers to migration.⁽²⁾

A number of important resources are available from the ocean, and they will undoubtedly become more important in the future. These include hydrocarbons, food, fertilizer, sand, gravel, and heavy minerals and metals.. Around the shores of almost all land masses and associated with the major surface currents are divergent zones where nutrients are especially abundant. At these locations, the ocean supports an enormous amount of plant and animal life. This biomass is the foundation for a major food industry, and aquaculture, now in development stages, promises to increase the productivity of these shallow coastal waters. However, farther from shore there are great deserts more devoid of life than the Sahara.

Hydrocarbon resources are found on the continental shelves and in other areas having great accumulations of sediments. However, in the abyssal ocean floor in the midplate gyre regions, the only known mineral resource is the now-famous manganese nodules; these are found over much of the very deep (> 5 km) ocean floor. In general, nodules that have immediate commercial interest because of their high cobalt, nickel, and copper content are limited to those fairly restricted biologically productive areas of the deep ocean floor which are covered with siliceous ooze.^(3,4)

While man is just beginning really to use the ocean it is apparent that, unless there is massive intervention, the distribution of resources in the sea will continue much as it is today. And, while the science of oceanography is still a young and developing one, our knowledge is adequate to enable us to recognize most of the geological processes that occur on the deep sea floor. We know where earthquakes and volcanos are likely to occur, where major currents occur, and where violent turbidity currents sweep across the continental margins to the abyssal plains. We know where to expect variations in temperature and chemical properties of the water. We know where to expect high biological productivity. We know too that mobile water can disperse a contaminant over great distances at rates as high as tens of centimeters per second.⁽⁵⁻⁷⁾ By observing the transport of by-products from atmospheric nuclear tests⁽⁸⁾ and by using natural traces, we have gained some indication of the routes and rates of this dispersion. In short, we know where the most stable, tranquil, remote, resource-limited areas of the oceans are located.

Accepting the need for a scientific assessment of the ocean floors as a nuclear waste repository, we pose several questions that require answers:

- What parts of the ocean floors are sensible to study, and why?
- What do we need to know about the natural processes and the effects of waste emplacement on those processes?
- What assurance can we expect that wastes will be isolated from mankind?
- What technology is necessary for such study and for such disposal?

25.2.2 The Ocean Environment

The oceans and their floors as they exist today are geologically young; the oldest oceanic rocks recovered from the deep ocean floor are only 150 million years old. In contrast, the continents are of the order of 3 to 4 billion years in age and contain exposed outcrops of ancient oceanic sediments hundreds of millions to more than a billion years in age.⁽³⁾

In the past decade, a revolution has taken place in the concept of sea-floor evolution. A concept--once called continental drift, now "plate tectonics"--says that the globe is made up of a number of crustal plates composed of oceanic or continental regions (Figure 25.9). These plates move in predictable directions and speeds and collide in seismically active regions. Plate boundaries are either areas of crustal destruction where the edges of plates are being thrust under or over other plates, or they are areas of construction where, if the earth's diameter is to remain constant, new crust must be made at a rate equal to the destruction rate. This growth takes place at the center of the midoceanic ridge (MOR), which is a globe-circling, volcanically active belt about 40,000-km long. At this "spreading center," new molten basalt is constantly being injected. (6,9)

It is within this framework of crustal unrest that we describe the principal features of the ocean environment, highlighting the characteristics that may apply to the problem of ultimate waste disposal. No attempt will be made to discuss the complex biological communities in the ocean provinces, except to call attention to the overall biological productivity of surficial waters (Figure 25.10) as an important indicator of midwater and bottom biological activity. (6,8,10)

The ocean floors are divided into three principal physiographic provinces, each occupying about a third of the world's ocean area (Figure 25.11):

- Continental margins (continental shelf, inland seas, marginal plateaus, continental slope, continental rises)
- Ocean basin floor (abyssal plains, abyssal hills, oceanic rises, deep sea trenches)
- Midoceanic (ridge flank and crest, rift valley and mountains, fracture zones).

It is helpful to remember that, with some obvious exceptions, geological processes on the continents are primarily erosional whereas in the oceans they are principally depositional.

25.2.2.1 The Continental Margins

The continental margin is one of the most dynamic ocean environments, with wide seasonal changes in water temperature, variable chemical and biological processes, and complex and unpredictable geological structures. Here lie most of the remaining unexploited pools of hydrocarbons as well as most of the world's great fishing grounds. Sediment accumulation is rapid, and this, combined with relatively steep slopes, provides optimum conditions for sediment failure (slumping, turbidity currents, etc.). These dynamic elements are enough to rule out the continental margins for not meeting our criteria of isolation and stability (see Table 25.3).

FIGURE 25.9. Schematic Diagram of the Dynamic Ocean Regimes (from Reference 6)

FIGURE 25.10. Distribution of Primary Production in the World Ocean (from Reference 6)

FIGURE 25.11. The Major Oceanic Provinces (from Reference 6)

TABLE 25.3. Comparison of the Major Oceanic Regimes (from Reference 11)

	<u>Stability</u>	<u>Resources</u>	<u>Area,</u> <u>km² x 10⁶</u>	<u>Inter-</u> <u>national</u>	<u>Accessibility</u>	<u>Sediment</u> <u>Thickness,</u> <u>km</u>	<u>Produc-</u> <u>tivity</u>	<u>Ranking</u>
Midoceanic Ridge:								2
Fracture Zone	low-high	low	small	yes	low-very low	0-2	} high to low	
Flanks	high	very low	large	yes	low-very low	0.5-1		
Crest	low	low	small	yes	medium-high	0-0.1		
Rift Valley	very low	low	small	yes	medium	0		
Continental Margin:								
Shelf	very low	very high	15-25	no	very high	2-20	high	
Slope	very low	medium	15-25	no	medium	2-20	medium	
Rise	medium	medium	20	both	low	5-10	low	
Delta and Fans	very low	high	5-10	both	medium	5-10	high	
Ocean Basin Floors:								1
Seamounts	very low	low	10	yes	very high	0-0.5	--	
Trenches	very low	very low	6	both	very low	1-5	medium	
Abyssal Hills (midplate)	high	very low	130	yes	very low	0.1-0.3	very low	
Abyssal Plains	low	medium	10-20	both	very low	0.5-2	very low	

25.2.2.2 The Ocean Basin Floor

This province, occupying another third of the ocean area, is the deepest of the three provinces (5 to 11 km) and includes the flat abyssal plains, the gently rolling abyssal hills, and the deepest parts of the oceans, the deep-sea trenches.

The abyssal plains (gradients $<1:1000$) contain numerous deposits of coarse continental debris swept here by sporadic rapid (hundreds of cm/sec) underwater avalanches or turbidity currents.

The abyssal hills, which occasionally form broad low swells (mid-ocean rises), lie seaward of the abyssal plains and are thought to be old ocean crust originally formed as extrusions of pillow basalt from the mid-ocean ridge spreading center. As the crust spread away from the MOR it cooled and sank, reaching a depth below the water surface of 5 km in about 50 million years. These vast abyssal-hill provinces (e.g., most of the North Pacific) are generally covered with 50 to 100 m of brown clay. The abyssal hills and rises that occur in the middle of subocean plates are seismically passive; where they also occur below the centers of wind-driven, surface-current gyres, they are typically quite stable and relatively unproductive biologically.⁽³⁾ Bottom currents are generally weak and variable.⁽¹¹⁾ From the viewpoint of stability, these enormous (10^8 km²) regions of the midplate/midgyre abyssal hills are unparalleled.

The dynamic deep-sea trenches often form the landward boundaries of abyssal hills near the collision zones of the subsea plates, especially in the Pacific and Indian oceans. Here the ocean crust is presumably being overridden by lighter crustal rock, with attendant crustal destruction, at rates of 2 to 6 cm per year. Many high-intensity earthquakes occur in or near these trenches, triggering massive submarine slides and accompanied by extensive volcanism. The arcuate circum-Pacific trenches are some of the most dynamic regions on the planet.⁽³⁾

25.2.2.3 The Mid-oceanic Ridge

The MOR is a sea-floor spreading center that runs around the globe and is the "constructive" plate-boundary (Figure 25.11). In the center of this symmetrically expanding ridge lies the hot, seismically active rift valley where new crust is continually being extruded. The temperature of the central portion of this rift valley may approximate that of molten basalt (1200°C), yet water circulation dissipates the heat so quickly that no rise in water temperature can be measured at the sea surface directly above this tremendous (20,000 km²) heat source. Sediment in this shallowest and youngest part of the ocean is generally too thin to be detected except as a current-winnowed veneer of carbonate shell debris seen in bottom photographs.⁽³⁾

25.2.2.4 Summary

These three provinces comprise the ocean environment that we have examined with a view toward selecting sites for the detailed studies described below. The best of them, from a purely oceanographic/geologic viewpoint, seem to be the midplate/midgyre regions and the flanks of midocean ridges (Table 25.3).⁽¹¹⁾

25.2.3 Release Pathways of Radionuclides from Submarine Geologic Disposal

To compare the rates of decay of the radionuclides against the rates of migration of the nuclides toward mankind, we have adopted a formalism based on a set of sequential barriers to release. Each of these barriers could possibly be breached either by singular events or by the slow and steady action of physical, chemical, and biological processes to dissolve the waste and transport the nuclides through the medium.

Each of these events or processes may be assigned a probability or a rate, and the overall rate of impact upon mankind will be the sum of the rates of the parallel paths of sequential events or processes less the rate of decay of the nuclides.

Conceptually, and to a great extent in the real case, the sequence of barriers can be grouped into regimes. These are shown in the flow diagram in Figure 25.12 and include the water column and, of course, the atmosphere and the surficial waters of the continents, the "benthic boundary layer" (that section of the water and sediment columns in which major chemical and physical gradients occur, i.e., from about 100 m above to about 1 m below the sediment/water interface), layers of unconsolidated to liquified sediments, and the basement rock.

Because the process of emplacement of the wastes in any one of the regimes will require passage through the upper regimes, radionuclides can enter each regime either by passage up through the barriers below, or from a mishap during the emplacement. In the absence of any mishap, however, the input of nuclides will occur from the emplaced, solidified, encapsulated waste as shown in Figure 25.12.

The methodology represented by the sequence of barriers (or processes breaching these barriers) appears to be amenable to mathematical interpretation, and such a set of models is under development. Interpretation of some of the barriers individually is possible in a crude manner, and some of the mathematical representation is now capable of giving early information based on rather restrictive assumptions. These pieces of information and some of the implications are cited later.

FIGURE 25.12. Flow Diagram (from Reference 11)

25.2.4 Status of Technology

25.2.4.1 Containment

In assessing the efficacy of the barriers, we have taken two approaches to the estimation of the times for release of the nuclides: (1) estimation from existing data of the transport times for water through the barriers, and (2) the mathematical representation of the barriers including all of the relevant processes.

From the historical data summarized elsewhere⁽¹²⁾ and new data collected from our North Pacific study sites, estimates have been made of the times for breakthrough of water which could bear nuclides for several classes of barriers (see Table 25.4). These estimates point out the need for further data in several crucial areas. Yet they appear to indicate that a combination of

a cask which can survive the time of strong thermal output, a waste form which has been designed for minimum solubility in the emplacement geology, plus the sorption capacities of the sediments could provide a sufficient barrier to the release of nuclides.

TABLE 25.4. Unsubstantiated Estimates of Barrier Efficacy

Barrier	Estimated Years to Break- through (a)	Notes
Cannister		
- metal (Zr, Mo, Ti)	$>10^3$ (b)	- for complete uniform corrosion
- ceramic	?	- manufacture difficult
Waste Form	$\geq 10^4$ (b)	- for complete dissolution at 10^{-5} g/cm ² /day
Rock		
- locally melted	?	- requires properties be measured and samples collected
- monolithic crystalline	?	
- pillow structure	$\gg 10^1$	
Sediment Column		
- at rate of pore water	$>10^6$	- neglects thermally driven transport
- with ion-exchange	?	- varies with ion; highest for transuranics
Benthic Boundary Layer	$\geq 10^2$	- varies with ion - both biologi- cal and chemical mechanisms either fix or release
Water Column		
- mid-water	$\leq 10^3$	- many questions regarding internal structure and biologi- cal mechanisms
- mixed layer (top 800 m)	10^0	- physical and biological move- ments rapid
Short Circuits/Added Protection		
- plug	?	- little experience in deep ocean
- accretion (MnO ₂ , SiO ₂)	?	- little evidence available
- thermal effects	?	- generally accelerate all processes
- biology	?	- little evidence available

a. Definition of these times is underway, and new estimates are continually being made available.

b. Depends on water circulation in medium and temperature.

Regardless of the media in which the radioactive waste is placed, a relatively insoluble waste form and a cask which will remain intact during emplacement are necessary. Glassy or ceramic materials have been produced with simulated high-level wastes which exhibit leach rates (or dissolution rates) less than 10^{-5} g/cm²/day at ambient conditions and in very pure water, and laboratory electrochemical experiments provide measurements of corrosion rates for metallic casks which extrapolate to lifetimes of about 100 years. Both of these rates are likely to be less in a medium in which rapid movement of water is precluded and in which the water is saturated with the elements that make up the cask and containers, but both will likely be accelerated in the presence of radiogenic heat. Neither the problem of the construction of a ceramic cask nor of its decomposition has been addressed, but we know that ceramic artifacts have survived millennia.

25.2.4.2 Water Column

If one defines "release" as "transport to mankind," the water column itself provides a limited dilutional barrier to release. Material balance arguments and the age of bottom water in the great basins indicate that movement of the dense water from the regions of formation in high latitudes to areas where this water is returned to the surface layers takes from 100 to 1000 years.⁽¹³⁾ Measurements of long-term, 19-month deep currents (100 m off the bottom) at a site in the North Pacific indicate a mean current velocity of 3 cm/s (see Figures 25.13 through 25.15). Overall, the internal horizontal transport and assumed vertical transport⁽¹⁴⁾ (where currents converge or diverge there is a vertical component of the current) is more rapid than originally expected.

In addition, while still in part a mystery, the biological community either in the surface waters or on the bottom may provide a path for transport either laterally or vertically. In short, the water column is likely to be a poor barrier for large quantities of nuclides but provides some protection against inadvertent release of smaller amounts such as from a single cask.

The model for the transport of the nuclides through the water column is far from defined. We merely feel certain now that the model will need to include not only simple bulk diffusion and advection coefficients, but also the effects of internal structures of the water column (e.g., eddies), major periodic motions, (e.g., tides and internal waves), biological transport either through the food chain or through diurnal and seasonal migrations, and even mechanisms driven by the activities of mankind.

FIGURE 25.13. Long-Term (9-Month) Current Meter Record at 31N-158W, Separated from Meters in Figures 25.14 and 25.15 by 30 km(11)

FIGURE 25.14. Long-Term (9-Month) Current Meter Record at 31N-158W, Separated from Meters in Figures 25.13 and 25.15 by 30 km(11)

FIGURE 25.15. Long-Term (9-Month) Current Meter Record at 31N-158W, Separated from Meters in Figures 25.13 and 25.14 by 30 km(11)

25.2.4.3 Benthic Boundary Layer Regime

There are indications that the sediment surface has special chemical properties. This is not surprising since chemical gradients exist in the top few centimeters of the sediments, most biological activity takes place in this zone, and manganese nodules are found mainly on or near the sediment surface. Preliminary conclusions from our field experiments indicate that plutonium, once collected in or on the sediment surface, passes back into the water at an unknown but rapid rate.^(8,15)

The mechanism of movement at this time is unknown but could result from chemical absorption and desorption on the sediment particles by transport of the ion-laden sediment particles, by water, or by transport of either the ion or the particles on which it is absorbed by the biological mechanisms. Instruments to measure both particle movement and radionuclide movement are being developed.

Initial experiments to characterize and estimate rates of biological mobility using baited cameras have shown that in very short times many creatures come to the bait, quickly devour it, and then move away. How they locate the food remains a mystery. At different depths, different species are encountered. In the midplate/midgyre regions we find rattail fish and amphipods,⁽³⁾ and a baited camera placed recently in a trench photographed large numbers of amphipods.⁽⁸⁾ The special chemical, biological, and physical gradients of the benthic boundary layer will be used in our model of nuclide transport.

25.2.4.4 Sediment Regime

If emplacement in the sediment is the chosen disposal method, then the canister barrier becomes important until the thermal output has been reduced to levels which produce minimal water transport (about 200 years). Initial estimates indicate that metal canisters can be built which, when kept below 100° C, will last for that same period. Laboratory and in situ corrosion studies are underway to verify this. After the thermally driven mechanism ends, the movement of the nuclides will be controlled by interstitial water movement, chemical potential, and the ion-exchange properties of the sediments.

Interstitial water moves upward through the sediments during compaction. The data we have on permeabilities of sediment as measured in the laboratory suggest that water moves, by compaction processes, toward the sediment/water interface at rates approximately equal to the rate of sediment accumulation (1 mm per thousand years or 1 m per million years). Laboratory permeability studies to date indicate that the coefficient of permeability (velocity of water for a hydraulic gradient of unity) may be less than 10^{-7} cm per second for the deeper unconsolidated sediment. The actual velocity of water within

a homogeneous sediment column, due to compaction stresses, depends on the rate of accumulated stress state (i.e., degree of consolidation). Other factors such as vertical variability in properties and cataclysmic events can have important influences on the rate of water migration. The permeability studies are being continued and will include an investigation of the effect of high pore water pressure on water mobility.

A model for radionuclide transport through soil by ground water⁽¹⁶⁾ has been modified to simulate vertical transport in the sediments. Important parameters for this model are the velocity of the water moving through the porous media, the dispersion coefficients of the flow as controlled by particle size and water velocity, and the distribution and absorption coefficients of the sediment for each ion. Some of the present problems inherent in this model are that it assumes trace concentrations, homogeneous media through which the water flows, and linear absorption isotherms, and it does not take into account the presence of competing ions.

Sensitivity studies using this model indicate that for a 333-year leach time (release rate) and a 100-meter column, a combination of a 1-meter-per-year water velocity and absorption coefficient of 10,000 or a 0.1-meter-per-year water velocity and absorption coefficient of 1000 would contain ⁹⁹Tc for 20 half-lives. For ¹³⁷Cs, a 1-meter/year water velocity and an absorption coefficient of 20 would be adequate. For the transuranics (²⁴²Pu and ²³⁰Th), a 1-meter/year water velocity and absorption coefficients of 100,000 and 50,000, respectively, would contain the isotopes for 20 half-lives; or for ²⁴²Pu with a water velocity of 0.1 meter/year and absorption coefficient of 10,000, the material would be contained for 20 half-lives.

The rates used in our sensitivity calculations are five or six orders of magnitude faster than the estimated water velocities in the sediments. In addition, assuming no greater absorption coefficients for these ions than for the respective Hanford soils⁽¹⁶⁾ (the deep ocean red clays are known to have higher sorption coefficients than most land soils for certain ions), and assuming a water flow of 0.1 meter/year (five orders of magnitude faster than expected), calculations suggest that the ²⁴²Pu would be contained in the sediment column for 3 million years (7.75 half-lives) and ²³⁰Th for 700,000 years (9.0 half-lives).

The heated cask may alter the rate of radionuclide movement through the sediment column. Using preliminary geotechnical and thermal properties of the red clay sediments and assuming either total conduction or total convection, cask temperature will reach 160° or 30° C, respectively. Even though the sediment restricts the flow, some material transport may be possible if a temperature of near 160° C is attained. Laboratory experiments are underway to determine the relative amount of conduction and convection in homogeneous red

clay heated at laboratory pressures. A representative flow model will then be constructed. Some of the mechanisms involving thermally driven instabilities and the problems of sediment disturbance during emplacement and hole closure are also being addressed.

Finally, there are some factors still to be assessed. There is the intriguing possibility that the cask can be cleverly designed to be the nucleus of some accretionary process. (Chert and manganese oxides naturally accrete in layers of nodules within or at the surface of the sediments.) Also thermal effects on all the processes are likely to alter rates, and these driving forces are just beginning to be assessed. All of these additional factors will be included in the final model.

25.2.4.5 Basement Rock Regime

The basaltic rock underlying the deep ocean sediments is the most poorly understood media because of the paucity of data and of measurements of its bulk properties. Recent developments both in the field (Deep Sea Drilling Project, project IPOD of NSF) and in the laboratory do, however, shed light on the nature of basaltic crust and on the relevant processes there.

Initial information from the Deep Sea Drilling Project, which has obtained core samples of the rock for several hundred meters below the basement rock/sediment interface, has revealed an increasing compaction of the sediments toward limestone or shale, then an interbedding of pillow lava with fine-grained pelagic sediment to a depth of hundreds of meters below the sediment/basement rock interface. This is underlaid by layers of more monolithic basalt. The deeper sections of basalt showed more differentiation of the minerals and larger crystals resulting from slower cooling. Although the bit penetrated some hundreds of meters into the basalt, the total length of the core was much less, indicating that in some areas the rock may have been weak and possibly fractured. As the bit passed through these regions, the material was pulverized and lost.

The rate of water movement through monolithic basalt beneath the deep ocean is unknown but almost certainly decreases in the deep crystalline (slowly cooled) formations. In the interbedded structure of the top few hundred meters of basalt, water probably moves more rapidly and may travel horizontally as well as vertically.

If the thermal output of wastes emplaced in the basement rock is sufficient, melting of the waste and the surrounding rock is possible. In this case, the canister need only survive the emplacement procedure. Laboratory studies of melting of a crystalline rock chemically similar to basalt (dolerite) have provided some important insight into understanding this melting phenomenon.⁽¹⁷⁾ The laboratory specimen (about 1 m³) cracked due to

thermal stresses and the molten rock/waste material flowed for a few centimeters into the cracks and solidified, resealing the openings. Calculations of thermal processes predicted the measured thermal profiles and the presence of a convection cell.⁽¹⁸⁾ Tests of the leaching properties of the resultant mixed, solidified rock/waste matrix showed rates not dissimilar to those for borosilicate glass waste forms.

25.2.4.6 Technology Status Conclusions

Of the 70% of the earth's surface covered by the oceans, the areas in the middle of the oceanic plates and the great water gyres (midplate/midgyre regions) seem to offer some promise as disposal sites. Assuming emplacement of the wastes within the sediments, it is possible to construct a methodology for the assessment of the performance of the disposal system -- a set of barriers to the release of the nuclides from the disposal site. We have established this methodology schematically and have prepared a mathematical model of the barriers. One part of this model has been tried in a crude manner and roughly confirms the earlier estimates.

Data regarding the several aspects of the seabed disposal system are being provided by ongoing studies of oceanographic parameters, effects, and engineering requirements. The first estimates from the model indicate a need for refined information regarding the physical, biological, thermal, and chemical properties of the sediments. The acquisition of this data will be the focus of the field and laboratory work for the coming period.

25.2.5 Transportation

Transportation from the fuel reprocessing plant to the disposal site would be accomplished in two stages. The first would be land transportation, the second sea transport from the embarkation point to the disposal site. Since the transportation engineering both on land and in the sea are many years ahead of the research and development activities in the Seabed Program, additional information and concepts over those presented in BNWL-1900⁽⁵⁾ are minimal. Since the disposal option has yet to be identified, the design of the ship for transportation has not begun. It is obvious, however, that in this design the following factors must be considered. The ship must be designed with adequate cooling and shielding to handle all kinds and ages of waste. The ship must be designed so that there will be minimal dock-side problems in transport of the cask from the land facility to the ship. A combination of land and sea transport (using a ship which will incorporate a total train or truck convoy) may be optimal in that it would eliminate secondary dock-side transfer.

W. B. Silker and M. R. Peterson⁽¹⁹⁾ have evaluated and rated as minimal the possibility of loss at sea of high-level radioactive waste containers during transit to the sea disposal site if adequate precautions were taken. In the event of a loss at sea, the task of locating and recovering single or multiple waste canisters is rated as very good with current levels of technology and experience.

25.2.6 Emplacement

In the matter of emplacement, as with transportation, research and development lags behind engineering. This is vividly exemplified by the fact that the ability to drill kilometers into the crystalline rock beneath the unconsolidated and liquified sediments has been demonstrated. Drill hole reentry has been achieved in the deep ocean. Accuracy of bottom positioning in 5 km of water has been quoted as less than one inch.⁽²⁰⁾ The capability exists for retrieving large objects^(19,20) and for doing detailed engineering work on the floors of the oceans 5 or more thousands of meters below the surface.

Since we do not as yet have sufficient data and understanding to specify the sediment types and emplacement depths for containment of the radionuclides for the time period required, only conceptual designs of the emplacement methodologies have been developed. Figure 25.16 shows these five subseabed emplacement concepts. Estimates of times, relative costs, and number of canisters per site are given for each concept. The free-fall penetrometer concept obviously is the least expensive in both construction and facilities cost, whereas placement of canisters only in the basalt is estimated to be the most expensive concept by a factor of 40. Holes containing canisters only in the liquified sediment and holes into the crystalline rock with canisters through the column are intermediate in cost. Recovery of the waste from any one of the five emplacement schemes is possible with existing engineering technology, but it would be costly; in our estimation, compared with other options costs of recovery (assuming the necessity of remote operation) would be very nearly equivalent. Recovery of material from the upper sediments (penetrometer emplacement) would obviously be less expensive than from basalt.

25.2.7 Conclusions

As indicated in both the Emplacement and the Transportation sections, the engineering capabilities for carrying out oceanic operations are far ahead of the research required to identify the critical emplacement parameters and breachment modes. Silker⁽¹⁹⁾ states, "... series of gigantic grapples were lowered and attached to the (Russian submarine) hull with the aid of a closed circuit television system. Just the idea of retrieving an object of this size is mind boggling. It just emphasizes the fact that, given the sufficient impetus, current technology and engineering capabilities can be scaled to accomplish the near impossible."

FIGURE 25.16. Subseabed Emplacement Concepts for Solidified High-Level Radioactive Wastes

25.2.8 Snapshot of Progress

In the continuing effort since 1973 to assess the oceans as potential sites for radioactive waste disposal, the Seabed Disposal Program participants have arrived at the following conclusions:

1. Several ocean provinces meet the criteria which have been identified as important. Research and development feasibility studies of two of these areas have begun: (a) a midplate/midgyre region in the Pacific and (b) a midocean ridge-flank/midgyre region in the Atlantic.

2. Recent current meter data covering 19 months for the bottom water of the North Pacific midplate/midgyre region shows a concentration of energy in the lower 100 meters with a net flow of 2 to 4 centimeters per second. These data indicate that even in these supposed "tranquil" ocean basin areas, water is moving with sufficient advective energy to rule out the ocean water itself as a principal barrier to the migration of long-lived waste.

3. A positive "buffering" aspect of the ocean water in the midplate/midgyre Pacific regions is evident from the fact that the various glacial cycles have had no obvious effect on the bottom environment. About 4 million years of history can be read from 10-meter-long core samples.

4. Data from these cores also reveal extreme uniformity of sedimentation during the last four million years. This record of environmental constancy from the present backwards in time is not found in many other places on earth.

5. Rock samples recovered a few hundred meters below the sediment/rock interface by the Deep Sea Drilling Project (IPOD) in semirelated areas of other ocean basins have revealed a fractured vein structure. Chemical studies of these samples suggest that fluids may percolate through this upper layer of crust. Information from the current IPOD activities, which will drill much deeper holes into the basement rock, will yield additional data on the barrier properties of the upper crust. No holes have been drilled in the North Pacific area and thus any extrapolation from holes drilled elsewhere should be done with extreme caution. No near-term studies are planned for the rock regime.

6. Preliminary calculations using an ion transport model and conservative estimates of upward water movement and sorption coefficients suggest that several tens of meters of sediment are sufficient to contain the ion for approximately ten half-lives. Next years' effort will include the verification of the assumed flow ratio of water and sorption coefficients as well as develop a better understanding of ion movement through a chemical gradient.

7. Physical property measurements on 10-meter cores obtained from the Pacific indicate that the sediments to those depths are sufficiently plastic to flow into an emplacement hole and close it in relatively short times, thereby filling the hole with material having properties very similar, if not identical, to the surrounding media. Thus, no man-made sediment bypasses should be left to short-circuit the sediment container.

8. Using existing engineering technology, we have begun a preliminary study of methodology in sediment emplacement. Four methods have been identified and their relative rates and costs compared.

9. During the engineering emplacement study it became very apparent that at the present time, the engineering capabilities of working in the deep ocean or ocean floor far surpass the current understanding of the ion transport phenomena.

10. Under the direction of the Nuclear Energy Agency (NEA), the U.S. in February 1976 was host to an International Seabed Disposal Workshop to address all aspects of seabed disposal. From this, a report will be prepared which will give recommendations to the NEA and prepare future cooperative research and development activities.

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25.3 ICE SHEET DISPOSAL

The continental ice sheets in theory could provide an international approach to the disposal of high-level nuclear waste. The technical feasibility of ice sheet disposal is intimately connected with the long-term stability of ice sheets. (We are not discussing emplacement in soil or rock beneath the ice.) Three international groups of glaciologists have recently indicated that further understanding of the evolutionary processes in ice sheets, the relationships with climate change, and the nature of future climatic changes is necessary before ice sheets could be seriously considered for nuclear waste disposal applications.

25.3.1 Descriptions of Concepts

Concepts for the disposal of high-level radioactive waste in ice sheets,⁽¹⁾ either Antarctica or Greenland, have been described in the literature.⁽²⁻⁴⁾

25.3.1.1 Meltdown or Free Flow

This disposal concept, illustrated in Figure 25.17, consists of selecting a location for disposal of each waste canister, preparing a shallow hole in the ice sheet, and lowering the canister into the hole where it is released and allowed to melt down or free flow to the bottom of the ice sheet.

Predrilling of surface holes to a depth of 50 to 100 meters would provide shielding for the operation during the initial phase of emplacement. At this depth, the snow (or firn) is transformed to ice bubbles.⁽⁵⁾ In one proposal the hole spacing would be about 1.0 kilometer, to preclude interference between individual canisters during descent or possible concentration at the ice sheet base after descent.

Each canister would melt its way down through the ice sheet to the bedrock at a rate, calculated from the penetration rates of thermal ice probes, of from 1 to 1.5 meters per day.⁽⁶⁾ Assuming an ice thickness of 3000 meters as a typical value,⁽⁷⁾ the time for meltdown to bedrock would be about 5 to 10 years.

Canister shape could be designed so that a vertical path from surface to bedrock was assured. In the design of the canister and selection of materials, consideration should be made of the pressure and the possibility of saline water at the base of the ice sheet.

Alternatives to this straightforward meltdown concept are more attractive from some points of view. If the canister were so designed, for example, that with the contained radioactive waste it had a density intermediate between those of ice and water, the time of meltdown could be very greatly

FIGURE 25.17. Ice Sheet Disposal Concepts

increased. It would approach that of the residence time of the ice particles at that point, after the canister had become thermally inert.

25.3.1.2 Anchored Emplacement

This disposal concept variation (Figure 25.17) is similar to the free-flow concept described previously except for the provision of interim retrievability. A hole is drilled in the ice sheet to a depth of 50 to 100 meters. A waste canister with attached cables 200 to 500 meters long is lowered into the hole. Meltdown begins and the descent is stopped at a depth of 200 to 500 meters by anchor plates on or near the surface, attached to the cables. The anchor cables or chains may also carry instrument leads that could be used to monitor the condition of the canister. The time for a canister to reach an anchored position, based on thermal probe rates,⁽⁶⁾ is estimated to be 6 to 18 months.

Eventually, as new snow and ice accumulated on the surface, the anchor plates would be covered at a rate of about 5 to 10 cm/year in Antarctic and 20 cm/year or more in Greenland. With 20-meter-high anchor markers, the waste canisters could be potentially retrievable for 200 to 400 years. It is estimated that about 30,000 years would be required for the system to reach bedrock at a typical site. During that time, waste and anchors would also tend to follow the ice flow pattern.

25.3.1.3 Surface Storage

A concept for waste disposal storage in a surface facility with ultimate disposal into the ice sheet is also illustrated in Figure 25.17. The facility would be a large surface storage unit supported by jack-up pilings or piers resting on load-bearing plates within the ice. Waste canisters would be placed in cubicles inside the facility, with air cooling provided by natural draft. The facility would be elevated above the ice surface for as long as possible to reduce snow drifting and to provide for heat dissipation. During this period the waste would be retrievable. Eventually, when the limit of the jack-up legs is reached, the entire facility would act as a heat source and begin to melt down through the ice sheet. It is estimated that the facility could be maintained above the ice sheet for a maximum of 400 years after construction.

25.3.2 Nature of the Disposal Medium and Areas

For several periods during the Pleistocene epoch (approximately the last 2 to 3 million years), ice sheets have covered up to 30% of the Earth's land surface. Today only the Greenland and Antarctic ice sheets remain. Together they cover about 11% of the land surface of the earth and theoretically constitute the world's largest reservoirs of fresh water (about 78% of the world's nonoceanic water) or some 2.7×10^7 cubic kilometers of water. For comparison, there is about 1.4×10^9 cubic kilometers of water in the world's oceans and seas.⁽⁸⁾

On all but the periphery of the ice sheets, snow is deposited as a powdery dry noncohesive aggregate. After deposition, intergranular bonds form between the grains, largely by sublimation, and the snow becomes firm and cohesive. As a result of the snow accumulation on the surface, a layer becomes buried deeper and deeper beneath the surface and the pressure on it steadily increases. As the snow is compressed, the density and intergranular bonding increase. In areas where no surface melting occurs, density of 0.8 g/cm^2 is eventually reached at depths of 50 to 100 meters,⁽⁹⁾ and the air spaces become sealed off. The density at which this occurs is usually taken as the transition from snow to ice.⁽¹⁰⁾ The process of transformation from snow to perennial ice occurs where more snow is deposited over a period of years than is dissipated by local wastage processes such as melting and evaporation. As the mass of ice builds up, stresses created by its weight cause it to flow. This visco-plastic flow may permit a balance between accumulation and wastage to develop, if conditions remain unchanged for sufficiently long periods. The total accumulation is then equal to the total wastage, most of which will be at lower elevations where melting may occur or ice may be lost as icebergs into surrounding waters.

The Greenland ice sheet is described, and some estimates of ice age given, in Reference 1 (sect. 5 and Appendix 5.B) and References 11 through 16. Descriptions of the Antarctic ice sheet are given in References 1 (Sect. 5), 5, and 18 through 20.

25.3.3 Motions of Continental Ice Sheets

The motion of the continental ice sheets has been treated theoretically by several investigators,⁽¹⁹⁻²³⁾ but until recently most of the measurements were of surface movement within range of fixed bedrock features. In the past few years, however, the U.S. and Australian Antarctic Programs have made use of "geoceivers," position-finding by the use of high-latitude orbiting satellites, and there are now indications that radio-echo sounding techniques can also be used.⁽²⁴⁻²⁶⁾

For the most part, the types of motion possible in the ice sheets are closely related to those in smaller glaciers, bottom sliding and differential movement in the ice, but in addition it is possible that slow convection occurs in the thicker parts of the Antarctic ice sheet.⁽²¹⁾ At present, there is no evidence to favor the suggestion that such convection does occur, but Hughes⁽²¹⁾ (in discussion following Reference 27) considers that the airborne radio-echo technique, as now developed, could fail to detect convecting pipes.

The general form of a continental ice sheet is determined largely by the accumulation pattern, the major topographic features below the ice, and dominantly by the visco-plastic properties of the ice itself. In dynamic equilibrium in regions of low subglacial relief, the ice assumes a dome-shaped form so that ice spreads radially from high altitude centers. By analogy with valley glaciers, where bottom melting conditions exist, it is believed that an ice sheet may move almost as a rigid block by sliding over the underlying land, with only a minor amount of sheet displacement in a relatively thin basal layer. In shallow ice, and especially with frozen bottom conditions, differential movement throughout the thickness is much more significant.⁽⁵⁾

In addition to these general types of ice motion are the surges and traveling waves that may occur in parts of glaciers and ice sheets as a result of controlling conditions that change with time. These conditions, for example snow fall or air temperature, may change seasonally, irregularly from year to year, or over long periods of time due to climatic changes. The changes cause disturbances that travel down a glacier from 2 to 5 times the average ice velocity.⁽²⁸⁾ Traveling waves and rapid advance of the "snout" do not necessarily depend on changing basal conditions. The response time is thought to be caused in part by the presence of a water layer of sufficient thickness at the base of the ice.⁽²¹⁾ The response time of the Antarctic ice sheet as a whole to changes is thought to be of the order of 10^3 to 10^4 years.

In some areas, parts of glaciers occasionally exhibit large and sudden increases in velocity, not obviously related to changing surface conditions. In valley glaciers, the snout may advance by several kilometers in such surges; it is also possible that major portions of the Antarctic ice sheet may periodically surge.⁽²⁹⁾ The profile along a flowline from Byrd Station to the Ross Ice Shelf suggests that this area is now recovering from a surge,⁽³⁰⁾ although the profiles may also be explained in terms of varying basal conditions.

Although the cause of surges is not fully understood, it is clear that the coupling of the ice to the bedrock in the presence of a water layer is crucial to the process.⁽³¹⁻³³⁾ Thus a possibility exists that radioactive waste at the interface, causing anomalous amounts of water, might precipitate surge-like behavior which could have global consequences.⁽³⁴⁾ Another investigator⁽⁴⁾ has considered that the thermal effects of the wastes at the base of an ice sheet could be negligible.

Recently Hughes⁽³⁵⁾ has attempted to show a close relationship between the possible surges of the West Antarctica ice sheet, the disintegration of that body, and the initiation of ice ages. He adduces some new observational support of the concept of major surges in West Antarctica, but argues that the East Antarctic ice sheet is inherently stable.

The motion of the Greenland ice sheet has been measured along the edges at a number of locations. The motions are controlled by variable local conditions and are probably not typical of the ice sheet as a whole. The surface of the interior of the ice sheet has been estimated to move at velocities of no more than tens of meters annually.⁽¹¹⁾

In places the edge of the Greenland ice sheet consists of ice streams. The ice streams are similar to mountain glaciers, but instead of being bounded by rock slopes are bounded by slower moving ice. Many of the large outlet glaciers in Greenland are relatively narrow outlets for ice streams that reach back miles into the ice sheet.

It is probable that the outflow from the West Antarctic ice sheet towards the Ross Ice Shelf is largely in the form of ice streams, confined between slower moving ice (from map produced by Scott Polar Research Institute, Cambridge, U.K., July 1972, as part of the SPRI-NSF Echo Sounding Program).

25.3.4 Meltwater at Base of Ice Sheet

Until very recently, the physical conditions at the ice bedrock interface were essentially unknown. Direct measurements have been limited to the very few bore holes that have penetrated to bedrock, and the remote-sensing techniques previously used, largely seismic-reflection work, have generally not been suitable for determining these conditions.⁽³⁶⁾

At Byrd Station (80° S, 120° W), where the only very deep ice core to bedrock in Antarctica (2175-m long) has been obtained,⁽³⁷⁾ the bottom temperature was about -1.6° C and free water was present. At Camp Century, Greenland (77° N, 63° W), a 1.375-meter-bore hole to bedrock gave a bottom temperature of -13° C.⁽¹⁴⁾ The age of the ice at the bottom of the "Byrd" hole, based on analyses of stable oxygen isotope ratios and by analogy with the Camp Century results, has been estimated at 84,000 to 96,000 years,⁽³⁸⁾ but recent studies of the microparticles in these cores⁽³⁹⁾ give an age of about 27,000 years.

With the old seismic techniques of ice measurement, some evidence had been obtained in a few places that the base of the ice sheet was wet, for example, at points along the Byrd Station strain net;⁽⁴⁰⁾ but generally little information about bottom conditions has been abstracted from these techniques.

In recent years, however, the radio-echo sounding technique, particularly the airborne technique, has been used extensively in Greenland and Antarctica.⁽⁴¹⁻⁴³⁾ Primarily the method is used for determining ice thickness, but the character of the reflection from the lower surface of the ice varies with the conditions there. Thus Oswald and Robin⁽⁴⁴⁾ obtained strong evidence of the existence of lakes underlying the central region of the East Antarctic ice sheet, including one in the "Tentative Antarctic Repository" of Zeller, Saunders and Angino.⁽³⁾ Further consideration of these experimental results leads Oswald⁽⁴⁵⁾ to suggest that basal melting occurs over a wide area in this region. Basal melting has also been inferred from the radio-echo results in the Lambert Glacier area of East Antarctica.⁽⁴²⁾

The presence of water beneath parts of the Antarctic ice sheet is now well established and may be very widespread. The full effect of the water layer on our theories of ice motion and the stability of ice sheets has not yet been considered. Melting at the base of the ice sheet had been proposed as one of the mechanisms for initiating the surges of the (East) Antarctic ice sheet, which in turn were considered to initiate the northern hemisphere glaciations.⁽⁴⁶⁾ It is reasonably established⁽⁴⁷⁾ that the dimensions of an ice sheet and its movement over the underlying material may be controlled to some extent by such a basal water layer, but it is by no means proven that basal melting does automatically produce surging.^(48,49)

25.3.5 Recent International Comments

For two examples of recent comments on the international scene regarding disposal of radioactive waste in Antarctica, see the last appendix in Volume 5 of this report. See also Reference 50, which discusses a September 25, 1974, meeting on the problem, attended by scientists from Argentina,

Australia, Japan, Norway, U.K., U.S.A., and U.S.S.R. A statement was issued summarizing the conclusions of that meeting:

1. We conclude that the Antarctic Ice Sheet is not a suitable site for the disposal of radioactive wastes that need to be isolated from the biosphere for periods of several hundred thousand years.
2. Over the last 20 years or so theoretical and observational studies of the Antarctic and Greenland Ice Sheets have allowed us to build up an understanding of the basic physics of ice sheets sufficient to go some way toward answering the glaciological questions posed by Dr. Zeller's proposal and by the Battelle Memorial Institute study.
3. Even a complete understanding of the behavior of the ice sheet with respect to the present boundary conditions, including climate, geothermal flux and sea level is insufficient to allow the precise determination of the ice sheet's future. For that we need at least a corresponding knowledge of the future changes in the boundary conditions.

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26.0 EXTRATERRESTRIAL DISPOSAL

26.0 EXTRATERRESTRIAL DISPOSAL

Extraterrestrial disposal for permanent removal of radioactive waste constituents from the Earth appears feasible. It may offer an international solution to the disposal of such waste. A fundamental requirement for space disposal is successful development of a partitioning process (see Volume 2) and of a waste container which can withstand the maximum credible launch and re-entry accident.

26.1 CONCEPT DESCRIPTION

The basic concept of extraterrestrial disposal⁽¹⁾ includes packaging waste materials in a safe manner and transporting the material by rocket or other means to a location off the Earth. Several different space trajectories have been considered. These include:

- a high-Earth orbit (of the order of 150,000 km)
- transport to the sun
- a solar orbit other than that of the Earth and planets
- solar system escape.

The potential space destinations considered are listed in Table 26.1 along with the incremental velocity (ΔV) beyond low-Earth-orbit velocity.

The two trajectories that seem most likely to assure true final disposal are:

- impact with the sun (which is not likely with present vehicles)
- escape from the solar system.

NASA studies indicate that transport of waste constituents to the sun or other extraterrestrial bodies (e.g., the moon or other planets) is technically feasible. These latter destinations have been considered only briefly because the regions are unexplored and/or of possible future scientific interest or might be of future value for alternative applications.

Vehicles that could be used include existing space vehicles and the planned space shuttle. A shuttle-tug combination is illustrated in Figure 26.1. The manned space shuttle launch vehicle with an expendable external propellant tank is placed into a low-Earth orbit, with liftoff assistance by two solid-fueled motors that are separated and dropped into the ocean, where they are recovered. After use, the expendable external propellant tank is separated from the orbiting shuttle and then de-orbited by a small retro-rocket. The tug, with the waste payload, is deployed from the payload bay of the orbiter. For some space destinations (e.g., escape from the solar system), a second tug is required to provide the final stage propulsion; thus two space shuttle launches must be made for one waste package. The orbiting shuttle returns to Earth at a prescribed landing site and would be ready for a repeat flight in about 2 weeks.

TABLE 26.1. Summary of Potential Space Destinations

Destination	Delta-V ^(a) km/sec	Advantages	Disadvantages
High-Earth orbit	4.11	Low Delta-V Launch any day Passive waste package Can be retrieved	Long-term container integrity required. Orbit lifetime not proven.
Solar orbits via:			
Single burn beyond Earth escape	3.65	Low Delta-V Launch any day Passive waste package	Long-term container integrity required. Earth re-encounter possible (may not be able to prove otherwise). Abort gap past Earth escape velocity. ^(b)
Circular solar orbit	4.11	Low Delta-V Launch any day	Long-term container integrity required. Orbit stability not proven. Abort gap past Earth escape velocity. ^(b)
Venus or Mars swingby	4.11	Low Delta-V	Long-term container integrity required. Limited launch opportunity (3 to 4 months every 19 to 24 months). Requires midcourse systems. Need space propulsion or have possibility of unplanned encounter.
Solar system escape:			
Direct	8.75	Launch any day Passive waste package Removed from solar system	High Delta-V. Abort gap past Earth escape velocity. ^(b)
Via Jupiter swingby	7.01	Removed from solar system	High Delta-V. Limited launch opportunity (2 to 3 months every 13 months). Requires midcourse systems. Abort gap past Earth escape velocity. ^(b)
Solar impact:			
Direct	24.08	Package destroyed Launch any day Passive waste package	Extremely high Delta-V. Abort gap past Earth escape velocity. ^(b)
Via Jupiter swingby	7.62	Package destroyed	High Delta-V. Limited launch opportunity (1 to 2 months every 13 months). Requires midcourse guidance systems. Abort gap past Earth escape velocity. ^(b)

a. Delta-V is the incremental velocity required to leave a low-earth orbit and is a direct indication of the size and propulsion energy of the rockets required.

b. An abort gap is a short time period wherein a controlled abort of the mission cannot be accomplished if the flight is off course.

FIGURE 26.1. Shuttle Launch Deployment Sequence for Extraterrestrial Disposal

26.1.1 Payload Considerations

Space disposal is unique in that the cost of transportation is likely to exceed \$2000/kg of payload. Thus, the weight of shielding could create a significant economic penalty. There are also unique requirements for capsule integrity to provide a reasonable degree of assurance of survival in the case of an abort.* Since the means for cooling are limited, the thermal output of the waste is a significant design factor. The weight of the waste itself is also significant.

Three different situations for space disposal were considered in a study reported in Reference 1 (page 8.3).

Case 1. Package the bulk waste as it comes from the reprocessing plant in solid form. The package can include incorporation of the waste in forms such as glasses, ceramics, or cermets.

Case 2a. Use space disposal only for the transuranium elements by removing them from nearly all fission products and

* For definitions of abort conditions, see Reference 2, pp. 21-24.

uranium and thorium in the waste. Only 1% of the fission products and 0.5% of the uranium are included with the transuranics. Transuranium elements contaminated with fission products would be in an oxide form.

Case 2b. Case 2a with 0.1% of fission products remaining with the transuranics for space disposal.

Case 3. Same as Case 2b with 99% of curium also removed from the transuranics to be disposed of in space.

Analysis of Case 1 indicated that it would be very costly, package integrity would be difficult to achieve, and the number of space flights would be excessive. Therefore, this case received less detailed consideration than Case 2.

Cases 2a, 2b, and 3 would require chemical separation (partitioning) of the constituents to be disposed of extraterrestrially from the remaining constituents, which would require separate management. In practice, the cost of additional partitioning and the additional weight which could be disposed of would be balanced to achieve optimum payload and cost.

Case 3 was considered only briefly. Curium-244 has a short half-life and generates up to 87% of the radioactivity and 93% of the heat in the LWR actinide waste. Thus, removal of the curium could reduce the shielding weight significantly; or, for the same weight package, more pure actinides could be disposed of. If the curium were removed, the curium would probably be stored on Earth for about 100 years until it decays to plutonium and other decay products. These longer-lived decay products, which have much lower heat and radioactivity output, could then be shipped to space separately.

The actual quantities which might be considered for space disposal will vary with fuel-cycle options. Some waste constituents are elements which have not been recovered because it was not economical to do so. Thus development of improved separation processes or development of new uses for radioisotopes could reduce the quantities presently considered to be waste.

Encapsulation of the waste constituents is needed to prevent release of radioactive materials in the event a flight aborts. Containment is needed during surface transportation, during all phases of flight and abort, and during some lifetime in flight. Consideration must also be given to removing decay heat and minimizing shield weight. A conceptual design for a capsule is shown in Figure 26.2.

For protection in the event of re-entry, an additional shell is added to the exterior of the waste capsule. A conceptual re-entry shield, which surrounds the waste capsule and is designed to ensure stability and minimize weight penalty, is shown in Figure 26.3.

FIGURE 26.2. Transuranic Waste Capsule for Space Disposal

FIGURE 26.3. Re-entry Shield and Transuranic Disposal Package for Solar Escape Destination

26.2 SAFETY CONSIDERATIONS

A safety analysis of extraterrestrial disposal has been performed and documented. Summary sections of this report from Reference 2, with minor editorial changes, follow.

The fundamental philosophy of nuclear safety requirements for radioactive waste disposal missions in space can be stated as follows: "Potential radiation exposure and harmful contamination of individuals, the population at large, and the (environment) shall be negligible."

This statement includes also celestial bodies as required by the Treaty to Promote Peaceful Exploration and Use of Outer Space which 60 nations, including the United States, have signed.

For operations during all phases of a nuclear waste disposal mission, the exposure and contamination values should be negligible for the population at large and within the permissible standards for the personnel involved in the mission. Those criteria governing radiation exposure to humans during ground handling and transportation which may be applicable can be found either in 10 CFR, Parts 20 and 71, or in References 3 and 4.

The nuclear waste disposal missions discussed here are similar to other space shuttle/space trip missions carrying radioactive materials to synchronous Earth orbit or to planetary orbits. The primary difference lies in the comparatively large amount of radioactive material containing actinides being transported during the waste disposal missions.

The purpose of the analysis⁽²⁾ was to evaluate the response of the nuclear waste package to the potential accidents that could occur during the various phases of waste disposal missions. It was possible to perform only a qualitative evaluation because the mission hardware and systems and the mission proper are in a preliminary state of definition.

Within the framework of their exploratory study, NASA concluded that nuclear waste disposal missions appear feasible. The waste package design concept with the various protective shells provides a means for preventing release of radioactive waste constituents in most hypothetical accidents. Additional testing would be required to confirm the concept.

Safety studies on accident models and package responses point out certain key issues to maintain overall nuclear safety for space disposal of nuclear waste packages. These key issues (Reference 1, pp. 813 and 851) are:

1. The waste package should be designed to maintain integrity throughout all waste handling, launch, and potential abort events without releasing significant amounts of its radioactive contents.
2. The external radiation of the package, whether intact or damaged, has to be held to a minimum so that recovery can be accomplished without

undue exposure to the population. This would require development of location and retrieval means for Earth-impacted waste packages.

3. Potential accident conditions that could lead to uncontrolled re-entry of the waste package have to be minimized. This would be accomplished by careful selection of trajectories, by use of highly reliable vehicles, and by developing space retrieval means (vehicle) for retrieval of these packages from accidental orbits beyond Earth escape.
4. The necessary lifetime of the capsule during flight depends largely on the destination. If the destination is direct solar impact, then a containment lifetime of only a few months and a shielding lifetime only long enough to prevent potential exposure to man may be all that is necessary. If the destination is a storage type of orbit, then a capsule lifetime of hundreds or thousands of years may be needed. If the destination is solar escape, a lifetime of only tens of years is required before the material is removed from our solar system. Thus, capsule lifetime needs can vary greatly, depending on the circumstances. At present, capsule design assuming a lifetime of a few hundred years can be achieved.

26.3 STATUS OF TECHNOLOGY

To use space disposal for transuranic waste appears feasible with current technology. This technology is considered to include the space shuttle and the space tug, which are advanced vehicles but which will use existing engineering technology.

Consideration has also been given to several different types of space flight. Concepts considered include: using the energy in the waste propulsion, solar sails, nuclear and ion propulsion, and acceleration of waste particles electrically from an orbiting platform. The first three of these appear to have potential for increasing the payload or providing increased propulsion velocity capability. It is possible that disposal into the sun or solar system escape could become more practical with development of such concepts, but they require additional study to evaluate them fairly and uniformly. Therefore, while such further study is warranted, it was concluded that the present analysis using state-of-the-art vehicles provides a base point for evaluation of feasibility.

Key considerations in the technical feasibility of extraterrestrial disposal are:

- providing a reliable space flight system to assure that the waste will reach and remain at its proper destination
- providing a high-integrity capsule to assure no failure during required times in space and to assure minimum spread of radionuclides in event of an abort.

Overall, extraterrestrial disposal has the potential for permanent removal of radioactive waste constituents from the Earth, depending largely on partitioning capabilities. The earliest likely date for routine disposal in space, about 15 years from now, is controlled mainly by the schedule for developing and achieving reliable operation of the space shuttle system.

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27.0 TRANSMUTATION

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The objective of any radioactive waste transmutation process is to accelerate the conversion of nuclides having long-term risk potential to shorter-lived nuclides with a lesser risk potential. This would be achieved by bombarding the nuclei with either photons or other subatomic particles.

Four principal types of transmutation devices can be considered for use in transmuting the high-risk-potential waste constituents:

1. charged particle accelerators
2. nuclear explosive devices
3. fission reactors
4. fusion reactors.

Transmutation processes using each of these four transmutation devices have been examined by using the following four criteria to evaluate the relative technical merits and the technical feasibility of each process.⁽¹⁾

- Criterion 1. Overall energy balance: For a radioactive waste transmutation process to be practical, it must consume much less energy than was originally obtained when the radioactive waste was created.
- Criterion 2. Overall waste balance: For a radioactive waste transmutation process to be practical, it must remove more long-term risk-potential radioactive waste than it creates.
- Criterion 3. Specific transmutation rate: To be considered practical, the transmutation rate should be many times that of the natural decay rate.
- Criterion 4. Total transmutation rate: For a radioactive waste transmutation process to be practical, it must be capable of eliminating a significant fraction of the long-lived, potentially hazardous radioactive waste constituents. Additionally, an adequate number of transmutation devices must be available over the proposed history of the process for it to be practical.

The proposed transmutation processes require that the radioactive waste be partitioned and/or fractionated prior to transmutation. Partitioning (see Section 16 for a detailed discussion) is defined as separation of the actinide elements from the fission products in high level waste. Fractionation is the

process of separating relatively pure individual (fission product, activation product, or actinide) elements from the mixtures that appear in spent fuels. Thus, the practicality of transmutation processing is wholly dependent upon the successful development and implementation of a partitioning or fractionation process to perform the required chemical separations.

The following sections briefly summarize the technology status and developmental requirements for light nuclide (activation and fission products) and actinide transmutation in the four different transmutation devices. The light nuclides have been treated separately from the actinides, because they have physical and nuclear characteristics that differ from those of the actinides; in addition, the concept of actinide transmutation has been studied in greater detail than has fission product transmutation.

27.1 LIGHT-NUCLIDE TRANSMUTATION

The light nuclides most often considered as candidates for transmutation are ^3H , ^{14}C , ^{85}Kr , ^{90}Sr , ^{93}Zr , ^{99}Tc , ^{129}I , and ^{137}Cs . The rationale for each of these nuclides being transmutation candidates varies, but at least one of the following two reasons applies to each of the nuclides:

1. The nuclide contributes substantially to the thermal energy production (^{90}Sr , ^{137}Cs) and/or the potential radioactive hazard (^3H , ^{14}C , ^{85}Kr , ^{90}Sr , ^{137}Cs) for the first several hundred years of waste decay.
2. The nuclide contributes substantially to the potential radioactive hazard for decay times greater than 1000 years (^{14}C , ^{93}Zr , ^{99}Tc , ^{129}I).

The rest of the light nuclides have been rejected as transmutation candidates, principally because of their short half-lives or low concentration in the waste. Nuclides which decay to low levels within a hundred years are expected to be manageable by isolation.

To be considered available for transmutation processing, the candidate light nuclides must be quantitatively separated (i.e., fractionated) from the waste streams in which they occur. Since the light nuclides have widely varying chemistry, separate fractionation processes would be required for each nuclide. Some of the candidate light nuclides are combined with significant amounts of stable isotopes which compete for the available transmutation particles, thus reducing the candidate nuclide transmutation rate. This competition is important enough in some cases (e.g., ^{85}Kr , ^{137}Cs) to possibly require isotopic enrichment of the candidate nuclide.

The four types of transmutation devices listed previously can be grouped into two categories according to whether or not transmutation is induced by neutrons. Non-neutron-induced transmutation concepts include charged particle

acceleration, coulomb excitation, and photon transmutation. Neutron-induced transmutation devices include proton accelerators producing spallation neutrons,^(2,3) thermonuclear explosives,⁽¹⁾ fission reactors,⁽⁴⁻⁶⁾ and fusion reactors.⁽⁷⁻⁹⁾

27.1.1 Non-neutron-Induced Transmutation

All of the non-neutron-induced light nuclide transmutation concepts have been evaluated to be impractical.⁽¹⁾ Scoping calculations indicate that the non-neutron transmutation devices consume a large fraction of the power that was generated in producing the candidate nuclides.

27.1.2 Neutron-Induced Transmutation

When evaluating neutron-induced light nuclide transmutation, the principal consideration is the degree to which the decay rate is accelerated by the neutron flux-neutron spectrum combination supplied by each of the transmutation neutron sources. The neutron flux levels or ranges typical of each neutron-induced transmutation device are: spallation neutron, $>10^{17}$ neutrons $\text{cm}^{-2} \text{sec}^{-1}$; thermal fission reactor (PWR), 3×10^{13} neutrons $\text{cm}^{-2} \text{sec}^{-1}$; fast fission reactor (LMFBR), 5×10^{15} neutrons $\text{cm}^{-2} \text{sec}^{-1}$; fusion reactor, 10^{15} to 5×10^{16} neutrons $\text{cm}^{-2} \text{sec}^{-1}$. The intermittent neutron flux present in a thermonuclear explosion is not directly comparable with that present in the other continuous neutron sources, and will be discussed separately in a later portion of this section.

The neutron spectra of all of the continuous neutron sources, except in the thermal fission reactor, are "hard" (i.e., high energy). LMFBRs have average neutron energies of several MeV. Both spallation neutron devices and fusion reactors can be expected to supply neutrons with average energies greater than 1.0 MeV. However, all of the hard-spectrum neutron sources, except possibly the LMFBR, should lend themselves readily to neutron thermalization in heavy water or beryllium blankets; this should result in thermal flux levels equal to or slightly greater than the original fast neutron flux levels.

A method for evaluating light-nuclide transmutation alternatives is to calculate the decay acceleration resulting from irradiation in two different continuous neutron flux-neutron spectra combinations. The results of this calculation for the eight candidate nuclides mentioned previously are summarized in Tables 27.1 and 27.2. The transmutation cross section was taken to be the sum of all neutron-induced reactions resulting in the production of another nuclide [e.g., (n, γ) , $(n, 2n)$, (n, α)] in a thermal (PWR) neutron spectrum and a fast (LMFBR) neutron spectrum.

TABLE 27.1. Summary of Effective Decay Times for Neutron-Induced Transmutation of ^3H , ^{14}C , ^{85}Kr , and ^{93}Zr

	^3H		^{14}C		^{85}Kr		^{93}Zr	
Radioactive half-life, years	12.33		5730		10.73		9.5×10^5	
Neutron spectrum	<u>PWR</u>	<u>LMFBR</u>	<u>PWR</u>	<u>LMFBR</u>	<u>PWR</u>	<u>LMFBR</u>	<u>PWR</u>	<u>LMFBR</u>
Effective neutron-activation cross section, barns	10^{-6}	10^{-6} (estd)	10^{-6}	10^{-6} (estd)	1.5	0.0087	10.0	0.0394
Time required to eliminate 99.9% of nuclide, years								
Decay only	123	123	57100	57100	107	107	9,470,000	9,470,000
$\phi = 10^{13(a)}$	123	123	57100	57100	106	107	2190	525,000
$\phi = 10^{14}$	123	123	57100	57100	99.6	106	219	55,200
$\phi = 10^{15}$	123	123	57100	57100	61.7	103	21.9	5560
$\phi = 10^{16}$	123	123	57000	57000	12.8	75.0	2.19	556
$\phi = 10^{17}$	123	123	57700	55700	1.44	20.4	0.22	55.6
$\phi = 10^{18}$	123	123	45300	45300	0.15	2.46	0.02	5.56

a. Neutrons $\text{cm}^{-2} \text{sec}^{-1}$

TABLE 27.2. Summary of Effective Decay Times for Neutron-Induced Transmutation of ^{90}Sr , ^{137}Cs , ^{99}Tc , and ^{129}I

	^{90}Cs		^{137}Cs		^{99}Tc		^{129}I	
Radioactive half-life, years	29.0		30.1		2.13×10^5		1.59×10^7	
Neutron spectrum	<u>PWR</u>	<u>LMFBR</u>	<u>PWR</u>	<u>LMFBR</u>	<u>PWR</u>	<u>LMFBR</u>	<u>PWR</u>	<u>LMFBR</u>
Effective neutron-activation cross section, barns	1.23	0.000158	0.17	0.039	44.5	0.20	34.5	0.24
Time required to eliminate 99.9% of nuclide, years								
Decay only	289	289	300	300	2,120,000	2,120,000	1.6×10^8	1.6×10^8
$\phi = 10^{13}(\text{a})$	284	289	299	300	491	102,000	635	90,000
$\phi = 10^{14}$	249	289	293	298	49.1	10,700	63.5	9010
$\phi = 10^{15}$	110	289	243	285	4.91	1070	6.35	901
$\phi = 10^{16}$	16.8	288	90.3	195	0.49	107	0.63	90.1
$\phi = 10^{17}$	1.77	283	12.4	47.0	0.05	10.7	0.06	9.01
$\phi = 10^{18}$	0.18	289	1.29	5.47	0.005	1.07	0.006	0.90

a. Neutrons $\text{cm}^{-2} \text{sec}^{-1}$

The results show that:

- Neutron-induced transmutation of ^3H and ^{14}C is not practical because of the extremely small cross sections of these nuclides.
- Neutron-induced transmutation of ^{85}Kr , ^{90}Sr , and ^{137}Cs becomes practical only at thermal flux levels greater than 10^{16} neutrons $\text{cm}^{-2} \text{sec}^{-1}$, or fast flux levels greater than 10^{17} neutrons $\text{cm}^{-2} \text{sec}^{-1}$.
- A significant reduction in the effective half-lives of ^{93}Zr , ^{99}Tc , and ^{129}I can be achieved in thermal neutron flux levels typical of modern PWRs ($\sim 3 \times 10^{13}$ neutrons $\text{cm}^{-2} \text{sec}^{-1}$). However, a thermal flux level of 10^{14} to 10^{15} neutrons $\text{cm}^{-2} \text{sec}^{-1}$ is desirable for ^{93}Zr , ^{99}Tc and ^{129}I to reduce the irradiation time to a few decades.

The values given in Tables 27.1 and 27.2 represent the "best case" results. Allowances have not been made for either time spent out of the transmutation device for reprocessing and refabrication, or for neutron absorption by other isotopes of the elements (e.g., ^{88}Sr , ^{133}Cs) which are being transmuted.

The present availability of technology associated with each of the transmutation neutron sources varies widely. PWR and BWR thermal fission reactors are commercialized and available. Small experimental fast-fission reactors have been built (EBR-II, Fermi) and LMFBRs are projected to become commercialized within the next decade. The concept of a proton-induced spallation-accelerator neutron source is theoretically feasible;^(2,3) however, the engineering considerations involved, and the expense of transmutation in a device producing relatively little or no usable power, would appear to make transmutation in other devices more attractive. The fusion reactor concept has not been proven practical on even a laboratory scale.

27.1.3 Nuclear Explosive Transmutation

Nuclear explosive devices have been proposed as transmutation devices because of the large neutron yield resulting from their detonation.⁽¹⁾ The concept involves placing the waste to be transmuted in a 1.5-km hole, and then subjecting the waste to multiple 100-kiloton thermonuclear device detonations. The principal considerations involved in nuclear explosive device transmutation follow.

- This concept can be considered only for waste nuclides with half-lives much greater than the 12.3-year half-life of tritium, because of the significant tritium residue production in the nuclear device. The principal candidate nuclides are ^{99}Tc , ^{135}Cs , and ^{129}I .

- As a result of neutron capture in stable cesium and iodine isotopes, only the nuclear explosive transmutation of isotopically separated ^{135}Cs and ^{129}I is considered feasible.
- The implementation of an underground nuclear test ban treaty would eliminate any consideration of this concept.
- It has been estimated that eleven 100-kiloton explosions would be required annually to transmute the ^{99}Tc produced by one 1000-MW(e) PWR. Transmutation of the long-lived components of the elemental cesium and iodine produced by a 1000-MW(e) PWR would require about twenty-six 100-kiloton explosions annually.

There are fundamental disagreements with regard to the light-nuclide transmutation rate in a thermonuclear explosion. The arguments are detailed in Reference 1.

27.1.4 Status and Developmental Requirements of Light-Nuclide Transmutation Technology

For those nuclides considered to be transmutable with thermal fission reactors (^{93}Zr , ^{99}Tc , ^{129}I), the principal requirement remains one of developing methods for quantitatively separating these nuclides from the shorter-lived waste constituents.

In the case of ^{85}Kr , ^{90}Sr , and ^{137}Cs , a transmutation device capable of neutron flux levels greater than 10^{16} neutrons $\text{cm}^{-2} \text{sec}^{-1}$ must be developed, along with the appropriate fractionation processes. Isotopic enrichment processes may also be required for ^{85}Kr and ^{137}Cs to reduce the number of neutrons lost to other krypton and cesium isotopes.

The transmutation of light nuclides other than those mentioned above is considered to be infeasible because they either decay to innocuous levels within a few hundred years; have transmutation cross sections which are too small to allow for a significant reduction in the effective half-life; or are present in the waste in small amounts.

27.2 ACTINIDE TRANSMUTATION

Reducing the risk potential via transmutation is not straightforward for actinide nuclides (atomic number greater than 88). A nonfission transmutation event in a long-lived actinide nuclide generally results in the production of another actinide (directly or via the decay of short-lived intermediate nuclides) which has a half-life and risk potential comparable to the parent nuclide. Thus, fission, which results in the production of shorter-lived fission products, is the only way to substantially reduce the risk potential of the actinides.

27.2.1 Actinide Transmutation in Accelerators

As was the case for fission product transmutation, there are four principal accelerator transmutation concepts:⁽¹⁾

1. direct bombardment of the actinides by charged particles having energies of tens to hundreds of MeV
2. acceleration of the beta decay process by coulomb excitation
3. photon transmutation using either electron bremsstrahlung or stimulated gamma emission
4. spallation neutron-induced transmutation.

Beta-decay acceleration using coulomb excitation is infeasible since nearly all of the long-lived actinides of interest are alpha emitters. Direct bombardment and photon excitation are also infeasible; scoping calculations indicate that neither of these concepts meet the fundamental criterion of consuming less power than was generated in creating the actinides to be transmuted.⁽¹⁾ Actinide transmutation in a spallation neutron accelerator apparently has not been explicitly proposed. However, the intense neutron flux available in projected spallation accelerators, combined with the relatively large actinide cross sections, indicate that this process is technically feasible, assuming the existence of a spallation accelerator. The spallation neutron actinide transmutation concept does not appear to be as attractive as fission or fusion reactor transmutation because of the projected high transmutation cost of spallation neutrons. This high cost would result principally because the major portion of the spallation accelerator cost must be written off against the waste transmutation operation, while the major portion of fission or fusion reactor costs are charged against power production.

27.2.2 Actinide Transmutation by Nuclear Explosives

The transmutation of actinides using a thermonuclear explosive as a transmutation neutron source has been evaluated as being technically feasible.⁽¹⁾ Some pertinent conclusions concerning actinide transmutation using nuclear explosives follow.

- It is estimated that 3.5 100-kiloton thermonuclear detonations will be required annually to transmute (fission) the neptunium plus americium plus curium produced in a 1000 MW(e) PWR.⁽¹⁾
- The fission products resulting from the actinide transmutation will necessarily remain in place and substantially untransmuted, since the 3.5 detonations per year required for actinide transmutation is

much less than the 11 to 26 detonations per year estimated to be required for transmuting the long-lived fission products.

27.2.3 Actinide Transmutation in Fission Reactors

Of all the transmutation concepts for reducing the potential hazard of radionuclides in waste, that of transmuting actinides in fission reactors is by far the most heavily investigated.^(1,4,5,10-16) The actinides are better candidates for fission reactor transmutation than the lighter nuclides, because actinide neutron cross sections are typically 10 to 1000 times larger than light-element neutron absorption cross sections. These larger cross sections result in the actinides transmuting at a significantly higher rate than the light nuclides, thus minimizing both the effective actinide half-life and the unfissioned actinide inventory which must be recycled. Since the neutron capture cross section of most of the actinides is comparable to or larger than the fission cross section, several successive neutron captures may be required to quantitatively convert a particular actinide to fission products.

Several different parameters involving the feasibility and effects of actinide transmutation in fission reactors have been investigated. The most important parameters investigated are:

- the mass and composition of the actinides being recycled
- the rate at which the recycled actinides are fissioned in the various types of fission reactors
- the effect of the recycled actinides on fission reactor criticality and reactivity
- the effect of the recycled actinides on the out-of-reactor nuclear fuel cycle (i.e., fabrication, shipping, reprocessing).

Results of the investigations in these areas are discussed in the following paragraphs.

A PWR producing 1000 MWyr(e) per year [1250 MW(e) @ 80% capacity factor] creates approximately 22 kg of high-level waste actinides per year which might be recycled.^(1,11) This 22 kg of actinides from the PWR consists principally of neptunium (approximately 70%), along with smaller amounts of americium (approximately 23%), curium (approximately 6%), and plutonium (approximately 1%); the exact composition depends on the elemental recoveries attained during reprocessing. Inclusion of only 0.1% of the uranium present in the discharged fuel with the recycled actinides would increase the recycled waste actinide mass by about 34 kg.

Sustained recycle of the 22 kg of actinides, along with annual 22-kg additions to the recycle actinide stream, results in an increase in the mass of the actinides being recycled from the 22 kg initially present, to 44 kg at equilibrium if the transmutation reactor is a PWR. A schematic diagram depicting this equilibrium mode of actinide recycle, along with an annual actinide mass balance, is shown in Figure 27.1. To put this in perspective, the mass of recycled actinides at equilibrium would comprise 0.12% to 0.15% of the total heavy metal in the PWR core in the case cited above.

The calculated rate at which the 40 to 50 kg of equilibrium recycled actinides mentioned previously would be transmuted (i.e., fissioned) ranges from about 3% per year to 9% per year.⁽¹¹⁾ A typical actinide transmutation rate is about 6% per year for each year the actinides are in the transmutation reactor. The principal variables affecting the transmutation rate are the neutron flux level, the neutron-flux energy spectrum, time spent out of the transmutation reactor, and the physical arrangement of the actinides (lumped vs homogeneously dispersed) in the transmutation reactor.^(1,4,5,11,15,16)

The neutronic effect of the recycled actinides on transmutation reactors has been investigated by four groups.^(1,4,5,11) The most commonly used measure of the neutronic effects is the infinite neutron multiplication factor (k_{∞}), which is defined as the neutron production rate divided by the neutron destruction rate for a reactor of infinite size. There are two facets to these neutronic considerations:

1. the effect of the recycled actinides on the transmutation reactor
2. the criticality of the recycled actinides alone.

The recycled actinides are detrimental to the neutron economy of the thermal transmutation reactors. The average (middle-of-cycle) decrease in k_{∞} in a 3.3%-enriched PWR transmutation reactor is about 1%, as compared to the same reactor without recycled actinides.^(1,4,5) It has been estimated that the enrichment of the PWR would have to be increased from 3.30% to 3.43% to make up for the recycled actinides if they were homogeneously distributed in the PWR.⁽¹⁾ If the recycled actinides are concentrated into every tenth fuel rod, the initial enrichment must be increased to about 3.47%.⁽¹⁾ The same recycled actinides in an LMFBR are not as detrimental to the neutron economy as in the PWR. Preliminary calculations indicate that the recycled actinides may contribute positively to the LMFBR k_{∞} for a significant portion of an irradiation cycle.

	① FRESH URANIUM FUEL (kg)	② FUEL CHARGE (kg)	③ FUEL DISCHARGE, MA ^a (kg)	④ ECONOMIC ACTINIDE RECOVERY, U+Pu (kg)	⑤ HIGH-LEVEL WASTE, MA ^a (kg)	⑥ ACTINIDE RECYCLE, MA ^a (kg)
ACTINIDES RESULTING ANNUALLY FROM URANIUM FUEL	34,190	34,190 U	32,993	32,959	33.5	22
ACTINIDES RESULTING ANNUALLY FROM ACTINIDE RECYCLE	0	44 MA ^a	44	13	0.5	22
TOTAL ANNUAL ACTINIDES FROM ALL SOURCES	34,190	34,234 MA	33,028	32,972	34	44

^aMA = MIXED ACTINIDES (SOME COMBINATION OF U, Np, Pu, Am, Cm, Bk, and Cf)

FIGURE 27.1. Schematic Diagram of Equilibrium PWR Actinide Recycle and Annual Mass Balance

Within a given equilibrium irradiation cycle, the neutronic behavior of the recycled actinides is somewhat atypical of other fissile materials. In both thermal and fast reactors, the k_{∞} of the recycled actinides alone increases with exposure. This increase in recycled actinide reactivity is due principally to the burnout of nonfissile species (e.g., ^{237}Np , ^{241}Am , and ^{243}Am), and not to the buildup of fissile nuclides.

The characteristic of the recycled actinides which dominates the impact of actinide transmutation on the nuclear fuel cycle is the neutron activity of the recycled actinides. Assuming a homogeneous distribution of recycled actinides in thermal transmutation reactor fuel, neutron activities of 10^{12} to 10^{14} neutrons second^{-1} tonne^{-1} of heavy metal (HM) can be expected for fresh and spent fuel.^{4,11} As a basis for comparison, the neutron activity of 1 tonne of plutonium-enriched PWR fuel at discharge is about 10^9 neutrons sec^{-1} tonne^{-1} HM. By a factor of 1000, the predominant neutron source in all thermal transmutation reactors results from the spontaneous fission of ^{252}Cf . The effects of this high neutron activity are felt principally in three sectors of the nuclear fuel cycle: fuel fabrication, fresh-fuel shipping, and spent-fuel shipping. The details of the expected effects for a neutron activity of 6×10^{13} neutrons sec^{-1} tonne^{-1} HM are as follows:⁽¹¹⁾

1. Fabrication: The thickness of limonite concrete shielding in a plutonium-enriched PWR fuel fabrication facility must be increased from the presently required 38 cm to an estimated 120 cm, a three-fold increase.
2. Fresh-fuel shipping: A fresh-fuel shipping cask containing four PWR fuel assemblies (1.75 tonnes HM) with a proportional amount of the recycled actinides would weigh about 81 tonnes and have a diameter of about 244 cm (8 ft).
3. Spent-fuel shipping: A spent-fuel shipping cask would be quite similar to the fresh-fuel shipping cask described above, since the neutron dose rate of the spent fuel was found to be larger than the gamma-ray dose rate.

The transmutation of actinides in a LMFBR results in greatly reduced recycled actinide neutron activity, a typical value being 7×10^{10} neutrons sec^{-1} tonne^{-1} HM. In this case, ^{242}Cm , ^{244}Cm , and ^{252}Cf are roughly equal neutron sources.

27.2.4 Actinide Transmutation in Fusion Reactors

The extremely high neutron flux levels (10^{15} to 10^{16} neutrons cm^{-2} sec^{-1}) projected for fusion reactors (CTRs) clearly indicate that actinide transmutation in fusion reactors would probably be technically feasible, subject to

limitations imposed by radiation damage and heat transfer.⁽¹⁾ However, the relatively primitive state of knowledge and technology concerning large CTRs indicates that any consideration of actinide transmutation in CTRs should be held in abeyance pending further research and development.

27.2.5 Status and Developmental Requirements of Actinide Transmutation Technology

The technology status of the actinide transmutation concepts are summarized in Table 27.1. Generally, the developmental status of the actinide transmutation concepts is a function of the developmental status of the transmutation device being considered. The effect of the actinide partitioning developmental status in evaluating these concepts was not considered (see Section 16.0). All of the proposed actinide transmutation concepts do, however, require the successful implementation of actinide partitioning.

TABLE 27.1. Developmental Status of Selected Actinide Transmutation Concepts

<u>Actinide Transmutation Concept</u>	<u>Developmental Status of Transmutation Concept^(a)</u>
Spallation neutron	Long-range development
Thermonuclear explosives	Short-range development
Fission reactor	Demonstration required
Fusion reactor	Long-range development

a. Independent of the status of actinide partitioning

Aside from the requirement of developing actinide partitioning, there are other developmental requirements which must be met for each concept. One requirement for all of the concepts is good neutron cross-section data for all of the actinides with a half-life greater than about 1 day. A summary of the present neutron cross-section status and the measurements required for many of the actinides are given in References 17-19. Studies similar to those begun for fusion transmutation reactors regarding the effect of actinide recycle on the in-reactor and out-of-reactor nuclear fuel cycle, the pros and cons of transmuting actinides from a risk viewpoint, and optimal methods for transmuting the actinides must be extended in the case of fission transmutation reactors. These methods must be initiated for the spallation neutron, thermonuclear explosive, and fusion reactor transmutation concepts.

Finally, both the spallation neutron accelerator and fusion reactor require considerable research and development before actinide transmutation can be seriously considered in these devices.

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