THE CAPABILITY FOR DISPOSING
OF HIGH-LEVEL WASTES SAFELY

In the Matter of	)	
Proposed Rulemaking on The Storage and Disposal of Nuclear Waste	)	PR-50,51 (44 FR61372)
(Waste Confidence Rulemaking)	)	

Prepared for:

UTILITY NUCLEAR WASTE MANAGEMENT GROUP

and

EDISON ELECTRIC INSTITUTE

(Document 2 of 4)

7 July 1980

NUCLEAR SAFETY ASSOCIATES, INC
5101 River Road
Bethesda, Maryland 20016

ACKNOWLEDGEMENT

The following organizations contributed to the preparation of this report.

American Nuclear Energy Council
Edison Electric Institute
Exxon Corporation
J L Smith Company
Lowenstein, Newman, Reis, Axelrad and Toll
Science Applications, Inc
Westinghouse Electric Corporation

Their participation in the preparation of the report is gratefully acknowledged, as is the effort of the group organized to review the draft material.

TABLE OF CONTENTS

Table of Contents (cont)

Table of Contents (cont)

Table of Contents (cont)

THE CAPABILITY FOR DISPOSING
OF HIGH-LEVEL WASTE SAFELY

Preface

This report was prepared for the Utility Nuclear Waste Management Group (UNWMG) and the Edison Electric Institute (EEI) as part of a detailed analysis of current technical and scientific information relating to the safety of waste management. It is part of the UNWMG-EEI Statement of Position in the Nuclear Regulatory Commission's proposed rulemaking on the storage and disposal of nuclear waste. This report focuses on the state of the technology presently available for geologic disposal of nuclear waste. A Summary Document and two other reports have also been prepared for purposes of the UNWMG-EEI presentation. The reports are entitled:

- "Long-term Safety of Nuclear Waste Disposal: A Basis for Confidence" (the "Basis for Confidence Document")
- "Interim Storage of Spent Fuel" (the "Storage Capability Document")

This report, which is sometimes referred to as the "Disposal Capability Document," first presents our views on the basic requirements of a waste disposal system. Next, it considers the alternatives for an ultimate disposal system and presents the rationale for a reference system (deep geologic disposal) selected for further discussion. All of the components of a total waste management system are then reviewed to consider the status of technology, potential alternatives, the existence of international activities and experience, and our assessment of each element. Schedule and cost are also discussed. Finally, we present our overall conclusions as to the degree of confidence in ultimate disposal.

Within this document the reader will note the development of two important principles. The first involves basic waste disposal requirements, while the second concerns the importance of utilizing a systems approach to achieve waste containment.

An acceptable nuclear waste management system must protect the public from undue risk resulting from excessive exposure to radiation, both now and in the future. Perspective as to potential risk from waste management can be gained by considering the existing levels of radiation exposure due to naturally occurring environmental sources.

The report reviews natural sources of radiation exposure and their variations as a starting point in developing guidance and perspective for the discussion of an acceptable level of risk. Naturally occurring radiation originates from cosmic rays, radionuclides produced by them, and radionuclides contained in the earth and materials derived therefrom. Natural external radiation dose to large groups in the US population vary up to about 100 mrem per year as a function of geographic location alone without any evidence of adverse health effects. Few people in the United States would consider their personal background radiation exposure as a factor in deciding where to locate and reside, thus demonstrating that an additional exposure of 100 or so mrem causes them no concern.

Accordingly, and consistent with the position of the Department of Energy (DOE), the report suggests that a goal of limiting dose increases resulting from waste disposal operations to a level which falls within a small fraction of the normal variations in natural background is totally appropriate. The report also notes that man is already subject to risks as a result of natural ore bodies contained in the earth's crust. Materials such as selenium, uranium, barium, and arsenic all occur in nature and present a hazard due to their presence in deposits which are in fairly close proximity to man. In particular, natural uranium ore--which is the raw material for reactor fuel from which high-level waste ultimately results--is itself a source of radiation exposure. The report develops a new approach for comparing nuclear waste to a naturally occurring uranium ore body; one which takes into account all of the pathways by which man could ingest radioactive materials entering the acces- sible environment.

Within this context, the second important principle underlying the report emerges, that is, the importance of utilizing a systems approach in the development of an overall containment strategy. Many barriers are available for waste containment in a repository. In addition to the natural barriers provided by a carefully selected site, they include engineered barriers such as waste forms, canisters, overpacks, and backfills. Each of these natural and engineered barriers is considered in detail in the report with respect to the status of technology, ongoing Federal programs, and activities and experiences on the international level. These barriers all act collectively to provide a given level of performance; it is, of course, this sum total of performance which is significant. Accordingly, the individual repository elements are considered in terms of their roles in providing overall system containment at least comparable to that of a natural ore deposit.

The analyses contained in the report reveal that, for a waste repository, the retention requirement is never more than a few hundred times that of an ore body; and that, for the most part (ie, after about 500 years) it is actually less. By then addressing the attributes of a repository system as compared to an ore body, the report concludes that since the combination of elements can be adjusted to furnish any reasonable overall level of containment desired, there is, indeed, a high level of confidence that the total repository system can provide the appropriate degree of retention.

I BASIC REQUIREMENTS OF A WASTE DISPOSAL SYSTEM

In these proceedings the Nuclear Regulatory Commission (NRC or the Commission) must decide whether it continues to have confidence that nuclear waste materials can and will be disposed of safely. In its Statement of Position[1] filed April 15, 1980 the DOE takes the position that:

1) spent nuclear fuel from licensed facilities ultimately can be disposed of safely off-site.
2) disposal facilities will be in operation between 1997 and 2006, and the initial increment of off-site storage facilities can be in operation by 1983, and
3) spent nuclear fuel from licensed facilities can be stored safely either on-site or off-site until disposed of ultimately.

UNWMG-EEI agrees with this position; it is our further view that the nation can, if it chooses to do so, safely proceed to ultimate disposal of nuclear waste in a salt formation on a schedule more expeditious than that proposed by DOE.

Before beginning our consideration of the capability to dispose of high-level waste in an acceptably safe manner, it is necessary to briefly discuss the definition of high-level waste. The Presiding Officer has ruled[2] that the waste form to be considered herein is spent fuel. It is our position that the fuel values in spent fuel are simply too great to be thrown away, and sooner or later the present "deferral" of reprocessing will be lifted and the spent fuel will be reprocessed. As we indicated in our memorandum on this matter[3] we therefore would prefer to consider high-level waste from reprocessing (hereafter HLW) the primary potential waste form, with spent fuel being considered only to the extent it differs from solidified HLW. However, as the DOE has already noted,[4] it is clear that for the purposes of this proceeding it is only necessary for the Commission to find that there is reasonable assurance that nuclear waste materials in <u>some</u> form can be safely disposed of by any single method. In light of the Presiding Officer's ruling, we have used spent fuel as the "base case" for our consideration. We will, however, note throughout our presentation similarities and differences between the two waste forms.

A WHAT IS MEANT BY "DISPOSING OF WASTES SAFELY?"

In reaching any determinations in this proceeding, it is first important to focus upon what is meant by "disposing of wastes safely."

Neither law nor common sense requires NRC to assure "absolute" safety; "absolute" safety does not exist in any activity of man. Nor is NRC required to find and select the best of all possible solutions. What NRC must do is find that a solution (or solutions) exists whereby nuclear waste can be disposed of with reasonable assurance that the health and safety of the public will be protected. Considering that some of the waste materials will outlast the present generation,* we believe that:

1) there should be a reasonably high degree of confidence in the "reasonable assurance," and
2) future generations should not be subjected to higher degrees of risk from the disposal of these wastes than the present generation is prepared to accept.

In disposing of nuclear wastes, it is important to recognize that there are two distinct aspects of the protection provided by the selected disposal system:

1) protection of the general public through protection of its supplies of air and water, and
2) protection of individuals from harm to themselves due to their intrusion into the waste repository.

These aspects are quite different. The failure to adequately distinguish between them leads to great confusion and to a lack of adequate perspective, particularly in relation to alleged "uncertainties" in the data base, in our degree of understanding of geohydrology, and in our predictive capability for geophysical phenomena.

---

* Note that this aspect is not unique to nuclear wastes.

The first of these aspects (protection of the general public) is the primary and perhaps only goal of waste disposal. Attainment of this primary goal is initially dominated by the fission product content of the waste, more specifically by the Sr-90 and Cs-137 content. If spent fuel is the waste form, the Sr-90/Cs-137 content typically results in a hazard index based on MPC for drinking water about 100 times that of the significant actinides.[5] For HLW the Sr/Cs requirement on this same basis is about 1000 times that of the significant actinides.[6] (When an index based on current ingestion limits is used, the fission product domination is by about a factor of 100 in either case--see extended discussion at pp I-14ff, infra.) This means that the demands on the system brought about by the fission product content for the relatively short (by geologic standards) period that they are controlling are about 100 times more stringent than those required for the long-lived content. It also means that by the time the long-lived content becomes controlling (in about 300-500 years)[9] the total hazard potential of the system will have dropped by this same factor of 100. At this point its capability of contaminating water or air is comparable to that of the natural ore body whence the fuel which produced the waste came.

The second aspect, that of protecting individuals from harming themselves by intruding into the waste repository, is a very different matter. Because of the concentrated form of the waste at the disposal site, protecting the individual intruder is a longer-term problem. However, it is not apparent this should be a "goal" of the system at all. It is questionable for deliberate intrusions. Further, for any intrusions, deliberate or inadvertant, exposure to the concentrated waste form would be limited to a small number of individuals, not to the general public. It is recognized that an intrusion may result in a pathway to the biosphere to some extent; but since the risk analyses of repositories assume massive breaches by natural forces, the possible effects of

---

* Depending on the hazard index used for reference, after 300-500 years the ratio of the total hazard potential of a spent fuel or HLW repository to that of an ore body varies from a few times times the ore body to less than the ore body. The exact value is not important. The significant point is that after a relatively short time (300-500 years) the total hazard potential of either spent fuel or HLW falls in the general range of the ore bodies whence they came.

intrusional breaches certainly are bounded.*

In order to more clearly understand the ramifications of the "short-term" versus "long-term" hazard potential on the techniques available to attain the waste disposal goal, two terms need to be defined. These two terms are containment and isolation,** which we define as follows:

Containment--keeping the waste within the confines of its place of interment to the degree necessary to prevent significant leakage to the biosphere which results in harm to the general public

Isolation--emplacing the waste in a place or manner that humans are not likely to intrude and come into contact with the concentrated waste form.

Containment is achieved through a repository system which takes into account all transport mechanisms to members of the public, including natural geohydrologic transport and geologic events (such as earthquakes) and pathways to the general public introduced by intrusional scenarios. In a deep-geologic repository, the factors that provide containment are the form of the waste (eg, spent fuel), the waste container, other engineered barriers, and the capability of the formation in which the waste container is placed to impede the movement of radionuclides away from the waste-emplacement location. Isolation is achieved through maximizing the inaccessability of the concentrated waste. These concepts are different. For example, spent fuel in a completely insoluble container would be perfectly contained, but such material placed in Grand Central Station would be completely non-isolated.

---

* See discussion of this point at pp 2-31 through 2-35 of our companion document, "Long-term Safety of Nuclear Waste Disposal: A Basis for Confidence," prepared by TASC.

** Isolation and containment have been used interchangeably throughout most waste disposal literature. However, DOE in its Statement of Position (at p I-15) does distinguish between the two terms. DOE's definition of "isolation" is somewhat different from that used here. It encompasses our definition but in addition adds the "containment" provided by all components of the waste repository system from the waste package itself to the accessible biosphere.

Emplacement in deep geologic formations provides a high degree of both containment and isolation. By providing both, it also creates the possibility for much confusion. Thus, it is important to keep in mind which of the two properties (containment or isolation) is being sought or being discussed.

The primary goal of waste disposal (protection of the general public) requires containment, it does not require isolation. Nor does it require absolute containment. What is required is sufficient containment to keep radioisotope concentrations to innocuous levels should any of the material reach the accessible biosphere.

Analyses presented in the DOE Draft EIS on Management of Commercially Generated Radioactive Waste[8] (hereafter DEIS) and other studies[9,10] show that given appropriate site selection and system design, it is highly unlikely (if not impossible) for geohydrologic or geologic processes to result in any release from a deep-geologic repository during the first several hundred years. Furthermore, the NRC Staff is now considering[11] that the waste form/canister/other engineered barriers should be capable of providing a large share (if not all) of the required containment during the first thousand years.* It is overall system performance (including all components) which provides the desired degree of protection, and in our view it is not required that any single component provide a predetermined proportion of this protection. However, it is well within current engineering and materials capability to provide a waste form/canister/engineered system (in fact numerous combinations thereof are possible) which will give added assurance that this "package" will provide substantial containment over the first few centuries. This should be true whether the waste form is spent fuel or HLW.

The Commission should have no difficulty in finding with a high degree of confidence that the wastes can be adequately contained so as to fully protect the health and safety of the public over the first millenium. Many authors[12,13] have shown that during the first millenium the hazard potential

---

* We believe 1000 years is unduly long for characterization of the period when fission products are dominant. The real change occurs at 300-500 years, but in the temporal sense of this discussion there is really not much difference between that period and 1000 years.

(hazard index, potential for contaminating water) of the solidified HLW becomes comparable to that of the equivalent ore body and, as discussed above, this is also true for spent fuel. This comparison has been attacked[14] as too simplistic, but it is a completely justified and useful comparison of the required degree of <u>containment</u> of the waste. There should be no misgiving about placing in a repository a quantity of contaminant which after a relatively short period of time becomes roughly equivalent to that which has existed in close proximity to man throughout his entire existence. This equivalent contaminant (ore) has existed in finely divided form--frequently directly in potable ground water--sufficiently close to the surface of the earth that man has to have been exposed to the effects of this quantity of contaminant over his entire development as a species.

By contrast spent fuel exists in a highly insoluble form,* it would be encapsulated in high-integrity containers, possibly surrounded by selected overpacks and other engineered features, and buried deep underground in places carefully selected to minimize or eliminate the possibility that water might come in contact with it.

Man has clearly withstood for ages past the impact of this quantity of contaminant in its available form located near the surface of the earth. We strongly suggest that no process exists which could cause the deeply and carefully buried waste to cause harm to man.

Admittedly, there is one aspect of spent fuel and its potential effects on man for which the comparison with the original ore body is less favorable. This is the fact that the spent fuel is considerably more concentrated than the ore body. Although 500-years' decay is sufficient to reduce the quantity of radioactive contaminants in spent fuel to only a few times that of the original ore body, the spent fuel is more concentrated than the ore by about a factor of 2500. This increased concentration does not affect the potential hazard of the spent fuel <u>to the public</u> (since this is a function of quantity, not concentration), but the effect <u>upon an individual</u> intruding into the waste could be more serious than that occasioned by his intrusion into the natural ore body--by

---

* HLW can be made into a form with leach-resistant properties as low as, or perhaps lower than, spent fuel.[15]

about a factor of 10,000.* However, to protect the individual intruder requires isolation of the waste--minimizing the possibility that the intrusion will occur--not containment. This role of concentration of the waste is illustrated by again considering the ore body. The ore bodies in general are very poorly isolated in a spatial sense. In fact they are quite available to man and he has frequently worked in them in a very intimate fashion. In contrast disposed spent fuel elements will be very well isolated spatially. Relatively few holes are drilled to a depth of 2000 feet** in relation to the number drilled or dug to the depth of most ore bodies. And the number of 2000-feet deep holes which would intersect a 1-foot diameter cylinder, 10 feet long, at that depth must be very few indeed. Thus the chance that intrusion will occur is small; but if it does occur the impact on the intruder will be considerably higher than that of intruding into uranium ore, due specifically to the higher concentration of the waste.

We are dealing here with a paradox. The concentration of very long-lived materials which will decay very slowly can best be decreased if containment is less than absolute and the material is permitted to disperse slowly, thus reducing its concentration. Similarly, the low leachability of the spent fuel and the very long-term behavior of water in the burial regime may have something to do with isolation of the waste materials. But to the extent that they do, they do so in precisely the opposite sense to that which is generally suggested. After the initial period of containment to permit decay of Sr-90, from the standpoint of any impact on the intruder it actually would be better if some movement (dilution) of the activity were to take place, for the purpose of reducing the waste concentration. In this regard the current thinking that the mined-out volume may be backfilled with overpack substances designed to absorb the materials which may eventually come out of the initial waste/ canister is a step in the right direction insofar as isolation is concerned. At the present

---

* HLW is about 15,000 times as concentrated as the ore after 500 years.

** It should be noted that the fuel values associated with spent fuel greatly increase the possibility that intrusion will occur to recover those values. However, in such a case the intruder would presumably know what he was doing and would take reasonable precautions.

time it is believed that spent fuel will be emplaced at about 60 to 70 canisters per acre. If the radioactive materials associated with each canister eventually were absorbed in a volume equivalent to the initial emplacement area and a vertical height of 10 meters, the concentration of the waste would be reduced by a factor of about 1000. The resulting concentration would then be less than ten times that of the initial ore body--say a very high grade ore. Thus after <u>containment</u> of the waste for 300-500 years and movement of the activity over a distance of about 10 meters, the spent fuel would be no more hazardous than a high-grade uranium ore body. And the repository would not be near the surface of the earth, it would be at a depth of perhaps 2000 feet in a carefully selected place.

In summary, a distinction should be made between <u>containment</u> and <u>isolation</u> and the system parameters and requirements associated with each. <u>Containment</u> is associated with protection of the general public, the quantity of the hazardous material remaining at the time of interest, and the transport mechanisms to the public which necessarily involve time and dilution or dispersion. These concerns are relatively short-term and, as such, are amenable to enhanced assurance of mitigation through engineered solutions. <u>Isolation</u> is associated with protection of intruding individuals, the concentration of the hazardous material at the repository at the time of intrusion, and the probability that intrusion will occur. This is a longer-term but secondary matter for which site selection is of importance.* Engineered barriers are not a direct mitigating measure for lack of isolation, but may help to reduce the probability of intrusion into the waste.

This important distinction between the requirements for <u>containment</u> and <u>isolation</u>, a distinction which is seldom grasped in the interminable arguments about geologic disposal, leads to the following conclusions:

1) A very high degree of containment (provided by the entire system) is required for only 300-500 years.
2) Therefore, long-term uncertainty in geologic stability and geohydrologic uncertainties (the so-called "gaps" and "uncertainties") are

---

* Site selection can reduce intrusion probability by locating the site distant from potential resources: important minerals, water, etc.

not that important. Put another way, our capability to predict geologic stability and hydrologic phenomena is greatest for the relatively short period of relatively higher risk.

3) For the longer term (after 300-500 years) a geologic repository presents a risk to the general public roughly comparable to that of a high grade ore body. Although higher concentrations might cause a greater risk to an intruder, his protection is obviously a secondary goal of the system, if at all. Moreover, for such protection, it would be isolation that would be needed; and the "uncertainties" discussed by earth scientists have precisely nothing to do with providing isolation --except in a negative sense. That is, if the waste materials do move a few meters a little faster than predicted, the concentration will be more quickly reduced as will be the need for isolation.

B WHAT IS A REASONABLE DEGREE OF RISK?

Another important aspect of what constitutes disposal of waste "safely" is the degree of risk arising from potential radiation exposure to which present and future generations should reasonably be exposed. It is recognized that it is the responsibility of EPA, not NRC, to establish standards that reflect an acceptable degree of risk. The present lack of EPA criteria and standards clearly complicates the tasks of both the DOE and NRC. Absent the EPA criteria and standards, it is in order to consider in these proceedings what degree of risk ought to be acceptable to this and future generations.

There are rational ways to approach the level of risk of radiation exposure which ought to be considered "acceptable". The potential for exposure of the public from any source should be considered in relation to exposures from other sources, particularly from naturally existing ones.[15] The relative benefits of providing protection against radiation from a specific source may also be judged in perspective with variations in personal exposures from background sources.

Therefore the radiation protection criteria relevant to the degree of risk considered acceptable for the disposal of radioactive waste should be compatible with current or revised standards for protection against radiation. They should also retain perspective in relation to the risk of health effects from natural environmental exposures and variations therein. A review of natural sources of radiation exposure and their variability can provide guidance and perspective for selecting the level of risk which ought to be acceptable.

Naturally occurring radiation in the environment originates from cosmic rays, radionuclides produced by cosmic rays, and from primordial radionuclides in the earth. In the United States, the air dose rates due to cosmic rays vary from about 27 to 95 mrad/year and the population weighted mean air dose rate has been estimated to be about 29 mrad/year.[16] Outdoor exposure rates due to terrestrial radioactivity vary from about 12 to 90 mrad/year in the United States and the population weighted mean is about 44 mrad/year.[17] People are also exposed to natural radiation by internally deposited radionuclides that have been inhaled or ingested. Typical dose equivalent rates to a representative US resident from naturally occurring sources are provided in Table I-1.* The estimated dose rate from cosmic rays includes a 10-percent reduction to account for structural shielding. Irradiation by primordial terrestrial radioactivity includes a 20-percent reduction for shielding by housing and a 20-percent reduction for self-shielding by the body. The dose to the lung from inhaled radioactivity is tabulated separately; doses to other tissues from inhaled radioactivity are included with other primordial radionuclides in the body. Estimated doses to the gastrointestinal tract do not include any contribution from radioactivity in the contents of the tract.

Individuals and population subgroups within the United States experience dose equivalent rates which vary widely from the average. Large segments of the population experience appreciably different natural exposures. External terrestrial dose equivalent rates experienced by people indoors range from about 15 mrem per year on the Atlantic and Gulf Coast plain to 30 mrem per year in midcontinent and to about 55 mrem per year along the Colorado Plateau.[18] Cosmic ray dose equivalent rates experienced by the population range from

---

* Due to the number and length of the tables included in this Section, all tables have been assembled at the conclusion of Section I. Figures are included at appropriate places in the text.

26 mrem in Florida to about 50 mrem per year in Denver, Colorado.[19] Thus natural external radiation doses to large groups in the US population vary from about 41 to 105 mrem per year by geographic location alone. Most of the variation in dose due to radionuclides in the body results from variations in Ra-226 in drinking water and in the lung from Rn-222 progeny.

A typical US resident receives about 16 mrem per year bone dose and about 0.2 mrem per year to soft tissues from ingested radium.[20] Even if the radium concentration in community water supplies does not exceed the current regulatory limit of 5 pCi/l,[21] dose rates experienced by users could be as much as 1 mrem per year to soft body tissue and about 150 mrem per year to bone from drinking water alone.

An average dose to the lung of about 100 mrem per year from radon progeny inhaled in dwellings has been attributed to construction materials, type of construction, ventilation, and radioactivity in land beneath.[22] Variations among these factors cause lung dose equivalents to vary by a few hundred millirem per year between specific population groups in the United States.

Throughout history, mankind has been exposed to roughly these levels of naturally occurring radiation. In some other parts of the world exposures have been even higher. For instance, along the southeastern coast of India where thorium-bearing monazite sand is abundant, external exposures to residents from natural radioactivity in the sandy soil range up to about 2000 mrad per year. In some towns along the coast of Brazil, external exposures from monazite sand in the soil average 550 mrem per year within a range from 90 to 2800 mrem annually. Even at these dose rates, effects on man's health or development have not been detected.[23]

Differences in personal living habits also affect exposure to naturally occurring radiation and radioactive material. Within a given locality the construction materials and style of buildings influence indoor radiation exposure rates. Gamma exposure rates in masonry[24] and in slab-on-grade dwellings[25] have been observed to be about the same as in natural outdoor areas surrounding the dwellings. But in wood frame dwellings, gamma exposure rates ranging from 70 to 80 percent of outdoor levels and averaging closer to 70 percent have been observed.[26] Yeates[27] measured gamma radiation intensity reduction of about 25 percent on the first floor and about 50 percent on the second floor of wood frame houses. Overall, the difference in direct radiation exposure to persons living and working in wood frame buildings rather than in concrete or masonry buildings can be about 10 mrem per year.

Lung exposure to radon and its progeny may also be affected significantly by living habits. For instance, concentrations of Pb-210 and Po-210, both long-lived progeny of Rn-222, have been found to be about 2 or 3 times higher in the lungs and ribs of cigaret smokers than in nonsmokers.[28,29] This amounts to 6 to 9 mrem per year additional dose to the lungs of a smoker, a 5- to 7-percent increase in total lung dose by comparison with a typical nonsmoker. If natural gas is used in a residence, lung doses will typically increase by about 2 percent, ie, about 2 mrem per year, as a result.[30] Perhaps the greatest influence on naturally occurring lung exposures is ventilation practices in dwellings which alone can influence radon and radon progeny concentrations in a house by a factor of as much as ten.[31] A recent review of variations in background radon concentrations can be found in Nuclear Safety.[32]

Another activity which affects exposure to naturally occurring radiation is air travel. At jet cruising altitudes of about 22,000 to 39,000 feet, the dose equivalent rate due to cosmic rays is about 0.3 to 0.5 mrem per hour. Based on this, the US population averaged dose equivalent from commercial air travel is about 1 mrem annually. However, if a person travels more frequently, eg, two trips per month of 4 hours duration each, his annual dose equivalent will be increased by about 30 to 50 mrem.

Where one vacations may also influence his annual radiation exposure. Spending a week skiing in Colorado will deliver about 1.5 mrem more than a week at an Atlantic beach, for instance.

These examples illustrate the effects personal living habits can have upon exposure to naturally occurring radiation and radioactive material. The range and variation in human exposure to natural radiation as a consequence of living habits and locations are illustrated in Figure I-1.

Population exposure rates from naturally occurring radiation and radioactive materials provides a rational basis for considering appropriate standards for disposal of radioactive material. It should be remembered that the doses from natural radioactivity and, to a large extent, from medical uses,

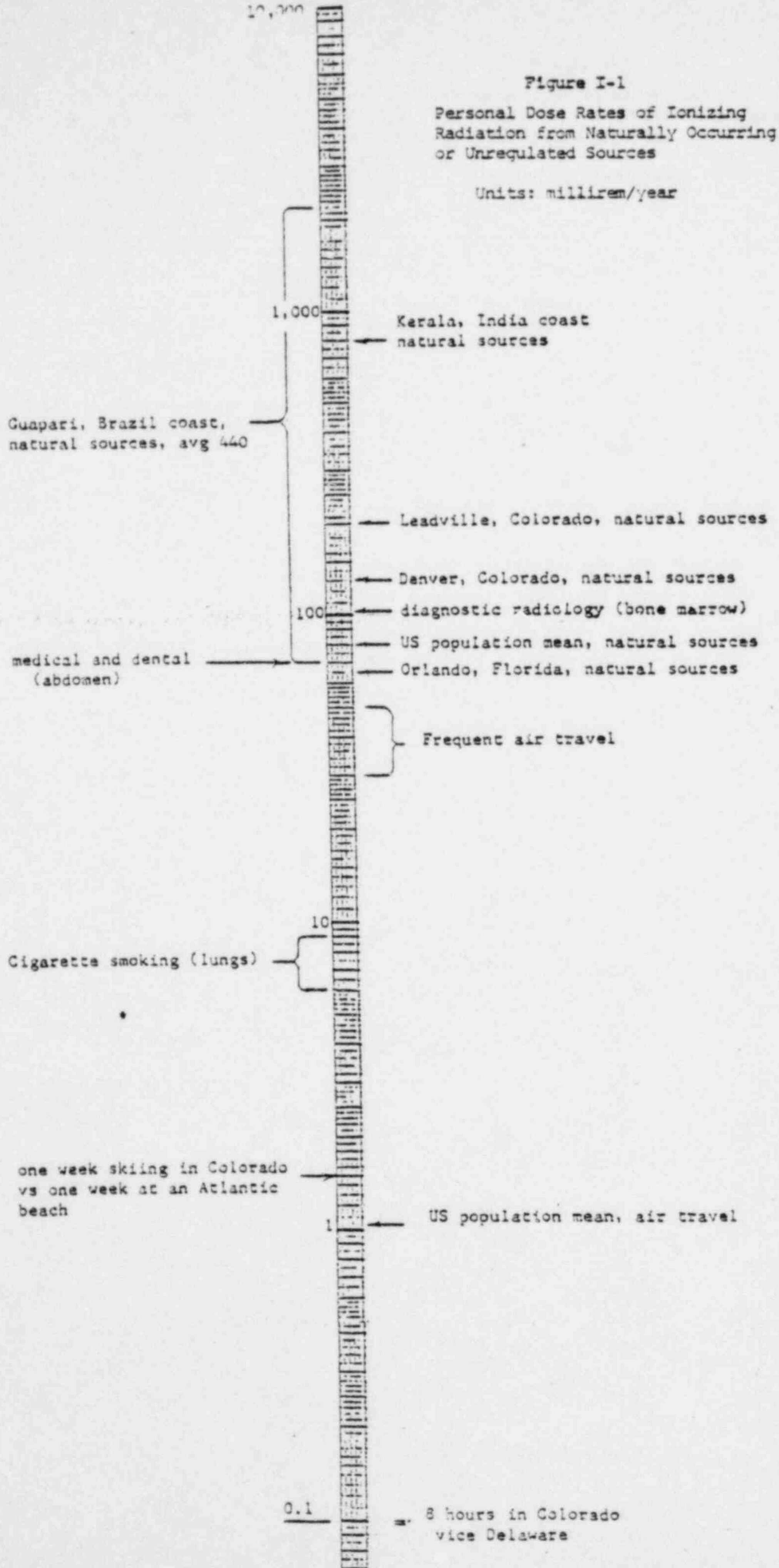


Figure I-1

Personal Dose Rates of Ionizing Radiation from Naturally Occurring or Unregulated Sources

Units: millirem/year

affect the entire US population. Conversely, the number of people who would be exposed to concentrations of radioactivity from waste disposal will be small.

In addition to the naturally occurring radiation dose levels discussed above, there are other largely unregulated exposures to which man is exposed. In fact the largest fraction of the man-made radiation exposure to residents of the United States is now administered in the practice of medicine and dentistry. Public exposure from this source exceeds that from other man-made sources by 10 to 100 times. Medical diagnostic radiology accounts for at least 90 percent of the man-made radiation exposure. Estimates based on the 1970 X-ray Exposure Study were that the per capita mean active bone marrow dose to adults in 1970 was 103 mrem from all diagnostic radiology procedures.[33] The mean abdominal dose per capita in the US population in 1970 from medical and dental radiation was estimated to be 72 mrem.[34] Medically administered radiation doses vary widely among individuals, of course. These estimates of localized doses are roughly equal to the average total body dose a US resident receives annually from natural radiation sources.

Clearly, most persons in the United States do not base their decisions on activities and locale so as to minimize their personal background radiation. Thus they demonstrate that 100 mrem/year is a dose increase which causes them no worry.

We suggest that a goal of limiting potential dose increases due to waste disposal operations to a level which falls within a few percent of natural background would be totally appropriate. In this we agree with the objective of DOE as stated in its Statement of Position[35] that risk of incremental exposures of the general population to a few percent of the normal variation in natural background radiation due to releases from a repository system, should they occur, would appear reasonably low.

C WHAT IS THE RELATIONSHIP OF SPENT FUEL TO AN ORE BODY?

One of the traditional ways of assessing the hazard of nuclear waste has been to construct a "hazard index" and to compare it to the equivalent index for a typical natural uranium ore body. Several "hazard indices" have been used, perhaps the most common one that based on permissible drinking water standards.[36] To construct this index the quantity of each isotope in a unit amount (generally the equivalent of 1 tonne of heavy metal in fuel) of spent fuel, HLW, or natural ore is divided by the maximum permissible concentration in drinking water for that isotope, and the quotients for all isotopes are summed. That is:

$$HI = \sum_i \frac{Q_i}{MPC_i}$$

where:

HI = Hazard Index, cc/tonne

Q_i = Quantity of Isotope i, C_i/tonne

MPC_i = Maximum Permissible Conc, C_i/cc

An example of such a calculation is shown in Figure I-2.[37] When done in this way, the hazard index for spent fuel remains above that for the natural ore body for about 10,000 years whereas that for HLW drops below the ore in about 300-500 years. However, the index for spent fuel drops to only about 10 times that of the ore in the same 300-500 years. After this period, during which the dominance of the fission products (particularly Sr-90 and Cs-137) disappears, further reductions occur only very slowly, for either spent fuel or HLW, since the index is then dominated by much longer lived isotopes.

Note that the results obtained by this approach are dependent only on the quantities of isotopes involved, not on the concentration thereof. This is quite proper when considering the protection of public water supplies. The capacity of the material (be it spent fuel, HLW, or ore) to contaminate water is indeed dependent only on the quantity, not the concentration. Furthermore, as shown earlier (page I-8), after the radioactive materials in a repository have moved only a few tens of feet from the initial package the concentrations will become similar to that of ore bodies. Thus any effect of concentration gradient in moving the radioactive materials back to man will create conditions in the repository similar to the ore body by the time such materials reach the outer edges of the mined cavities.

Now we would like to suggest a new approach to considering the relationship of nuclear waste to the ore body, one which takes into account all of the

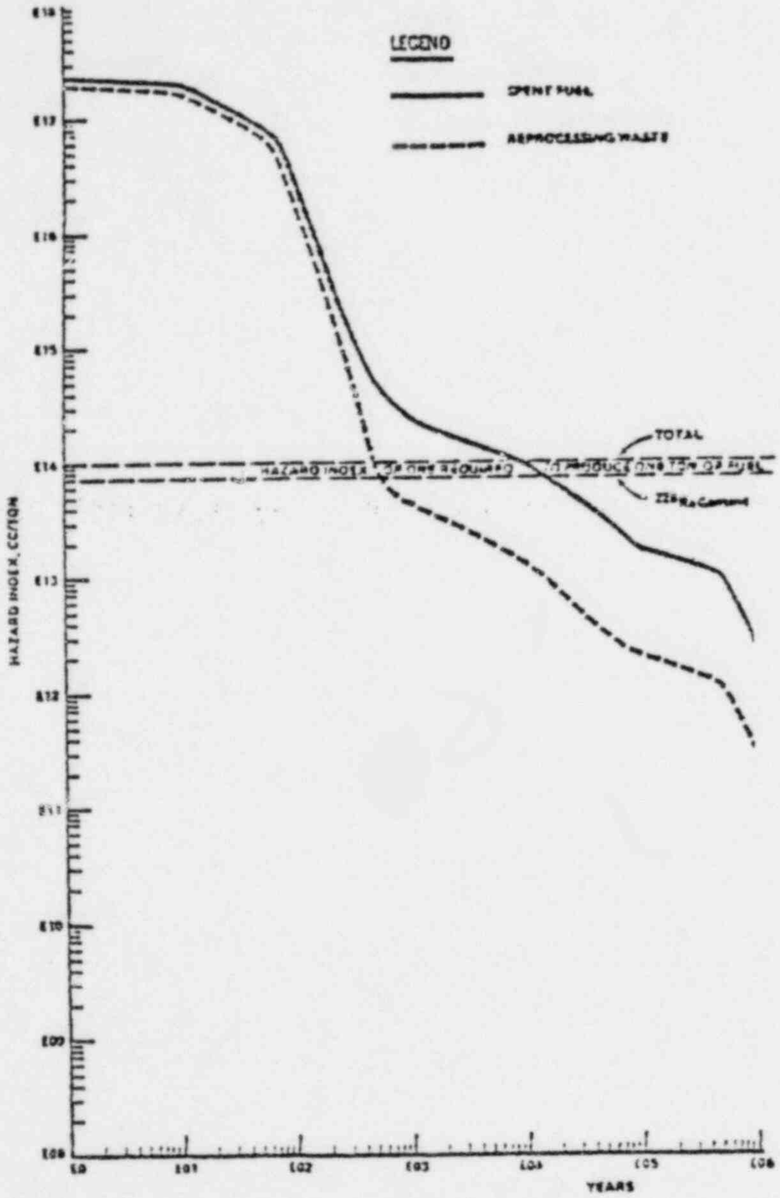


Figure I-2 Comparison of Hazard Indices of Reprocessing Waste and Spent Fuel Elements[38] pathways by which man could ingest radioactive materials which might return to the accessible environment. We suggest it is possible to determine, with a high degree of accuracy, the inventory of radioisotopes as a function of time in a quantity of buried waste resulting from the production of a specified quantity of energy. It is likewise easy to determine, with a similarly high degree of accuracy, the inventory of radioisotopes in the ore body required to produce that same specified quantity of energy. This inventory does not change significantly with time.

Likewise, if one selects a value which should be acceptable as an annual exposure dose, one can also, with a high degree of accuracy, determine the quantity of each isotope which must be ingested by an individual, by all available pathways, in order to produce the selected limiting dose.

Then, if one takes the ratio of the inventory of each isotope--in either waste or ore--to the ingestion required to produce the selected limiting dose, one can derive the factor of containment required to assure that the selected dose is not exceeded in man. We call this ratio the Retention Quotient or RQ. As we discuss below, it can be calculated with very little uncertainty for either waste or ore.

Table I-2 shows the number of curies of each isotope of importance that will produce in an individual receptor a dose of 5 mrem/year.* These values have been developed by multiplying the adult ingestion dose factors from Regulatory Guide 1.109[38] by the organ weighting factors from ICRP-26.[39] Thus these values represent the quantity of each individual isotope required to produce organ-weighted total body dose of 5 mrem/year in an adult.

The isotopic inventories in spent fuel from 1E+04 GWe-year of power production are shown in Table I-3. These inventories are taken from the DEIS.[40] The inventories of these same isotopes in HLW are given in Table I-4.

---

* We will use 5 mrem/year for illustrative purposes. It is a few percent of natural background radiation and is in the range suggested by DOE. All calculations contained herein are linear with respect to the assumed dose and thus can be readily scaled to any other dose limit.

A computer program[41] was developed to produce as a function of time the Retention Quotients for spent fuel, HLW, or ore for any selected dose to an individual. The Retention Quotient is equal to:

$$RQ = \sum_i \frac{Q_i}{DF_i}$$

where:

RQ = reciprocal of the fraction of the total inventory which must reach a receptor (man) in order to give that receptor the annual dose limit selected

Q_i = total inventory of isotope i in repository or ore body, curies

DF_i = isotopic dose factor, curie of isotope i required to produce selected annual dose.

The values of RQ for spent fuel are given in Table I-5; those for HLW in Table I-6.* These values are plotted in Figures I-3 and I-4, respectively.** These values are large starting for spent fuel at nearly 1E+19 at t = 0, dropping to about 5E+16 at 500 years, and not changing by as much as a factor of ten during the next one million years.*** But apply precisely the same reasoning to an ore body. As shown in Table I-7 the total ore body required to produce 1E+04 GWe-year of power will contain almost 9E+05 curies of U-238 and of each of its daughters. Table I-8 gives the precisely analogous RQ values for the ore body. This value, totally dominated by Ra-226, is about 4E+16, a value which remains essentially constant over the million-year time period considered herein.

---

* Solely for purposes of illustration the assumed dose is 5 mrem/year to an individual who uses water from a river near the repository or ore body for drinking and for irrigating his food products.

** Isotopes which contribute at least 1 percent to the total RQ value at some point in a million years are plotted. Three other long-lived isotopes (C-14, I-129, Tc-99) are also shown.

*** Note that over the period from 10,000 years to 100,000 years the value increases, due to the in-growth rate of Ra-226, such that at 100,000 years the required RQ is again about half that at 500 years.

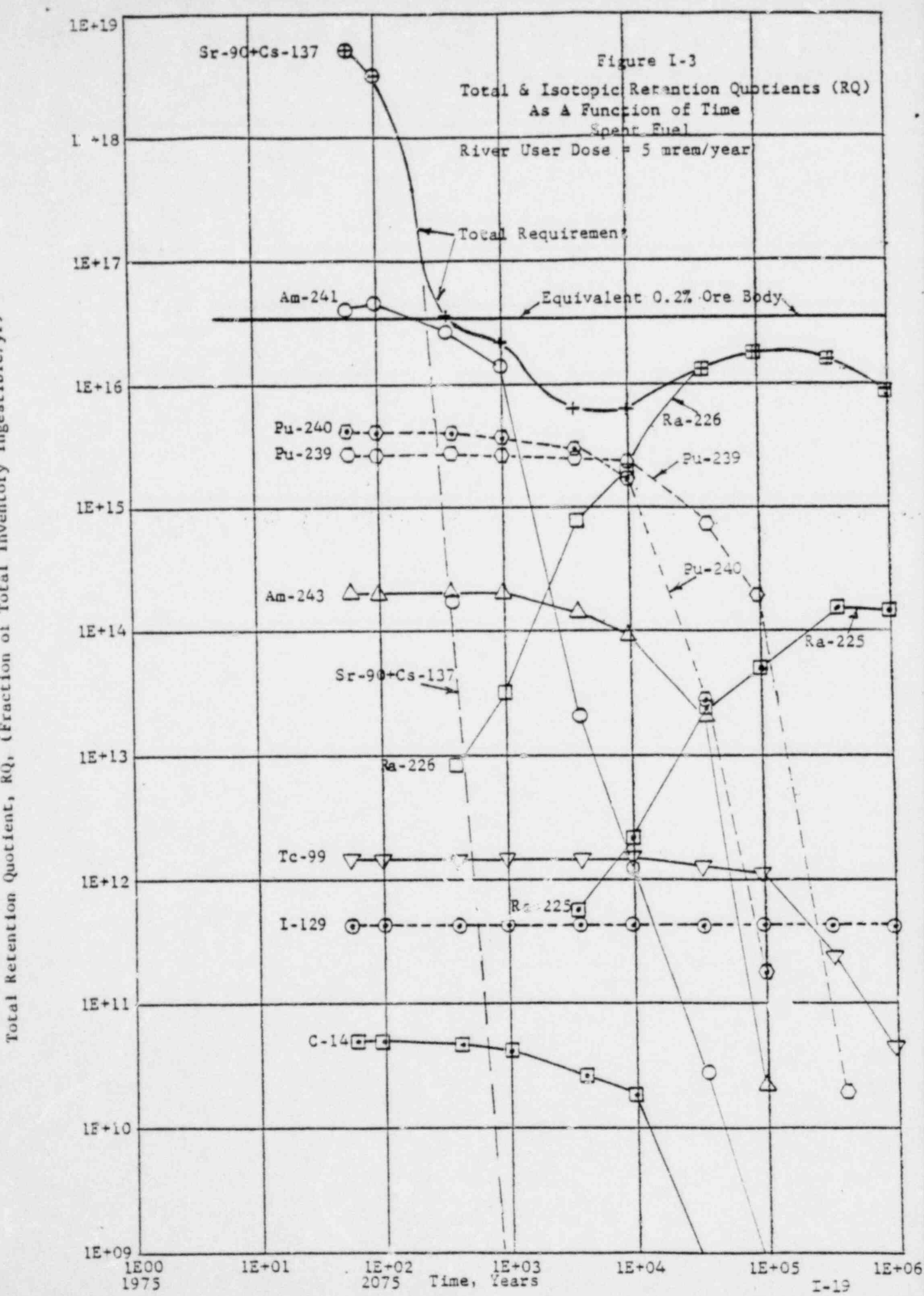


Figure I-3
Total & Isotopic Retention Quotients (RQ)
As A Function of Time
Spent Fuel
River User Dose = 5 mrem/year
Sr-90+Cs-137
Total Requirement
Equivalent 0.2% Ore Body
Am-241
Ra-226
Pu-240
Pu-239
Am-243
Ra-225
Tc-99
I-129
C-14
Total Retention Quotient, RQ, (Fraction of Total Inventory Ingestible/yr)$^{-1}$
Time, Years
1E00
1975
1E+01
1E+02
2075
1E+03
1E+04
1E+05
1E+06
1E+09
1E+10
1E+11
1E+12
1E+13
1E+14
1E+15
1E+16
1E+17
1E+18
1E+19

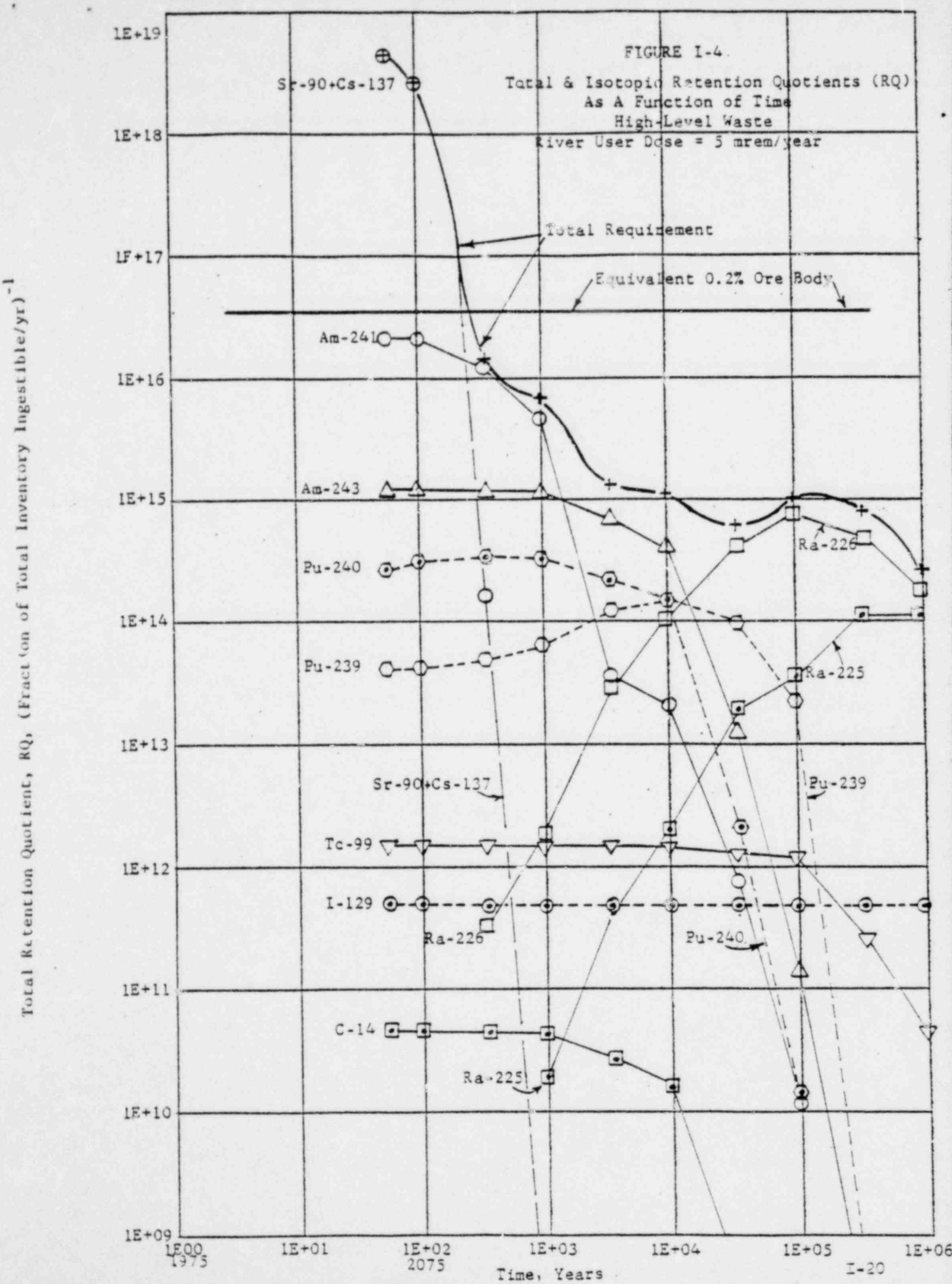


FIGURE I-4
Total & Isotopic Retention Quotients (RQ)
As A Function of Time
High-Level Waste
River User Dose = 5 mrem/year
Total Retention Quotient, RQ, (Fraction of Total Inventory Ingestible/yr)$^{-1}$
Time, Years
Sr-90+Cs-137
Total Requirement
Equivalent 0.2% Ore Body
Am-241
Am-243
Pu-240
Pu-239
Tc-99
I-129
Ra-226
Ra-225
C-14
1E+19
1E+18
1E+17
1E+16
1E+15
1E+14
1E+13
1E+12
1E+11
1E+10
1E+09
1E00
1975
1E+01
1E+02
2075
1E+03
1E+04
1E+05
1E+06

Since we have applied the same reasoning to spent fuel, waste, and ore, it is appropriate to take the ratio of the RQ values for each to see how the requirements compare. This has been done and the resulting ratios are shown in Table I-9 and plotted in Figure I-5. It can be seen that for spent fuel the required waste repository retention is never more than 150 times that of the ore body; after 500 years it is about equivalent to the ore body; and it remains at 15 to 50 percent of the ore body value for a million years. The RQ for both spent fuel and HLW drop below the ore within the first 1000 years. In both cases there is a buildup in the 10,000- to 100,000-year range as Ra-226 builds back in.* For HLW the same pattern of buildup is seen but it remains only a few percent of the ore body over this entire period.** We believe this representation is more realistic than the more simplistic char- acterization based solely on drinking water shown in Figure I-2. The differ- ence in the two representations is explained by the dominance of the ingestion pathway by Sr-90 and Cs-137 in the early years and by Ra-226 in the later years.

So far we have discussed the total retention requirement (RQ) from source (repository or ore body) to receptor (man). It should be recognized that not everything which might reach the accessible environment results in dose to man. In fact very considerable dilution takes place in the environment and only a small fraction of the radioactive material which might reach a river will reach an individual receptor. This environmental dilution (RQ_e) can be estimated with some fair degree of accuracy. To illustrate this the following reasonable possibility has been postulated:

> A member of the general public who obtains drinking water from a river (flow 4000 cfs***) which is near the repository site and who obtains a substantial fraction of his food from the river and from crops irrigated by this river water. Simply for purposes of illustration and without implying that this level would constitute an appropriate limit, the value of 5 mrem/year was used for this person.

---

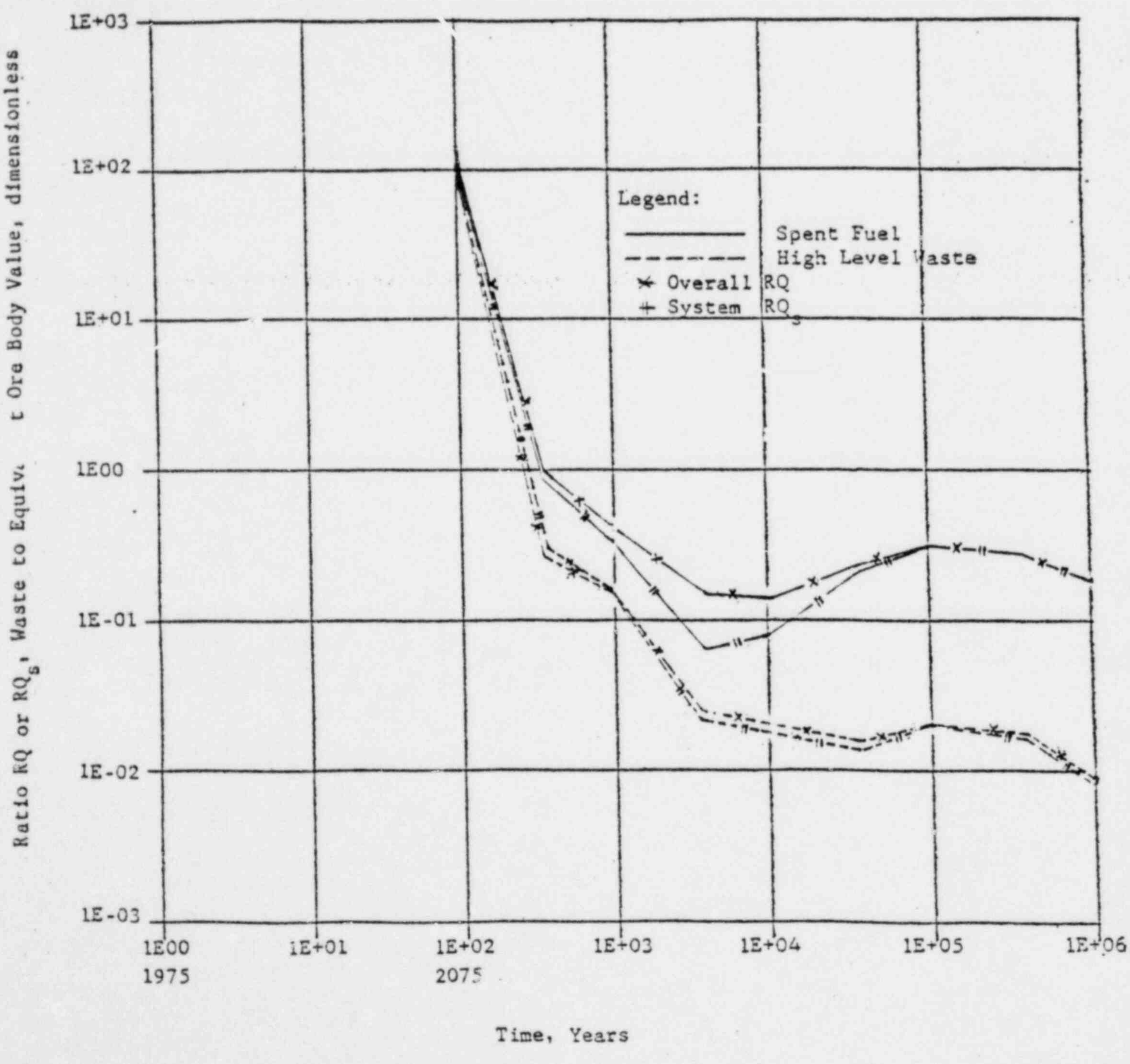
* The buildup comes from the Cm-242--Pu-238 decay and from in-growth from U-238.

** The difference in HLW comes from the smaller quantities of Pu-238 and U-238 contained therein.

*** About the flow of the Savannah River.

Figure I-5

Ratio of RQ & RQ_s Values for Spent Fuel & HLW to the Equivalent Ore Body Values



The same computer code that was used to develop the RQ values also provides the degree of retention or Environmental Retention Quotient (RQ_e) which is provided by the river. The ratio of the total RQ to the RQ_e gives the Retention Quotient which the remainder of the system must provide if the receptor is not to exceed the assumed dose. We call this the System Retention Quotient (RQ_s). It represents the overall system retention which must be provided by the total system from source (repository or ore body) to the accessible environment (in this example, the river). The values of RQ_e for spent fuel, HLW, or ore related to the previously defined river water user are shown in Table I-10; the ratios $RQ/RQ_e = RQ_s$ are shown in Table I-11 and plotted in Figure I-6. The RQ_s values for HLW are given in Table I-12 and plotted in Figure I-7.

Looking back at Figure I-5 (wherein the ratio of RQ values for spent fuel/ore and HLW/ore were shown) the ratios of RQ_s values are also shown. It can be seen that the system requirements follow the overall RQ requirements very closely. This is not surprising--it simply means that the environmental dilution behavior for all three (spent fuel, HLW, ore) is quite similar.

Figure I-6 shows that once the Sr-90 and Cs-137 have decayed the total system retention for spent fuel (from total inventory to appearance in water source) needs to be between 1E+07 and 1E+08, varying somewhat with time. This means that one part in 1E+07 to 1E+08 of the total effective inventory could be permitted to reach the water source each year. Individual isotopic requirements range from 1E+09 to less than unity. (Except for three long-lived isotopes of interest, isotopes whose RQ_s values are never more than about 1E+06 are not shown on Figures I-6 or I-7.) Figure I-7 is a similar plot for HLW. The system requirements for HLW are about a factor of ten lower than those for spent fuel.

D WHAT IS THE RELATIONSHIP OF A REPOSITORY TO AN ORE BODY?

Although all of the foregoing analyses and calculations provide useful information as to the retention required of a repository system as compared to an ore body, they do not specifically address the attributes of a repository system as compared to an ore body. In essence, the remaining question is:

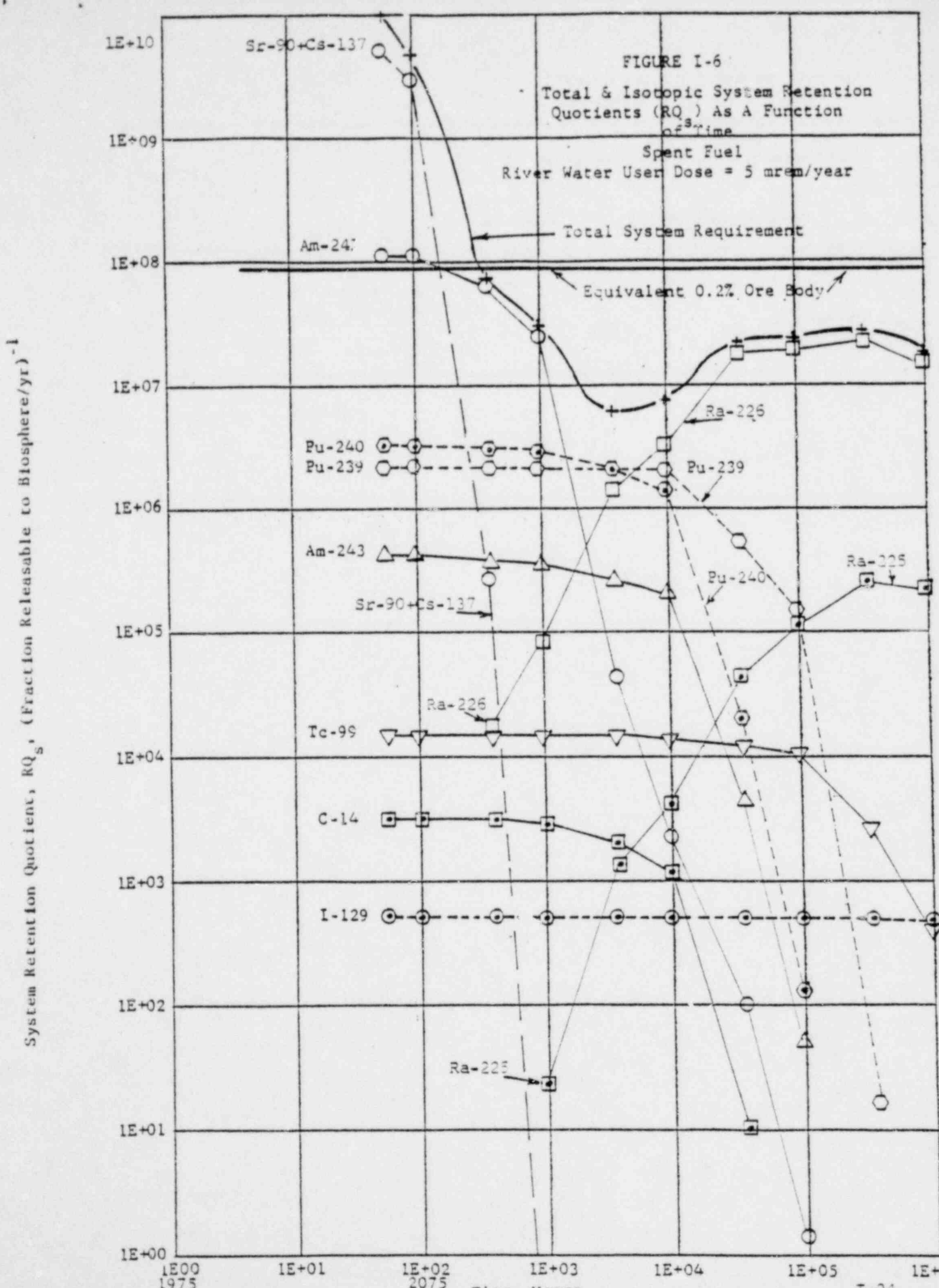


FIGURE I-6
Total & Isotopic System Retention Quotients (RQ_s) As A Function of Time
Spent Fuel
River Water User Dose = 5 mrem/year
Total System Requirement
Equivalent 0.2% Ore Body
Sr-90+Cs-137
Am-241
Ra-226
Pu-240
Pu-239
Am-243
Ra-225
Tc-99
C-14
I-129
System Retention Quotient, RQ_s, (Fraction Releasable to Biosphere/yr)$^{-1}$
1E+00
1E+01
1E+02
1E+03
1E+04
1E+05
1E+06
1E+07
1E+08
1E+09
1E+10
1E00
1975
1E+01
1E+02
2075
1E+03
1E+04
1E+05
1E+06
Time, Years

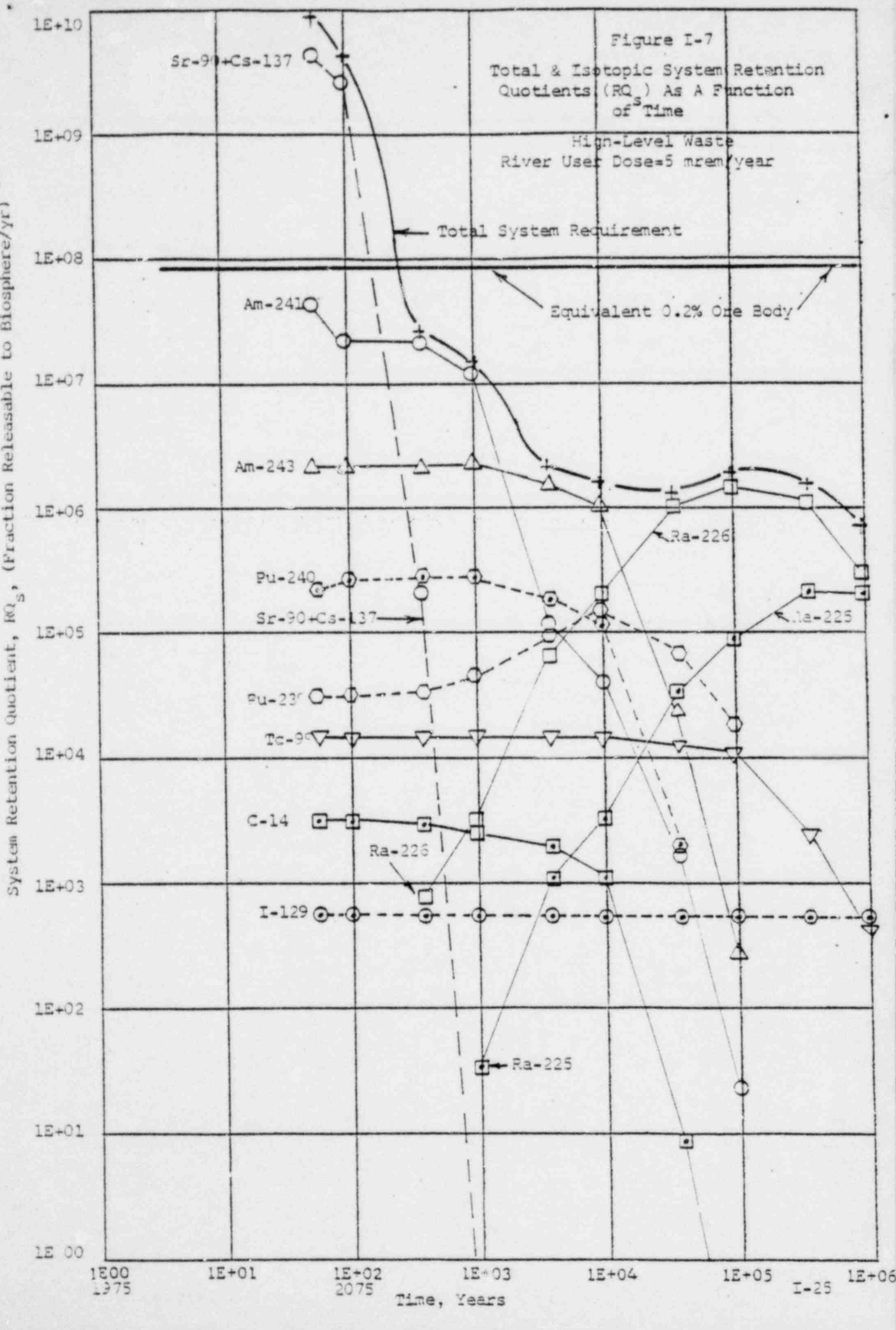


Figure I-7
Total & Isotopic System Retention Quotients (RQ_s) As A Function of Time
High-Level Waste
River User Dose=5 mrem/year
Total System Requirement
Equivalent 0.2% Ore Body
Sr-90+Cs-137
Am-241
Am-243
Ra-226
Pu-240
Sr-90+Cs-137
Ra-225
Pu-239
Tc-99
C-14
Ra-226
I-129
Ra-225
System Retention Quotient, RQ_s, (Fraction Releasable to Biosphere/yr)$^{-1}$
1E+10
1E+09
1E+08
1E+07
1E+06
1E+05
1E+04
1E+03
1E+02
1E+01
1E 00
1E00 1975
1E+01
1E+02 2075
1E+03
1E+04
1E+05
1E+06
Time, Years

with what confidence can we expect the total repository system to provide the retention that is desired?

One reasonable way to approach this question is by analogy to the natural ore body whence came the fuel in the first place. Figures I-6 and I-7 show that a system retention quotient of about 9E+07 is required if a receptor is not to exceed the assumed dose limit from an ore body. We have eons of evidence that mankind has been able to coexist successfully with ore bodies, thus suggesting that the RQ for the ore body must be in this general order of magnitude. How then does a repository compare to an ore body? When waste is placed in a repository there are six "barriers" between the waste and man, as illustrated in Figure I-8. Only three of these six barriers also apply to uranium ore bodies, which are also illustrated on the figure. The six barriers which serve to reduce or eliminate exposure of man from a waste repository are:

<u>Barrier #1--Site Selection</u> (See also Section III-B, infra)

Site selection applies only to the repository. Ores are distributed widely. Most, if not all, at some time in the past have been in direct contact with water. Some still are. By contrast a repository site will be selected to give a high degreee of assurance that water will not contact the waste, or if it does, that there will be a long pathway back to man with reasonable evidence that the radioactive materials will be held up in the intervening soils for long periods of time.[42,43,44,45] Selection of a water-free repository should ensure that the waste is more effectively contained than an ore body located in proximity to water, or stated another way, a water-free site can contribute all or nearly all of the necessary containment quite by itself.

<u>Barrier #2--Waste Form</u> (See also Section III-C, infra)

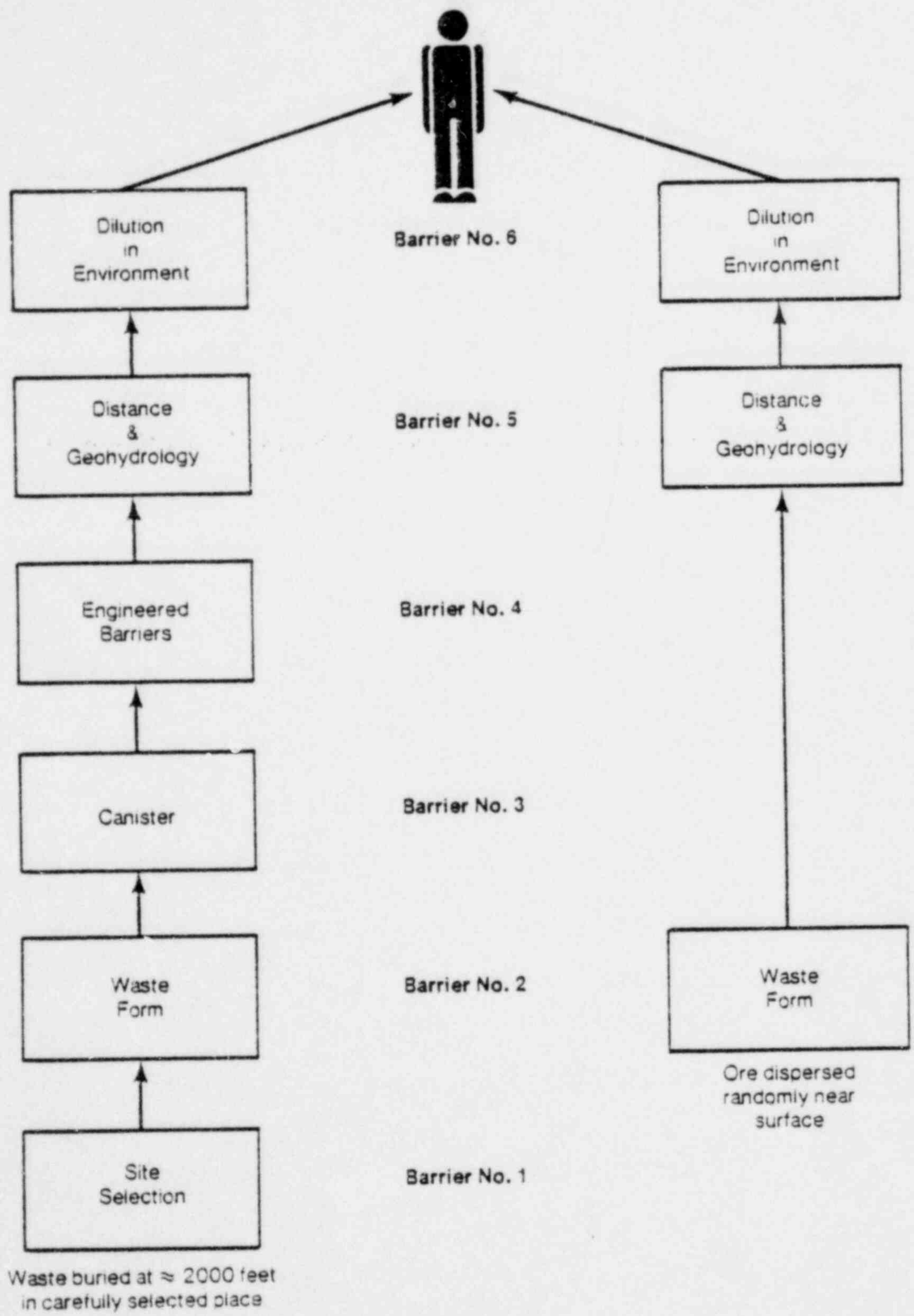
Spent fuel consists of a highly insoluble Zircaloy cladding which contains the waste in a highly insoluble sintered ceramic form (UO_2). Thus the contained radioisotopes would not dissolve rapidly even if water were to contact the waste.* This barrier applies to both repository and ore.

---

* In the case of HLW the waste could be put into a number of forms (eg, glass) with equal or better spent fuel insolubility.[46]

Figure I-8

Barriers between Buried Radioactivity and Man



Natural ore bodies are also in a highly insoluble form; were this not so they would no longer be in place. However, ores are in a more highly dispersed form than spent fuel and thus would be expected to leach more readily than the fuel elements. On balance spent fuel is probably in at least as unleachable form as ore and likely more so.

Barrier #3--The Canister (See also Section III-C, infra)
The waste form can be surrounded by a high-integrity container. No container can be expected to last forever, but it is completely reasonable to conclude that a container can be designed and built which will give reasonable assurance of a very high degree of containment during handling and emplacement, and can provide additional barriers (if deemed necessary for overall system performance) during the initial years when the fission products are dominant. This barrier obviously applies only to the repository.

Barrier #4--Additional Engineered Features (See also Section III-D, infra)
The space between the waste canisters and the undisturbed rock of the mine can be used for other engineered barrier features. These could include additional overpack materials, treatment of the mine-rock interface to seal any residual cracks, filling of the space between the canisters and the mine walls with materials selected to impede the ingress of water and/or, if water were to intrude, to absorb radioactive materials which might come out of the waste package itself. It is also possible that some of the overpack materials could be selected to adjust the chemistry of key radionuclides to enhance their holdup in the surrounding geohydrological regime.[47,48,49] Again this barrier applies only to the repository.

Barrier #5--Geohydrology and Distance to Receptor (See also Sections III-E, F, G, and H, infra)
This barrier applies both to the repository and to ore although not necessarily equally. Part of the site selection process (Barrier #1) is to choose a deep location where any radioactive material which does eventually pass the undisturbed rock interface will have a long path to travel

to reach man, a path which has favorable characteristics for holding up these materials for long times during their travel from repository to man.[50,51,52] Even if there is movement of radioactive material from the waste through the previous barriers, and if groundwater does come in contact with the waste, the combination of very slow movement of the groundwater itself and of holdup of most radionuclides on the intervening soil will act to greately reduce the dose a receptor might otherwise receive. It should be noted that this factor has to be one of the major ones protecting man from ore bodies. Ore body locations are not selected to protect man. The depth factor alone has therefore to heavily favor the waste over the ore in terms of minimizing release of radionuclides to the accessible environment.

Barrier #6--Dilution in the Environment (See pages I-21 and 23, supra)
If radioactive material does eventually succeed in escaping the multibarriered labyrinth in which it has been placed and some of it shows up in a potable water supply, be that a well or a river, not all of that which reaches the water supply will reach man. In fact only a small fraction will result in producing dose to man. The effect of this factor has been taken into account in determining the RQ_s for both waste and ore and is roughly the same for both.

E SUMMARY

This section presented what UNWMG-EEI consider to be the basic requirements of a waste disposal system. A series of questions were discussed which addressed safe disposal, risk, degree of potential hazard associated with spent fuel as compared to a natural uranium ore body, and the degree of containment which must be provided by a repository system as compared to that which nature has provided for an ore body. The discussion demonstrated that, taking into account the types and levels of radiation exposure that society finds acceptable, and comparing the performance of prospective components of an overall geologic repository system to the natural barriers of an uranium ore body, it is evident that wastes can be disposed of safely, as judged by any rational and reasonable criteria.

In this discussion, the concepts of containment and isolation of waste were identified as two differing objectives of the disposal system. Reasonable risk levels with respect to potential radiation exposure in the event of releases from the system were suggested based on a comparison with the natural variations in background radiation. The time frame of greatest concern was identified in order to better understand the needed repository system performance. Further, a new analytical method (the Retention Quotient) was developed and presented as a useful tool in assessing the ability of the repository system (including its components) to meet the performance criteria that will be set. The natural and engineered barriers of the overall repository system were compared with the ore body's natural barriers to provide perspective for judging our technological capability to protect the public now and in the future. As shown in this section, it is entirely feasible, utilizing the ore body's natural system as a benchmark, to provide combinations of natural and engineered barriers which will adequately protect society.

The information contained in this section, including the new approach to analyzing retention requirements for an overall repository system, can also provide an additional useful tool in the design of the repository system as well as in establishing and assessing performance criteria for regulatory purposes.

In the following sections we demonstrate that the technology is essentially available to provide the required system of barriers to meet the yet to be defined criteria. It should be emphasized we do not suggest that all the barriers will necessarily be required to meet the system criteria. Which barriers are used will depend upon the specific system performance requirements. It is the fact that many barriers are available, including engineered barriers not included in the natural system, which dictates a finding of confidence in waste disposal capability.

Table I-1

Summary of Representative Dose Equivalent Rates in the United States from All Naturally Occurring Radiation Sources[a]

Source	Dose Equivalent Rates (mrem/yr)				
	Gonads	Lungs	Bone		GI Tract
			Surfaces	Marrow	
Cosmic Radiation[b]	28	28	28	28	28
Cosmogenic Radionuclides	0.7	0.7	0.8	0.7	0.7
External Primordial Radionuclides[b]	26	26	26	26	26
Primordial Radionuclides in the Body	24	21	58	22	21
Inhaled Radionuclides[c]		106			
	79	182	113	77	76

[a] Adapted from NCRP Report No. 45, Table 44.

[b] Indoor dose equivalent rates

[c] Doses to organs other than lung included in "Primordial Radionuclides in the Body."

TABLE I-2

CURIES OF EACH NUCLIDE INGESTED TO PRODUCE
AN ORGAN WEIGHTED TOTAL BODY DOSE TO AN ADULT OF 5 MREM/YEAR

NUCLIDE	CURIES	NUCLIDE	CURIES
H----3	3.7E-05	RA-223	2.7E-09
C---14	5.0E-06	RA-224	7.6E-09
MN--54	1.1E-06	RA-225	2.1E-09
FE--55	3.0E-07	RA-226	1.9E-11
CO--60	3.4E-07	RA-228	3.6E-11
NI--59	1.4E-06	AC-225	5.0E-08
NI--63	2.0E-07	AC-227	1.2E-08
SE--79	5.0E-06	TH-227	3.8E-08
RB--87	9.7E-07	TH-228	2.2E-08
SR--90	1.7E-09	TH-229	2.8E-09
Y---90	2.0E-07	TH-230	1.2E-08
ZR--93	8.5E-06	TH-232	1.7E-08
NB-93M	5.4E-06	TH-234	1.8E-07
TC--99	7.0E-06	PA-231	5.7E-09
RU-106	1.1E-07	PA-233	1.3E-06
PD-107	1.6E-05	U--232	5.2E-09
AG110M	3.4E-07	U--233	2.4E-08
CD113M	7.5E-07	U--234	2.5E-08
SN-126	2.4E-07	U--235	2.5E-08
SB-125	1.5E-07	U--236	2.6E-08
SB-126	2.2E-07	U--238	2.4E-08
TE125M	1.2E-06	NP-237	1.6E-08
I--129	2.2E-08	NP-239	9.7E-07
CS-134	3.5E-08	PU-238	3.4E-08
CS-135	3.9E-07	PU-239	3.1E-08
CS-137	5.3E-08	PU-240	3.1E-08
CE-144	1.3E-07	PU-241	1.5E-06
PR-144	6.9E-01	PU-242	3.3E-08
PM-147	2.3E-06	AM-241	2.2E-08
SM-151	3.9E-06	AM242M	2.1E-08
EU-152	8.0E-07	AM-243	2.1E-08
EU-154	3.8E-07	CM-242	2.1E-07
EU-155	2.1E-06	CM-243	2.8E-08
PB-210	1.3E-09	CM-244	3.6E-08
BI-210	3.6E-07	CM-245	1.9E-08
PO-210	1.4E-08		

Table I-3

Once-Through Cycle--Total System Actinide Radioactivity Inventory in Repositories for Times to One Million Years, Curies[a]

Radionuclides	Year 2000	Year 2050	Year 2070	Geologic Time (Years Beyond 1975) 500	1,000	5,000	10,000	50,000	100,000	500,000	1,000,000
^{245}Cm	7.79E+03	6.49E+04	6.48E+04	6.26E+04	6.00E+04	4.29E+04	2.82E+04	9.85E-02	1.49E+01	0.	0.
^{244}Cm	3.64E+07	1.19E+08	5.54E+07	1.06E+01	5.09E-00	0.	0.	0.	0.	0.	0.
^{243}Cm	1.36E+05	6.49E+05	4.21E+05	6.65E+01	1.32E-03	0.	0.	0.	0.	0.	0.
^{242}Cm	3.63E+05	2.75E+06	2.51E+06	3.97E+05	4.06E+04	4.86E+04	0.	0.	0.	0.	0.
^{243}Am, ^{239}Np	1.24E+06	1.02E+07	1.01E+07	9.77E+06	9.24E+06	6.50E+06	4.13E+06	1.10E+05	1.19E+03	0.	0.
^{242m}Am, ^{242}Am	9.29E+05	6.69E+06	6.11E+06	9.68E+05	9.90E+04	1.19E-03	0.	0.	0.	0.	0.
^{241}Am	3.57E+07	1.15E+09	1.27E+09	7.21E+08	2.24E+08	5.81E+05	2.35E+04	9.87E-02	1.49E+01	0.	0.
^{242}Pu	7.39E+04	5.91E+05	5.91E+05	5.91E+05	5.90E+05	5.86E+05	5.81E+05	5.40E+05	4.93E+05	2.27E+05	9.50E+04
^{241}Pu	3.02E+09	7.75E+09	3.04E+09	6.37E+04	6.01E+04	4.30E+04	2.93E+04	9.87E-02	1.49E+01	0.	0.
^{240}Pu	2.23E+07	1.69E+08	1.68E+08	1.62E+08	1.54E+08	1.02E+08	6.11E+07	1.01E+06	5.99E+03	0.	0.
^{239}Pu	1.48E+07	1.10E+08	1.10E+08	1.08E+08	1.07E+08	9.58E+07	8.35E+07	2.71E+07	6.55E+06	7.60E+02	5.15E-05
^{238}Pu	8.81E+07	5.87E+08	5.03E+08	2.23E+07	5.32E+05	1.17E-03	0.	0.	0.	0.	0.
^{236}Pu	1.08E+03	2.86E+02	2.21E+00	0.	0.	0.	0.	0.	0.	0.	0.
^{237}Np, ^{233}Pa	2.92E+04	2.52E+05	2.68E+05	5.03E+05	6.87E+05	8.22E+05	8.22E+05	8.12E+05	7.99E+05	7.02E+05	5.97E+05
^{238}U, ^{234}Th, ^{234m}Pa	4.86E+04	3.60E+05	3.60E+05	3.60E+05	3.60E+05	3.60E+05	3.60E+05	3.60E+05	3.60E+05	3.60E+05	3.60E+05
^{236}U	1.06E+04	8.46E+04	8.47E+04	8.64E+04	8.88E+04	1.03E+05	1.15E+05	1.32E+05	1.[illegible]E+05	1.30E+05	1.29E+05
^{235}U, ^{231}Th	1.70E+03	1.34E+04	1.34E+04	1.35E+04	1.36E+04	1.44E+04	1.53E+04	1.92E+04	2.[illegible]E+04	2.10E+04	2.10E+04
^{234}U	4.68E+04	4.40E+05	4.70E+05	6.30E+05	6.51E+05	6.46E+05	6.39E+05	5.83E+05	5.[illegible]E+05	2.51E+05	1.52E+05
^{233}U	4.00E+00	4.66E+01	5.77E+01	3.72E+02	1.06E+03	7.79E+03	1.64E+04	7.80E+04	1.41E+05	3.27E+05	3.17E+05
^{232}U	7.46E+02	5.06E+03	4.19E+03	8.56E+01	6.95E-01	0.	0.	0.	0.	0.	0.
^{231}Pa	1.35E+00	1.53E+01	1.81E+01	7.07E+01	1.45E+02	7.02E+02	1.39E+03	5.7E+03	8.58E+03	1.05E+04	1.05E+04
^{230}Th	6.39E+00	1.47E+02	2.25E+02	2.09E+03	5.02E+03	2.68E+04	5.30E+04	2.16E+05	3.22E+05	3.32E+05	1.67E+05
^{229}Th, 7 Daughters[b]	1.77E+02	9.87E+01	1.78E+00	5.68E+01	3.25E+02	1.17E+04	4.51E+04	5.01E+05	1.03E+06	2.63E+06	2.55E+06
^{228}Th, 6 Daughters[c]	4.32E+03	3.63E+04	3.01E+04	6.16E+02	5.03E+01	1.60E-01	3.60E-01	2.11E+00	4.38E+00	2.24E+01	4.46E+01
^{227}Ac, 7 Daughters[d]	1.74E+00	6.79E+01	9.96E+01	5.47E+02	1.16E+03	5.61E+03	1.11E+04	4.57E+04	6.37E+04	8.42E+04	8.41E+04
^{232}Th, 2 Daughters[e]	5.77E-06	3.07E-04	5.45E-04	5.25E-03	1.20E-02	6.86E-02	1.50E-01	9.03E-01	1.38E+00	9.60E+00	1.91E+01
^{226}Ra, 5 Daughters[f]	1.37E-01	8.57E+00	1.81E+01	1.02E+03	5.20E+03	9.40E+04	2.47E+05	1.30E+06	2.02E+06	1.81E+06	1.01E+06
^{210}Pb, 2 Daughters[g]	1.22E-02	1.61E+00	4.10E+00	4.81E+02	2.60E+03	4.70E+04	1.24E+05	6.49E+05	1.01E+06	9.06E+05	5.02E+05
TOTAL	3.27E+09	9.90E+09	5.16E+09	1.03E+09	5.97E+08	2.08E+08	1.52E+08	3.34E+07	1.36E+07	7.77E+06	5.99E+06

a. Values less than 1.0E-09 have been designated as zero.
b. ^{229}Th, 7 Daughters are ^{225}Ra, ^{225}Ac, ^{221}Fr, ^{217}At, ^{213}Bi, ^{209}Pb and ^{209}Tl is 9% of ^{229}Th and ^{213}Po is 91% of ^{229}Th.
c. ^{228}Th, 6 Daughters are ^{224}Ra, ^{220}Rn, ^{216}Po, ^{212}Pb, ^{212}Bi and ^{208}Tl is 36% ov ^{228}Th and ^{212}Po is 64% of ^{228}Th.
d. ^{227}Ac, 7 Daughters are ^{227}Th, ^{223}Ra, ^{219}Rn, ^{215}Po, ^{211}Pb, ^{211}Bi and ^{207}Tl.
e. ^{232}Th, 2 Daughters are ^{228}Ra and ^{228}Ac.
f. ^{226}Ra, 5 Daughters are ^{222}Rn, ^{218}Po, ^{214}Pb, ^{214}Bi, and ^{214}Po.
g. ^{210}Pb, 2 Daughters are ^{210}Bi and ^{210}Po.

NOTE: In accounting for the activity in this manner, branching decay in the case of ^{208}Tl (36%) - ^{212}Po (64%); and ^{209}Tl (9%) - ^{213}Po (91%) were counted as a single daughter in each case. Minor branching (1% or less) was ignored.

Table I-3 (cont)

Once-Through Cycle--Total System Fission-Product Radioactivity Inventory in Repositories for Times to One Million Years, Curies[a]

Major Radionuclides	Year 2000	Year 2050	Year 2079	Geologic Time (Years Beyond 1979) 100	1,000	2,000	5,000	10,000	50,000	500,000	1,000,000
^{3}H	7.69E+06	7.19E+07	7.29E+06	9.32E-04	0.	0.	0.	0.	0.	0.	0.
^{14}C	4.17E+04	3.16E+05	3.16E+05	3.01E+05	2.83E+05	1.74E+05	9.53E+04	7.57E+02	1.79E+00	0.	0.
^{54}Mn	1.51E+04	2.40E+02	1.24E-05	0.	0.	0.	0.	0.	0.	0.	0.
^{55}Fe	2.19E+07	5.27E+04	2.58E+04	0.	0.	0.	0.	0.	0.	0.	0.
^{60}Co	5.46E+07	4.17E+07	2.99E+06	0.	0.	0.	0.	0.	0.	0.	0.
^{59}Ni	1.46E+05	1.15E+06	1.15E+06	1.14E+06	1.14E+06	1.10E+06	1.05E+06	7.45E+05	4.83E+05	1.51E+04	1.99E+02
^{63}Ni	2.16E+07	1.26E+08	1.17E+06	5.60E+04	1.20E+05	1.08E-08	0.	0.	0.	0.	0.
^{79}Se	1.65E+04	1.21E+05	1.21E+05	1.30E+05	1.30E+05	1.24E+05	1.18E+05	7.70E+04	4.52E+04	6.26E+02	3.09E+00
^{85}Kr	2.11E+08	4.14E+08	1.15E+08	6.41E-04	0.	0.	0.	0.	0.	0.	0.
^{87}Rb	8.14E-01	6.47E+00	6.47E+00	6.47E+00	6.47E+00	6.47E+00	6.47E+00	6.47E+00	6.47E+00	6.47E+00	6.47E+00
^{90}Sr, ^{90}Y	4.67E+09	2.01E+10	1.23E+10	5.81E+05	2.58E+00	0.	0.	0.	0.	0.	0.
^{93}Zr, ^{93m}Nb	1.57E+05	1.24E+06	1.24E+06	1.24E+06	1.24E+06	1.24E+06	1.24E+06	1.22E+06	1.19E+06	9.68E+05	7.84E+05
^{99}Tc	6.10E+05	4.82E+06	4.82E+06	4.81E+06	4.80E+06	4.74E+06	4.66E+06	4.09E+06	3.46E+06	9.25E+05	1.78E+05
^{106}Ru, ^{106}Rh	7.57E+07	2.15E+06	2.21E+00	0.	0.	0.	0.	0.	0.	0.	0.
^{107}Pd	4.69E+03	3.66E+04	3.66E+04	3.66E+04	3.66E+04	3.66E+04	3.66E+04	3.64E+04	3.63E+04	3.49E+04	3.32E+04
^{110m}Ag	2.13E+04	1.46E+02	3.02E-07	0.	0.	0.	0.	0.	0.	0.	0.
^{113m}Cd	2.19E+05	5.25E+05	2.05E+05	4.27E-04	0.	0.	0.	0.	0.	0.	0.
^{125}Sb, ^{125m}Te	4.66E+07	1.17E+07	6.94E+04	0.	0.	0.	0.	0.	0.	0.	0.
^{126}Sn, ^{126}Sb	4.52E+04	3.55E+05	3.55E+05	3.54E+05	3.52E+05	3.43E+05	3.31E+05	2.51E+05	1.78E+05	1.11E+05	3.49E+02
^{129}I	1.57E+03	1.24E+04	1.24E+04	1.24E+04	1.23E+04	1.24E+04	1.23E+04	1.22E+04	1.22E+04	1.21E+04	1.19E+04
^{134}Cs	2.77E+08	4.07E+07	4.73E+06	0.	0.	0.	0.	0.	0.	0.	0.
^{135}Cs	1.22E+04	1.00E+05	1.00E+05	1.00E+05	1.00E+05	1.00E+05	9.99E+04	9.97E+04	9.79E+04	8.92E+04	7.95E+04
^{137}Cs, ^{137m}Ba	6.49E+09	2.87E+10	1.81E+10	1.61E+06	1.56E+01	0.	0.	0.	0.	0.	0.
^{144}Ce, ^{144}Pr	3.24E+07	4.99E+05	9.13E-03	0.	0.	0.	0.	0.	0.	0.	0.
^{147}Pm	2.56E+08	8.23E+07	4.21E+05	0.	0.	0.	0.	0.	0.	0.	0.
^{151}Sm	5.06E+07	3.18E+08	2.71E+08	1.09E+07	2.23E+05	2.99E-09	0.	0.	0.	0.	0.
^{152}Eu	2.97E+05	6.66E+05	2.10E+05	1.36E-08	0.	0.	0.	0.	0.	0.	0.
^{154}Eu	1.57E+08	4.31E+06	1.30E+08	4.41E+00	1.31E-09	0.	0.	0.	0.	0.	0.
^{155}Eu	6.49E+06	7.50E+05	3.55E+02	0.	0.	0.	0.	0.	0.	0.	0.
Minor Radionuclides											
^{46}Sc	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
^{51}Cr	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
^{58}Co	1.13E-02	1.75E-09	0.	0.	0.	0.	0.	0.	0.	0.	0.
^{59}Fe	2.35E-07	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
^{65}Zn	1.70E+02	1.11E+00	1.19E-09	0.	0.	0.	0.	0.	0.	0.	0.
^{89}Sr	5.71E-08	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
^{91}Y	3.89E-04	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
^{95}Zr	7.78E-06	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
^{95m}Nb	4.56E-04	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
^{95}Nb	3.99E-03	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
^{103}Ru	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
^{123}Sn	3.33E+01	7.76E-03	0.	0.	0.	0.	0.	0.	0.	0.	0.
^{124}Sb	2.85E-07	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
^{127m}Te, ^{127}Te	1.45E+01	1.28E-03	0.	0.	0.	0.	0.	0.	0.	0.	0.
^{141}Ce	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
^{148}Pm	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
^{153}Gd	1.63E+03	1.26E+01	1.13E-08	0.	0.	0.	0.	0.	0.	0.	0.
^{160}Tb	3.76E-04	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
^{181}W	6.20E-05	4.17E-08	0.	0.	0.	0.	0.	0.	0.	0.	0.
TOTAL	1.26E+10	5.04E+10	3.11E+10	2.66E+07	8.25E+06	7.77E+06	7.59E+06	6.32E+06	5.51E+06	2.09E+06	1.09E+06

a. Values less than 1.0E-9 have been designated as zero.

Table I-4

U and Pu Recycle--Total System Actinide Radioactivity Inventory in Repositories for Times to One Million Years, Ci[a]

Major Radionuclide	Year			Geologic Time (Years Beyond 1975)							
	~2000	2050	2070	500	1,000	5,000	10,000	50,000	100,000	500,000	1,000,000
^{245}Cm	3.82E+04	1.25E+06	1.24E+06	1.20E+06	1.15E+06	8.25E+05	5.43E+05	1.89E+04	2.86E+02	0.	0.
^{244}Cm	1.01E+08	1.36E+09	6.34E+08	1.21E+02	5.83E-07	0.	0.	0.	0.	0.	0.
^{243}Cm	1.69E+05	1.72E+06	1.12E+06	1.77E+02	3.49E-03	0.	0.	0.	0.	0.	0.
^{242}Cm	8.51E+05	2.25E+07	2.05E+07	3.25E+06	3.32E+05	3.38E-03	0.	0.	0.	0.	0.
^{243}Am, ^{239}Np	2.10E+06	5.35E+07	5.34E+07	5.15E+07	4.92E+07	3.43E+07	2.18E+07	5.31E+05	6.25E+03	0.	0.
^{242m}Am, ^{242}Am	2.06E+06	5.47E+07	5.00E+07	7.92E+06	8.09E+05	9.67E-03	0.	0.	0.	0.	0.
^{241}Am	3.58E+07	5.56E+08	5.43E+08	2.86E+08	1.29E+08	1.38E+06	5.44E+05	1.90E+04	2.86E+02	0.	0.
^{242}Pu	1.61E+03	2.23E+04	2.25E+04	2.38E+04	2.42E+04	2.41E+04	2.38E+04	2.22E+04	2.02E+04	9.73E+03	3.90E+03
^{241}Pu	6.11E+07	2.06E+08	8.14E+07	1.20E+06	1.16E+06	8.27E+05	5.43E+05	1.90E+04	2.86E+02	0.	0.
^{240}Pu	4.68E+05	1.07E+07	1.26E+07	1.37E+07	1.31E+07	8.68E+06	5.20E+06	8.60E+04	5.10E+02	0.	0.
^{239}Pu	2.11E+05	1.63E+06	1.65E+06	1.90E+06	2.25E+06	4.20E+06	5.48E+06	3.22E+06	8.09E+05	9.41E+00	6.37E-06
^{238}Pu	2.22E+06	3.02E+07	2.39E+07	6.48E+06	7.75E+05	9.59E-03	0.	0.	0.	0.	0.
^{236}Pu	4.05E+01	1.83E+03	1.41E+01	0.	0.	0.	0.	0.	0.	0.	0.
^{237}Np, ^{233}Pa	2.90E+04	4.1[illegible]E+05	4.21E+05	5.17E+05	5.88E+05	6.43E+05	6.45E+05	6.41E+05	6.31E+[illegible]	5.54E+05	4.72E+05
^{238}U, ^{234}Th, ^{234m}Pa	4.55E+02	2.97E+03	2.97E+03	2.97E+03	2.97E+03	2.97E+03	2.97E+03	2.97E+03	2.97E+02	2.97E+03	2.98E+03
^{236}U	1.04E+02	9.80E+02	9.87E+02	1.13E+03	1.34E+03	2.58E+03	3.57E+03	5.01E+03	5.03E+03	4.97E+03	4.90E+03
^{235}U, ^{231}Th	1.54E+01	1.05E+02	1.05E+02	1.06E+02	1.08E+02	1.34E+02	1.32E+02	5.66E+02	7.38E+02	7.94E+02	7.93E+02
^{234}U	4.58E+02	6.60E+03	8.25E+03	2.45E+04	3.05E+04	3.09E+04	3.05E+04	2.74E+04	2.39E+04	8.45E+03	2.83E+03
^{233}U	6.40E-01	3.11E+01	4.89E+01	4.22E+02	1.04E+03	6.37E+03	1.31E+04	6.17E+04	1.11E+05	2.58E+05	2.50E+05
^{232}U	9.89E+00	3.62E+04	2.99E+04	5.12E+02	4.97E+00	0.	0.	0.	0.	0.	0.
^{231}Pa	1.03E+00	1.19E+01	1.20E+01	1.23E+01	1.27E+01	1.66E+01	2.29E+01	1.30E+02	2.70E+02	3.97E+02	3.97E+02
^{230}Th	2.04E+00	2.36E+01	2.49E+01	8.28E+01	2.15E+02	1.26E+03	2.52E+03	1.02E+04	1.56E+04	1.14E+04	3.70E+03
^{229}Th, 7 Daughters[b]	2.30E+03	4.71E+01	1.08E+00	6.45E+01	3.43E+02	9.92E+03	3.66E+04	3.96E+05	8.11E+05	2.07E+06	2.01E+06
^{228}Th, 6 Daughters[c]	1.30E+02	2.97E+05	2.15E+05	4.40E+03	3.57E+01	2.01E+03	4.59E+03	3.04E+02	6.42E+02	3.22E+01	6.54E+01
^{227}Ac, 7 Daughters[d]	2.27E+00	6.05E+01	7.72E+01	9.77E+01	1.02E+02	1.33E+02	1.83E+02	1.04E+03	2.16E+03	3.18E+03	3.17E+03
^{232}Th, 2 Daughters[e]	7.09E-07	5.74E-06	6.54E-06	6.13E-05	1.42E-04	8.59E-04	1.97E-03	1.30E-02	2.75E-02	1.43E-01	2.35E-01
^{226}Ra, 5 Daughters[f]	7.61E-02	2.17E+00	3.41E+00	4.73E+01	2.21E+02	4.37E+03	1.17E+04	6.15E+04	9.43E+04	6.86E+04	2.22E+04
^{210}Pb, 2 Daughters[g]	8.49E-03	4.88E+01	9.41E-01	2.24E+01	1.10E+02	2.19E+03	5.35E+03	3.09E+04	4.72E+04	3.43E+04	1.11E+04
TOTAL	2.07E+08	2.30E+09	1.43E+09	3.74E+08	1.99E+08	5.06E+07	3.49E+07	5.21E+06	2.38E+06	3.03E+06	2.79E+06

a. Values less than 1.0E-09 have been designated as zero.

b. ^{229}Th, 7 Daughters are ^{225}Ra, ^{225}Ac, ^{221}Fr, ^{217}At, ^{213}Bi and ^{209}Pb and ^{209}Tl is 9% of ^{229}Th and ^{213}Po is 91% of ^{229}Th.

c. ^{228}Th, 6 Daughters are ^{224}Ra, ^{220}Rn, ^{216}Po, ^{212}Pb, ^{212}Bi and ^{208}Tl is 36% of ^{228}Th and ^{212}Po is 64% of ^{228}Th.

d. ^{227}Ac, 7 Daughters are ^{227}Th, ^{223}Ra, ^{219}Rn, ^{215}Po, ^{211}Pb, ^{211}Bi and ^{207}Tl.

e. ^{232}Th, 2 Daughters are ^{228}Ra and ^{228}Ac.

f. ^{226}Ra, 5 Daughters are ^{222}Rn, ^{218}Po, ^{214}Pb, ^{214}Bi, and ^{214}Po.

g. ^{210}Pb, 2 Daughters are ^{210}Bi and ^{210}Po.

NOTE: In accounting for the activity in this manner, branching decay in the case of ^{208}Tl (36%)-^{212}Po (64%); and ^{209}Tl (9%)-^{213}Po (91%) were counted as a single daughter in each case. Minor branching (1% or less) was ignored.

Table I-4 (cont)

U and Pu Recycle--Total System Fission Product Radioactivity Inventory in Repositories for Times to One Million Years, Ci(a)

	Year			Geologic Time (Years Beyond 1975)							
Major Radionuclide	2000	2050	2070	500	1,000	5,000	10,000	50,000	100,000	500,000	1,000,000
^{3}H	2.78E+06	1.51E+06	1.14E+06	1.50E-04	0.	0.	0.	0.	0.	0.	0.
^{14}C	6.51E+04	2.99E+05	2.98E+05	2.54E+05	2.27E+05	1.65E+05	9.00E+04	7.15E+02	1.70E+00	0.	0.
^{54}Mn	5.22E+05	2.48E+02	1.29E-05	0.	0.	0.	0.	0.	0.	0.	0.
^{55}Fe	7.42E+07	4.23E+06	2.05E+04	0.	0.	0.	0.	0.	0.	0.	0.
^{60}Co	1.26E+08	1.80E+07	2.72E+06	0.	0.	0.	0.	0.	0.	0.	0.
^{59}Ni	2.13E+05	9.90E+05	9.89E+05	9.86E+05	9.82E+05	9.48E+05	9.08E+05	6.42E+05	4.17E+05	1.32E+04	1.72E+02
^{63}Ni	3.17E+07	1.17E+08	7.01E+07	4.82E+06	1.12E+05	9.30E-09	0.	0.	0.	0.	0.
^{79}Se	1.42E+04	1.24E+05	1.24E+05	1.24E+05	1.23E+05	1.18E+05	1.12E+05	7.30E+04	4.28E+04	6.03E+02	2.93E+00
^{85}Kr	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
^{87}Rb	6.42E-01	5.72E+00	5.72E+00	5.72E+00	5.72E+00	5.72E+00	5.72E+00	5.72E+00	5.72E+00	5.72E+00	5.72E+00
^{90}Sr, ^{90}Y	1.62E+09	1.78E+10	1.07E+10	5.08E+05	2.25E+00	0.	0.	0.	0.	0.	0.
^{93}Zr, ^{93m}Nb	1.48E+05	1.22E+06	1.23E+06	1.23E+06	1.23E+06	1.22E+06	1.22E+06	1.20E+06	1.17E+06	9.75E+05	7.74E+05
^{99}Tc	5.31E+05	4.78E+06	4.77E+06	4.77E+06	4.76E+06	4.70E+06	4.62E+06	4.05E+06	3.43E+06	9.18E+05	1.76E+05
^{106}Ru, ^{106}Rh	3.77E+07	2.43E+06	2.50E+00	0.	0.	0.	0.	0.	0.	0.	0.
^{107}Pd	4.44E+03	4.61E+04	4.61E+04	4.61E+04	4.61E+04	4.61E+04	4.60E+04	4.59E+04	4.56E+04	4.29E+04	4.17E+04
^{110m}Ag	6.73E+03	1.99E+02	4.11E-07	0.	0.	0.	0.	0.	0.	0.	0.
^{113m}Cd	2.77E+05	1.08E+06	4.01E+05	8.25E-04	0.	0.	0.	0.	0.	0.	0.
^{125}Sb, ^{125m}Te	1.53E+07	1.41E+07	8.22E+04	0.	0.	0.	0.	0.	0.	0.	0.
^{126}Sn, ^{126}Sb	4.17E+04	4.17E+05	4.16E+05	4.15E+05	4.14E+05	4.03E+05	3.89E+05	2.95E+05	2.09E+05	1.31E+04	4.09E+02
^{129}I	2.44E+03	1.32E+04	1.32E+04	1.32E+04	1.32E+04	1.32E+04	1.32E+04	1.32E+04	1.32E+04	1.29E+04	1.27E+04
^{134}Cs	1.74E+08	1.99E+07	4.64E+04	0.	0.	0.	0.	0.	0.	0.	0.
^{135}Cs	1.15E+04	1.28E+05	1.28E+05	1.28E+05	1.27E+05	1.27E+05	1.27E+05	1.26E+05	1.25E+05	1.14E+05	1.01E+05
^{137}Cs, ^{137m}Ba	5.55E+09	2.66E+10	1.80E+10	1.60E+06	1.55E+01	0.	0.	0.	0.	0.	0.
^{144}Ce, ^{144}Pr	1.14E+07	4.72E+06	6.62E-03	0.	0.	0.	0.	0.	0.	0.	0.
^{147}Pm	1.97E+08	6.16E+07	4.13E+05	0.	0.	0.	0.	0.	0.	0.	0.
^{151}Sm	4.48E+07	3.40E+08	2.90E+08	1.16E+07	2.17E+05	3.20E-09	0.	0.	0.	0.	0.
^{152}Eu	2.92E+05	1.14E+06	3.59E+05	2.56E-04	0.	0.	0.	0.	0.	0.	0.
^{154}Eu	1.41E+08	5.22E+08	2.24E+08	5.69E+00	2.26E-09	0.	0.	0.	0.	0.	0.
^{155}Eu	4.20E+06	6.75E+05	4.17E+02	0.	0.	0.	0.	0.	0.	0.	0.
Minor Radionuclide											
^{46}Sc	2.38E+00	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
^{51}Cr	2.72E-03	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
^{58}Co	1.21E+04	1.64E-09	0.	0.	0.	0.	0.	0.	0.	0.	0.
^{59}Fe	1.97E-01	.	0.	0.	0.	0.	0.	0.	0.	0.	0.
^{65}Zn	1.22E+04	1.01E+00	1.08E-29	0.	0.	0.	0.	0.	0.	0.	0.
^{89}Sr	5.90E-01	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
^{91}Y	4.50E+00	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
^{95}Zr	8.99E-02	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
^{95m}Nb	5.27E+00	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
^{95}Nb	4.16E-02	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
^{103}Ru	2.71E-02	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
^{123}Sn	1.21E+01	7.43E-03	0.	0.	0.	0.	0.	0.	0.	0.	0.
^{124}Sb	2.72E-03	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
^{127m}Te, ^{127}Te	6.31E+01	1.18E-03	0.	0.	0.	0.	0.	0.	0.	0.	0.
^{141}Ce	1.25E-04	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
^{148}Pm	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
^{153}Gd	4.79E-02	1.11E+01	9.22E-29	0.	0.	0.	0.	0.	0.	0.	0.
^{160}Tb	2.21E-01	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
^{181}W	1.08E-03	3.48E-08	0.	0.	0.	0.	0.	0.	0.	0.	0.
TOTAL	1.33E+10	4.74E+10	2.94E+10	2.58E+07	8.29E+06	7.74E+06	7.53E+06	6.45E+06	5.46E+06	2.09E+06	1.11E+06

a. Values less than 1.0E-29 have been designated as zero.

RQ VALUES FOR SPENT FUEL
(DOSE TO RIVER WATER USER 5 MREM/YEAR)

YEAR 2050

NUCLIDE	INVENTORY (CI)	RQ
SR--90	1.0E+10	6.1E+18
CS-137	1.4E+10	2.7E+17
AM-241	1.2E+09	5.3E+16
Y---90	1.0E+10	4.9E+16
PU-238	5.9E+08	1.7E+16
PU-240	1.7E+08	5.4E+15
SUB-TOTAL		6.5E+18
PU-241	7.8E+09	4.9E+15
PU-239	1.1E+08	3.5E+15
CM-244	1.2E+08	3.3E+15
EU-154	4.5E+08	1.2E+15
CS-134	4.1E+07	1.2E+15
NI--63	1.4E+08	6.8E+14
AM-243	5.1E+06	2.4E+14
AM242M	3.4E+06	1.6E+14
CO--60	4.2E+07	1.2E+14
SM-151	3.2E+08	8.1E+13
SB-125	5.9E+06	3.9E+13
PM-147	8.3E+07	3.6E+13
CM-243	6.5E+05	2.3E+13
PU-242	5.9E+05	1.8E+13
U--234	4.4E+05	1.8E+13
FE--55	5.3E+06	1.7E+13
CM-242	2.8E+06	1.3E+13
RU-106	1.1E+06	9.5E+12
NP-237	1.3E+05	7.9E+12
NP-239	5.1E+06	5.9E+12
U--238	1.2E+05	5.1E+12
TE125M	5.9E+06	4.8E+12
CM-245	6.5E+04	3.4E+12
U--236	8.5E+04	3.3E+12
CE-144	2.5E+05	2.0E+12
TC--99	4.8E+06	1.6E+12
U--232	5.1E+03	9.7E+11
EU-152	6.7E+05	8.3E+11
SB-126	1.8E+05	8.3E+11
NI--59	1.2E+06	8.0E+11
SN-126	1.8E+05	7.5E+11
CD113M	5.3E+05	7.0E+11
I--129	1.2E+04	5.6E+11
C---14	3.2E+05	6.3E+10
TOTAL		6.5E+18

YEAR 2070

NUCLIDE	INVENTORY (CI)	RQ
SR--90	6.2E+09	3.7E+18
CS-137	9.1E+09	1.7E+17
AM-241	1.3E+09	5.9E+16
Y---90	6.2E+09	3.0E+16
PU-238	5.4E+08	1.5E+16
PU-240	1.7E+08	5.4E+15
SUB-TOTAL		4.0E+18
PU-239	1.1E+08	3.5E+15
PU-241	3.0E+09	1.9E+15
CM-244	5.5E+07	1.6E+15
NI--63	1.2E+08	5.8E+14
EU-154	1.9E+08	5.1E+14
AM-243	5.1E+06	2.4E+14
AM242M	3.1E+06	1.5E+14
SM-151	2.7E+08	6.9E+13
U--234	4.7E+05	1.9E+13
PU-242	5.9E+05	1.8E+13
CM-243	4.2E+05	1.5E+13
CM-242	2.5E+06	1.2E+13
CO--60	3.0E+06	9.7E+12
NP-237	1.3E+05	8.4E+12
NP-239	5.1E+06	5.8E+12
U--238	1.2E+05	5.1E+12
CM-245	6.5E+04	3.4E+12
U--236	8.5E+04	3.3E+12
TC--99	4.8E+06	1.6E+12
CS-134	4.7E+04	1.4E+12
SB-126	1.8E+05	8.3E+11
U--232	4.2E+03	8.2E+11
NI--59	1.2E+06	8.0E+11
SN-126	1.8E+05	7.5E+11
TH-234	1.2E+05	6.5E+11
I--129	1.2E+04	5.6E+11
RA-226	4.3E+03	5.6E+11
CD113M	2.1E+05	2.7E+11
U--235	6.7E+03	2.7E+11
EU-152	2.1E+05	2.6E+11
CS-135	1.0E+05	2.6E+11
SB-125	3.5E+04	2.3E+11
TH-229	4.3E+03	2.0E+11
C---14	3.2E+05	6.3E+10
TOTAL		4.0E+18

RQ VALUES FOR SPENT FUEL
(DOSE TO RIVER WATER USER 5 MREM/YEAR)

500 YEARS AFTER 1975

NUCLIDE	INVENTORY (CI)	RQ
AM-241	7.2E+08	3.4E+16
PU-240	1.6E+08	5.2E+15
PU-239	1.1E+08	3.5E+15
PU-238	2.2E+07	6.5E+14
AM-243	4.9E+06	2.3E+14
SR--90	2.9E+05	1.8E+14
SUB-TOTAL		4.4E+16
NI--63	5.6E+06	2.8E+13
U--234	6.3E+05	2.6E+13
AM242M	4.8E+05	2.4E+13
PU-242	5.9E+05	1.8E+13
NP-237	2.5E+05	1.6E+13
CS-137	8.1E+05	1.5E+13
RA-226	1.7E+02	9.1E+12
NP-239	4.9E+06	5.6E+12
U--238	1.2E+05	5.1E+12
U--236	8.6E+04	3.4E+12
CM-245	6.3E+04	3.3E+12
SM-151	1.1E+07	2.8E+12
CM-242	4.0E+05	1.9E+12
TC--99	4.8E+06	1.6E+12
Y---90	2.9E+05	1.4E+12
SB-126	1.8E+05	8.2E+11
NI--59	1.1E+06	7.9E+11
SN-126	1.8E+05	7.5E+11
TH-234	1.2E+05	6.5E+11
I--129	1.2E+04	5.6E+11
U--235	6.8E+03	2.7E+11
CS-135	1.0E+05	2.6E+11
PA-233	2.5E+05	2.0E+11
TH-230	2.1E+03	1.8E+11
PB-210	1.6E+02	1.2E+11
NB-93M	6.2E+05	1.2E+11
ZR--93	6.2E+05	7.3E+10
C---14	3.0E+05	6.0E+10
PU-241	6.4E+04	4.0E+10
SE--79	1.3E+05	2.6E+10
RA-223	6.8E+01	2.5E+10
U--232	8.6E+01	1.6E+10
U--233	3.7E+02	1.6E+10
PA-231	7.1E+01	1.3E+10
TOTAL		4.4E+16

1000 YEARS AFTER 1975

NUCLIDE	INVENTORY (CI)	RQ
AM-241	3.2E+08	1.5E+16
PU-240	1.5E+08	4.9E+15
PU-239	1.1E+08	3.4E+15
AM-243	4.7E+06	2.2E+14
SUB-TOTAL		2.4E+16
RA-226	8.7E+02	4.6E+13
U--234	6.5E+05	2.7E+13
NP-237	3.4E+05	2.2E+13
PU-242	5.9E+05	1.8E+13
PU-238	5.3E+05	1.5E+13
NP-239	4.7E+06	5.4E+12
U--238	1.2E+05	5.1E+12
U--236	8.9E+04	3.5E+12
CM-245	6.0E+04	3.2E+12
AM242M	5.0E+04	2.4E+12
TC--99	4.8E+06	1.6E+12
SB-126	1.8E+05	8.2E+11
NI--59	1.1E+06	7.9E+11
SN-126	1.8E+05	7.4E+11
PB-210	8.7E+02	6.7E+11
TH-234	1.2E+05	6.5E+11
NI--63	1.3E+05	6.5E+11
I--129	1.2E+04	5.6E+11
TH-230	5.0E+03	4.3E+11
PA-233	3.4E+05	2.7E+11
U--235	6.8E+03	2.7E+11
CS-135	1.0E+05	2.6E+11
CM-242	4.1E+04	1.9E+11
NB-93M	6.2E+05	1.2E+11
ZR--93	6.2E+05	7.3E+10
PO-210	8.7E+02	6.1E+10
C---14	2.8E+05	5.6E+10
RA-223	1.7E+02	5.3E+10
SM-151	2.0E+05	5.2E+10
CM-243	1.3E+03	4.7E+10
U--233	1.1E+03	4.5E+10
PU-241	6.0E+04	3.8E+10
PA-231	1.5E+02	2.7E+10
SE--79	1.3E+05	2.6E+10
RA-225	4.1E+01	1.9E+10
TH-229	4.1E+01	1.4E+10
TOTAL		2.4E+16

RQ VALUES FOR SPENT FUEL
(DOSE TO RIVER WATER USER 5 MREM/YEAR)

5000 YEARS AFTER 1975

NUCLIDE	INVENTORY (CI)	RQ
PU-240	1.0E+08	3.3E+15
PU-239	9.6E+07	3.1E+15
RA-226	1.6E+04	8.4E+14
AM-243	3.3E+06	1.5E+14
AM-241	5.8E+05	2.7E+13
U--234	6.5E+05	2.6E+13
NP-237	4.1E+05	2.6E+13
PU-242	5.9E+05	1.8E+13
PB-210	1.6E+04	1.2E+13
U--238	1.2E+05	5.1E+12
SUB-TOTAL		7.5E+15
U--236	1.0E+05	4.0E+12
NP-239	3.3E+06	3.7E+12
TH-230	2.7E+04	2.3E+12
CM-245	4.3E+04	2.3E+12
TC--99	4.7E+06	1.6E+12
PO-210	1.6E+04	1.1E+12
SB-126	1.7E+05	8.0E+11
NI--59	1.1E+06	7.6E+11
SN-126	1.7E+05	7.2E+11
RA-225	1.5E+03	7.0E+11
TH-234	1.2E+05	6.5E+11
I--129	1.2E+04	5.6E+11
TH-229	1.5E+03	5.2E+11
U--233	7.8E+03	3.3E+11
PA-233	4.1E+05	3.2E+11
U--235	7.2E+03	2.9E+11
RA-223	7.0E+02	2.6E+11
CS-135	1.0E+05	2.6E+11
PA-231	7.0E+02	1.3E+11
NB-93M	6.2E+05	1.2E+11
ZR--93	6.2E+05	7.3E+10
AC-227	7.0E+02	6.1E+10
BI-210	1.6E+04	4.4E+10
C---14	1.7E+05	3.5E+10
AC-225	1.5E+03	2.9E+10
PU-241	4.3E+04	2.7E+10
SE--79	1.2E+05	2.5E+10
TH-227	7.0E+02	1.9E+10
PD-107	3.7E+04	2.3E+09
RA-228	2.3E-02	6.4E+08
TOTAL		7.5E+15

10,000 YEARS AFTER 1975

NUCLIDE	INVENTORY (CI)	RQ
PU-239	9.4E+07	2.7E+15
RA-226	4.1E+04	2.2E+15
PU-240	6.1E+07	2.0E+15
AM-243	2.1E+06	9.8E+13
PB-210	4.1E+04	3.2E+13
U--234	6.4E+05	2.6E+13
NP-237	4.1E+05	2.6E+13
PU-242	5.8E+05	1.8E+13
U--238	1.2E+05	5.1E+12
SUB-TOTAL		7.1E+15
TH-230	5.3E+04	4.5E+12
U--236	1.2E+05	4.5E+12
PO-210	4.1E+04	2.9E+12
RA-225	5.6E+03	2.7E+12
NP-239	2.1E+06	2.4E+12
TH-229	5.6E+03	2.0E+12
TC--99	4.7E+06	1.6E+12
CM-245	2.8E+04	1.5E+12
AM-241	2.9E+04	1.3E+12
SB-126	1.7E+05	7.7E+11
NI--59	1.1E+06	7.3E+11
SN-126	1.7E+05	7.0E+11
U--233	1.6E+04	6.9E+11
TH-234	1.2E+05	6.5E+11
I--129	1.2E+04	5.6E+11
RA-223	1.4E+03	5.1E+11
PA-233	4.1E+05	3.2E+11
U--235	7.7E+03	3.0E+11
PA-231	1.4E+03	2.6E+11
CS-135	1.0E+05	2.5E+11
AC-227	1.4E+03	1.2E+11
BI-210	4.1E+04	1.2E+11
NB-93M	6.2E+05	1.2E+11
AC-225	5.6E+03	1.1E+11
ZR--93	6.2E+05	7.3E+10
TH-227	1.4E+03	3.7E+10
SE--79	1.2E+05	2.4E+10
C---14	9.5E+04	1.9E+10
PU-241	2.9E+04	1.8E+10
PD-107	3.7E+04	2.3E+09
RA-228	5.0E-02	1.4E+09
TOTAL		7.1E+15

RQ VALUES FOR SPENT FUEL
(DOSE TO RIVER WATER USER 5 MREM/YEAR)

50,000 YEARS AFTER 1975

NUCLIDE	INVENTORY (CI)	RQ
RA-226	2.2E+05	1.2E+16
PU-239	2.7E+07	8.7E+14
PB-210	2.2E+05	1.7E+14
SUB-TOTAL		1.3E+16
PU-240	1.0E+06	3.2E+13
RA-225	6.3E+04	3.0E+13
NP-237	4.1E+05	2.6E+13
U--234	5.8E+05	2.4E+13
TH-229	6.3E+04	2.2E+13
TH-230	2.2E+05	1.8E+13
PU-242	5.4E+05	1.6E+13
PO-210	2.2E+05	1.5E+13
U--236	1.3E+05	5.2E+12
U--238	1.2E+05	5.1E+12
U--233	7.8E+04	3.3E+12
AM-243	5.5E+04	2.6E+12
RA-223	5.7E+03	2.1E+12
TC--99	4.1E+06	1.4E+12
AC-225	6.3E+04	1.2E+12
PA-231	5.7E+03	1.1E+12
TH-234	1.2E+05	6.5E+11
BI-210	2.2E+05	6.0E+11
SB-126	1.3E+05	5.9E+11
I--129	1.2E+04	5.6E+11
SN-126	1.3E+05	5.3E+11
NI--59	7.5E+05	5.2E+11
AC-227	5.7E+03	4.9E+11
U--235	9.6E+03	3.8E+11
PA-233	4.1E+05	3.2E+11
CS-135	9.9E+04	2.5E+11
TH-227	5.7E+03	1.5E+11
NB-93M	6.1E+05	1.1E+11
ZR--93	6.1E+05	7.2E+10
NP-239	5.5E+04	6.3E+10
CM-245	9.9E+02	5.2E+10
AM-241	9.9E+02	4.6E+10
SE--79	7.7E+04	1.5E+10
RA-229	3.0E-01	8.3E+09
PD-107	3.6E+04	2.3E+09
PU-241	9.9E+02	6.2E+08
C---14	7.6E+02	1.5E+08
TOTAL		1.3E+16

100,000 YEARS AFTER 1975

NUCLIDE	INVENTORY (CI)	RQ
RA-226	3.4E+05	1.8E+16
PB-210	3.4E+05	2.6E+14
PU-239	6.6E+06	2.1E+14
RA-225	1.3E+05	6.1E+13
SUB-TOTAL		1.9E+16
TH-229	1.3E+05	4.6E+13
TH-230	3.3E+05	2.8E+13
NP-237	4.0E+05	2.5E+13
PO-210	3.4E+05	2.4E+13
U--234	5.2E+05	2.1E+13
PU-242	4.9E+05	1.5E+13
U--233	1.4E+05	5.9E+12
U--236	1.3E+05	5.2E+12
U--238	1.2E+05	5.1E+12
RA-223	8.6E+03	3.1E+12
AC-225	1.3E+05	2.6E+12
PA-231	8.6E+03	1.6E+12
TC--99	3.5E+06	1.2E+12
BI-210	3.4E+05	9.4E+11
AC-227	8.6E+03	7.4E+11
TH-234	1.2E+05	6.5E+11
I--129	1.2E+04	5.6E+11
SB-126	8.9E+04	4.1E+11
U--235	1.0E+04	4.1E+11
SN-126	8.9E+04	3.7E+11
NI--59	4.8E+05	3.3E+11
PA-233	4.0E+05	3.2E+11
CS-135	9.8E+04	2.5E+11
TH-227	8.6E+03	2.3E+11
PU-240	6.0E+03	1.9E+11
NB-93M	6.0E+05	1.1E+11
ZR--93	6.0E+05	7.0E+10
AM-243	6.0E+02	2.9E+10
RA-229	6.3E-01	1.7E+10
SE--79	4.5E+04	9.0E+09
PD-107	3.6E+04	2.3E+09
CM-245	1.5E+01	7.8E+08
AM-241	1.5E+01	6.9E+08
NP-239	6.0E+02	6.9E+08
RA-224	6.7E-01	9.2E+07
C---14	1.8E+00	7.6E+05
TOTAL		1.9E+16

RQ VALUES FOR SPENT FUEL
(DOSE TO RIVER WATER USER 5 MREM/YEAR)

500,000 YEARS AFTER 1975

NUCLIDE	INVENTORY (CI)	RQ
RA-226	3.0E+05	1.6E+16
PB-210	3.0E+05	2.3E+14
RA-225	3.3E+05	1.6E+14
TH-229	3.3E+05	1.2E+14
SUB-TOTAL		1.7E+16
TH-230	3.0E+05	2.6E+13
NP-237	3.5E+05	2.2E+13
PO-210	3.0E+05	2.1E+13
U--233	3.3E+05	1.4E+13
U--234	2.5E+05	1.0E+13
PU-242	2.4E+05	7.2E+12
AC-225	3.3E+05	6.5E+12
U--236	1.3E+05	5.1E+12
U--238	1.2E+05	5.1E+12
RA-223	1.1E+04	3.8E+12
PA-231	1.1E+04	2.0E+12
AC-227	1.1E+04	9.1E+11
BI-210	3.0E+05	8.5E+11
TH-234	1.2E+05	6.5E+11
I--129	1.2E+04	5.5E+11
U--235	1.1E+04	4.2E+11
TC--99	9.3E+05	3.1E+11
TH-227	1.1E+04	2.8E+11
PA-233	3.5E+05	2.8E+11
SB-126	5.6E+04	2.6E+11
SN-126	5.6E+04	2.3E+11
CS-135	8.9E+04	2.3E+11
NB-93M	4.9E+05	9.2E+10
RA-228	3.2E+00	8.8E+10
ZR--93	4.9E+05	5.8E+10
PU-239	7.6E+02	2.4E+10
NI--59	1.5E+04	1.0E+10
PD-107	3.5E+04	2.2E+09
RA-224	3.2E+00	4.2E+08
TH-232	3.2E+00	2.5E+08
TH-228	3.2E+00	1.5E+08
SE--79	6.4E+02	1.3E+08
RB--87	6.5E+00	6.7E+06
TOTAL		1.7E+16

1,000,000 YEARS AFTER 1975

NUCLIDE	INVENTORY (CI)	RQ
RA-226	1.7E+05	9.0E+15
RA-225	3.2E+05	1.5E+14
PB-210	1.7E+05	1.3E+14
TH-229	3.2E+05	1.1E+14
NP-237	7.0E+05	1.9E+13
TH-230	1.7E+05	1.4E+13
U--233	3.2E+05	1.3E+13
PO-210	1.7E+05	1.2E+13
AC-225	3.2E+05	6.3E+12
U--234	1.5E+05	6.2E+12
U--238	1.2E+05	5.1E+12
SUB-TOTAL		9.5E+15
U--236	1.3E+05	5.0E+12
RA-223	1.1E+04	3.8E+12
PU-242	9.5E+04	2.9E+12
PA-231	1.1E+04	2.0E+12
AC-227	1.1E+04	9.1E+11
TH-234	1.2E+05	6.5E+11
I--129	1.2E+04	5.4E+11
BI-210	1.7E+05	4.7E+11
U--235	1.1E+04	4.2E+11
TH-227	1.1E+04	2.8E+11
PA-233	3.0E+05	2.4E+11
CS-135	8.0E+04	2.0E+11
RA-228	6.4E+00	1.8E+11
NB-93M	3.9E+05	7.3E+10
TC--99	1.8E+05	6.0E+10
ZR--93	3.9E+05	4.6E+10
PD-107	3.3E+04	2.1E+09
RA-224	6.4E+00	8.3E+08
SB-126	1.8E+02	8.1E+08
SN-126	1.8E+02	7.4E+08
TH-232	6.4E+00	4.9E+08
TH-228	6.4E+00	2.9E+08
NI--59	2.0E+02	1.4E+08
RB--87	6.5E+00	6.7E+06
SE--79	3.1E+00	6.2E+05
TOTAL		9.5E+15

TABLE I-6

RQ VALUES FOR HLW
(DOSE TO RIVER WATER USER 5 MREM/YEAR)

YEAR 2050

NUCLIDE	INVENTORY (CI)	RQ
SR--90	8.8E+09	5.3E+18
CS-137	1.4E+10	2.7E+17
Y---90	8.8E+09	4.3E+16
CM-244	1.4E+09	3.8E+16
AM-241	5.6E+08	2.6E+16
SUB-TOTAL		5.7E+18
EU-154	5.3E+08	1.4E+15
AM242M	2.7E+07	1.3E+15
AM-243	2.7E+07	1.3E+15
CS-134	4.0E+07	1.2E+15
PU-238	3.0E+07	8.8E+14
NI--63	1.2E+08	5.8E+14
PU-240	1.1E+07	3.4E+14
PU-241	2.1E+08	1.3E+14
CO--60	3.8E+07	1.1E+14
CM-242	2.3E+07	1.1E+14
SM-151	3.4E+08	8.7E+13
CM-245	1.3E+06	6.6E+13
CM-243	1.7E+06	6.1E+13
PU-239	1.6E+06	5.2E+13
SB-125	7.1E+06	4.7E+13
PM-147	8.2E+07	3.5E+13
NP-239	2.7E+07	3.1E+13
FE--55	4.2E+06	1.4E+13
NP-237	2.1E+05	1.3E+13
RU-106	1.2E+06	1.1E+13
U--232	3.6E+04	6.9E+12
TE125M	7.1E+06	5.7E+12
RA-224	4.2E+04	5.6E+12
TH-228	4.2E+04	1.9E+12
CE-144	2.4E+05	1.9E+12
TC--99	4.8E+06	1.6E+12
CD113M	1.1E+06	1.4E+12
EU-152	1.1E+06	1.4E+12
SB-126	2.1E+05	9.7E+11
SN-126	2.1E+05	8.8E+11
NI--59	9.9E+05	6.9E+11
PU-242	2.2E+04	6.8E+11
I--129	1.3E+04	6.0E+11
EU-155	8.8E+05	4.1E+11
C---14	3.0E+05	6.0E+10
TOTAL		5.7E+18

YEAR 2070

NUCLIDE	INVENTORY (CI)	RQ
SR--90	5.4E+09	3.2E+18
CS-137	9.0E+09	1.7E+17
Y---90	5.4E+09	2.6E+16
AM-241	5.4E+08	2.5E+16
CM-244	6.3E+08	1.8E+16
SUB-TOTAL		3.4E+18
AM-243	2.7E+07	1.3E+15
AM242M	2.5E+07	1.2E+15
PU-238	2.9E+07	8.4E+14
EU-154	2.2E+08	6.0E+14
NI--63	1.0E+08	5.0E+14
PU-240	1.7E+07	4.0E+14
CM-242	2.1E+07	9.7E+13
SM-151	2.9E+08	7.4E+13
CM-245	1.2E+06	6.5E+13
PU-239	1.7E+06	5.3E+13
PU-241	9.1E+07	5.1E+13
CM-243	1.1E+06	4.0E+13
NP-239	2.7E+07	3.1E+13
NP-237	2.1E+05	1.3E+13
CO--60	2.7E+06	7.9E+12
U--232	3.0E+04	5.7E+12
RA-224	3.1E+04	4.0E+12
TC--99	4.8E+06	1.6E+12
TH-228	3.1E+04	1.4E+12
CS-134	4.6E+04	1.3E+12
SB-126	2.1E+05	9.7E+11
SN-126	2.1E+05	8.8E+11
PU-242	2.3E+04	6.9E+11
NI--59	9.9E+05	6.8E+11
I--129	1.3E+04	6.0E+11
CD113M	4.0E+05	5.3E+11
EU-152	7.5E+05	4.5E+11
U--234	8.3E+03	3.4E+11
CS-135	1.3E+05	3.3E+11
SB-125	4.2E+04	2.8E+11
PM-147	4.1E+05	1.8E+11
PA-233	2.1E+05	1.7E+11
NB-93M	6.2E+05	1.1E+11
ZR--93	6.2E+05	7.3E+10
C---14	3.0E+05	5.9E+10
TOTAL		3.4E+18

RQ VALUES FOR HLW
(DOSE TO RIVER WATER USER 5 MREM/YEAR)

500 YEARS AFTER 1975

NUCLIDE	INVENTORY (CI)	RQ
AM-241	2.9E+08	1.3E+16
AM-243	2.6E+07	1.2E+15
PU-240	1.4E+07	4.4E+14
AM242M	4.0E+06	1.9E+14
PU-238	6.5E+06	1.9E+14
SR--90	2.5E+05	1.5E+14
CM-245	1.2E+06	6.3E+13
PU-239	1.9E+06	6.1E+13
SUB-TOTAL		1.5E+16
NP-239	2.6E+07	3.0E+13
NI--63	4.8E+06	2.4E+13
NP-237	2.6E+05	1.6E+13
CM-242	3.3E+06	1.5E+13
CS-137	8.0E+05	1.5E+13
SM-151	1.2E+07	3.0E+12
TC--99	4.8E+06	1.6E+12
Y---90	2.5E+05	1.2E+12
U--234	2.9E+04	1.0E+12
SB-126	2.1E+05	9.7E+11
SN-126	2.1E+05	8.8E+11
PU-241	1.2E+06	7.6E+11
PU-242	2.4E+04	7.3E+11
NI--59	9.9E+05	6.8E+11
I--129	1.3E+04	6.0E+11
RA-226	7.9E+00	4.2E+11
CS-135	1.3E+05	3.3E+11
PA-233	2.6E+05	2.0E+11
U--232	6.1E+02	1.2E+11
NB-93M	6.2E+05	1.1E+11
RA-224	6.3E+02	8.2E+10
ZR--93	6.2E+05	7.3E+10
C---14	2.8E+05	5.7E+10
U--236	1.1E+03	4.4E+10
U--238	9.9E+02	4.2E+10
TH-228	6.3E+02	2.9E+10
SE--79	1.2E+05	2.5E+10
U--233	4.2E+02	1.8E+10
TH-230	8.3E+01	7.0E+09
CM-243	1.8E+02	6.3E+09
PB-210	7.5E+00	5.8E+09
TH-234	9.9E+02	5.4E+09
TOTAL		1.5E+16

1000 YEARS AFTER 1975

NUCLIDE	INVENTORY (CI)	RQ
AM-241	1.7E+08	6.0E+15
AM-243	2.5E+07	1.2E+15
PU-240	1.3E+07	4.2E+14
PU-239	2.3E+06	7.2E+13
CM-245	1.2E+06	6.0E+13
NP-239	2.5E+07	2.8E+13
PU-238	7.8E+05	2.2E+13
AM242M	4.1E+05	2.0E+13
NP-237	2.9E+05	1.8E+13
SUB-TOTAL		7.8E+15
RA-226	3.7E+01	2.0E+12
TC--99	4.8E+06	1.6E+12
CM-242	3.3E+05	1.6E+12
U--234	3.1E+04	1.2E+12
SB-126	2.1E+05	9.6E+11
SN-126	2.1E+05	8.7E+11
PU-242	2.4E+04	7.4E+11
PU-241	1.2E+06	7.3E+11
NI--59	9.8E+05	6.8E+11
I--129	1.3E+04	6.0E+11
NI--63	1.1E+05	5.6E+11
CS-135	1.3E+05	3.2E+11
PA-233	2.9E+05	2.3E+11
NB-93M	6.2E+05	1.1E+11
ZR--93	6.2E+05	7.3E+10
SM-151	2.2E+05	6.5E+10
C---14	2.7E+05	5.3E+10
U--236	1.3E+03	5.2E+10
U--233	1.0E+03	4.4E+10
U--238	9.9E+02	4.2E+10
PB-210	3.7E+01	2.8E+10
SE--79	1.2E+05	2.5E+10
RA-225	4.3E+01	2.0E+10
TH-230	2.2E+02	1.8E+10
TH-229	4.3E+01	1.5E+10
TH-234	9.9E+02	5.4E+09
RA-223	1.3E+01	4.7E+09
PD-107	4.6E+04	2.9E+09
PO-210	3.7E+01	2.6E+09
PA-231	1.3E+01	2.4E+09
U--235	5.4E+01	2.1E+09
TOTAL		7.9E+15

RQ VALUES FOR HLW
(DOSE TO RIVER WATER USER 5 MREM/YEAR)

5000 YEARS AFTER 1975

NUCLIDE	INVENTORY (CI)	RQ
AM-243	1.7E+07	8.2E+14
PU-240	8.7E+06	2.8E+14
PU-239	4.2E+06	1.3E+14
AM-241	1.1E+06	5.0E+13
CM-245	8.3E+05	4.3E+13
RA-226	7.3E+02	3.9E+13
NP-237	3.2E+05	2.0E+13
NP-239	1.7E+07	2.0E+13
SUB-TOTAL		1.4E+15
TC--99	4.7E+06	1.6E+12
U--234	3.1E+04	1.3E+12
SB-126	2.0E+05	9.4E+11
SN-126	2.0E+05	8.5E+11
PU-242	2.4E+04	7.3E+11
NI--59	9.5E+05	6.6E+11
I--129	1.3E+04	6.0E+11
RA-225	1.2E+03	5.9E+11
PB-210	7.3E+02	5.6E+11
PU-241	8.3E+05	5.2E+11
TH-229	1.2E+03	4.4E+11
CS-135	1.3E+05	3.2E+11
U--233	6.4E+03	2.7E+11
PA-233	3.2E+05	2.5E+11
NB-93M	6.2E+05	1.1E+11
TH-230	1.3E+03	1.1E+11
U--236	2.6E+03	1.0E+11
ZR--93	6.2E+05	7.3E+10
PO-210	7.3E+02	5.1E+10
U--238	9.9E+02	4.2E+10
C---14	1.7E+05	3.3E+10
AC-225	1.2E+03	2.5E+10
SE--79	1.2E+05	2.4E+10
RA-223	1.7E+01	6.1E+09
TH-234	9.9E+02	5.4E+09
PA-231	1.7E+01	3.1E+09
PD-107	4.6E+04	2.9E+09
U--235	6.7E+01	2.7E+09
BI-210	7.3E+02	2.0E+09
AC-227	1.7E+01	1.4E+09
TH-227	1.7E+01	4.4E+08
RB--87	5.7E+00	5.9E+06
TOTAL		1.4E+15

10,000 YEARS AFTER 1975

NUCLIDE	INVENTORY (CI)	RQ
AM-243	1.1E+07	5.2E+14
PU-239	5.5E+06	1.8E+14
PU-240	5.2E+06	1.7E+14
RA-226	2.0E+03	1.0E+14
CM-245	5.4E+05	2.9E+13
AM-241	5.4E+05	2.5E+13
NP-237	3.2E+05	2.0E+13
NP-239	1.1E+07	1.3E+13
SUB-TOTAL		1.1E+15
RA-225	4.6E+03	2.2E+12
TH-229	4.6E+03	1.6E+12
TC--99	4.6E+06	1.6E+12
PB-210	2.0E+03	1.5E+12
U--234	3.1E+04	1.2E+12
SB-126	2.0E+05	9.1E+11
SN-126	2.0E+05	8.2E+11
PU-242	2.4E+04	7.3E+11
NI--59	9.1E+05	6.3E+11
I--129	1.3E+04	6.0E+11
U--233	1.3E+04	5.5E+11
PU-241	5.4E+05	3.4E+11
CS-135	1.3E+05	3.2E+11
PA-233	3.2E+05	2.5E+11
TH-230	2.5E+03	2.1E+11
U--236	3.6E+03	1.4E+11
PO-210	2.0E+03	1.4E+11
NB-93M	6.1E+05	1.1E+11
AC-225	4.6E+03	9.1E+10
ZR--93	6.1E+05	7.2E+10
U--238	9.9E+02	4.2E+10
SE--79	1.1E+05	2.2E+10
C---14	9.0E+04	1.8E+10
RA-223	2.3E+01	8.4E+09
BI-210	2.0E+03	5.5E+09
TH-234	9.9E+02	5.4E+09
PA-231	2.7E+01	4.3E+09
U--235	9.1E+01	3.6E+09
PD-107	4.6E+04	2.9E+09
AC-227	2.7E+01	2.0E+09
TH-227	2.7E+01	6.1E+08
RB--87	5.7E+00	5.9E+06
TOTAL		1.1E+15

RQ VALUES FOR HLW
(DOSE TO RIVER WATER USER 5 MREM/YEAR)

50,000 YEARS AFTER 1975

NUCLIDE	INVENTORY (CI)	RQ
RA-226	1.0E+04	5.5E+14
PU-239	3.2E+06	1.0E+14
RA-225	5.0E+04	2.4E+13
NP-237	3.2E+05	2.0E+13
TH-229	5.0E+04	1.9E+13
AM-243	2.9E+05	1.4E+13
PB-210	1.0E+04	7.9E+12
PU-240	8.6E+04	2.7E+12
U--233	6.2E+04	2.6E+12
TC--99	4.1E+06	1.4E+12
U--234	2.7E+04	1.1E+12
CM-245	1.9E+04	9.9E+11
AC-225	5.0E+04	9.8E+11
AM-241	1.9E+04	8.8E+11
TH-230	1.0E+04	8.7E+11
PO-210	1.0E+04	7.2E+11
SB-126	1.5E+05	6.9E+11
PU-242	2.2E+04	6.8E+11
SN-126	1.5E+05	6.2E+11
I--129	1.3E+04	6.0E+11
SUB-TOTAL		7.5E+14
NI--59	6.4E+05	4.4E+11
NP-239	2.9E+05	3.4E+11
CS-135	1.3E+05	3.2E+11
PA-233	3.2E+05	2.5E+11
U--236	5.0E+03	2.0E+11
NB-93M	6.0E+05	1.1E+11
ZR--93	6.0E+05	7.1E+10
RA-223	1.3E+02	4.8E+10
U--238	9.9E+02	4.2E+10
BI-210	1.0E+04	2.9E+10
PA-231	1.3E+02	2.4E+10
SE--79	7.3E+04	1.5E+10
PU-241	1.9E+04	1.2E+10
U--235	2.8E+02	1.1E+10
AC-227	1.3E+02	1.1E+10
TH-234	9.9E+02	5.4E+09
TH-227	1.3E+02	3.4E+09
PD-107	4.6E+04	2.9E+09
C---14	7.2E+02	1.4E+08
RB--87	5.7E+00	5.9E+06
TOTAL		7.5E+14

100,000 YEARS AFTER 1975

NUCLIDE	INVENTORY (CI)	RQ
RA-226	1.6E+04	9.4E+14
RA-225	1.0E+05	4.8E+13
TH-229	1.0E+05	3.6E+13
PU-239	8.1E+05	2.6E+13
NP-237	3.2E+05	2.0E+13
PB-210	1.6E+04	1.2E+13
U--233	1.1E+05	4.7E+12
AC-225	1.0E+05	2.0E+12
TH-230	1.6E+04	1.3E+12
TC--99	3.4E+06	1.2E+12
PO-210	1.6E+04	1.1E+12
U--234	2.4E+04	9.8E+11
PU-242	2.0E+04	6.2E+11
I--129	1.3E+04	6.0E+11
SUB-TOTAL		9.9E+14
SB-126	1.1E+05	4.9E+11
SN-126	1.1E+05	4.4E+11
CS-135	1.3E+05	3.2E+11
NI--59	4.2E+05	2.9E+11
PA-233	3.2E+05	2.5E+11
U--236	5.0E+03	2.0E+11
AM-243	3.1E+03	1.5E+11
NB-93M	5.9E+05	1.1E+11
RA-223	2.7E+02	9.9E+10
ZR--93	5.9E+05	6.9E+10
PA-231	2.7E+02	5.1E+10
BI-210	1.6E+04	4.4E+10
U--238	9.9E+02	4.2E+10
AC-227	2.7E+02	2.3E+10
PU-240	5.1E+02	1.6E+10
CM-245	2.9E+02	1.5E+10
U--235	3.7E+02	1.5E+10
AM-241	2.9E+02	1.3E+10
SE--79	4.3E+04	8.6E+09
TH-227	2.7E+02	7.1E+09
TH-234	9.9E+02	5.4E+09
NP-239	3.1E+03	3.6E+09
PD-107	4.6E+04	2.9E+09
PU-241	2.9E+02	1.8E+08
RB--87	5.7E+00	5.9E+06
C---14	1.7E+00	3.4E+05
TOTAL		1.0E+15

RQ VALUES FOR HLW
(DOSE TO RIVER WATER USER = MREM/YEAR)

500,000 YEARS AFTER 1975

NUCLIDE	INVENTORY (CI)	RQ
RA-226	1.1E+04	6.1E+14
RA-225	2.6E+05	1.2E+14
TH-229	2.6E+05	9.2E+13
NP-237	2.8E+05	1.7E+13
U--233	2.6E+05	1.1E+13
PB-210	1.1E+04	8.8E+12
AC-225	2.6E+05	5.1E+12
TH-230	1.1E+04	9.7E+11
PO-210	1.1E+04	8.0E+11
I--129	1.3E+04	5.9E+11
SUB-TOTAL		8.7E+14
U--234	8.5E+03	3.4E+11
TC--99	9.2E+05	3.1E+11
PU-242	9.7E+03	3.0E+11
CS-135	1.1E+05	2.9E+11
PA-233	2.8E+05	2.2E+11
U--236	5.0E+03	1.9E+11
RA-223	4.0E+02	1.5E+11
NB-93M	4.9E+05	9.1E+10
PA-231	4.0E+02	7.5E+10
ZR--93	4.9E+05	5.8E+10
U--238	9.9E+02	4.2E+10
AC-227	4.0E+02	3.4E+10
BI-210	1.1E+04	3.2E+10
SB-126	6.6E+03	3.0E+10
SN-126	6.6E+03	2.8E+10
U--235	4.0E+02	1.6E+10
TH-227	4.0E+02	1.1E+10
NI--59	1.3E+04	9.0E+09
TH-234	9.9E+02	5.4E+09
PD-107	4.4E+04	2.8E+09
PU-239	9.4E+00	3.0E+08
SE--79	6.0E+02	1.2E+08
RB--87	5.7E+00	5.9E+06
TOTAL		8.7E+14

1,000,000 YEARS AFTER 1975

NUCLIDE	INVENTORY (CI)	RQ
RA-226	3.7E+03	2.0E+14
RA-225	2.5E+05	1.2E+14
TH-229	2.5E+05	8.9E+13
NP-237	2.4E+05	1.5E+13
U--233	2.5E+05	1.1E+13
AC-225	2.5E+05	5.0E+12
PB-210	3.7E+03	2.9E+12
I--129	1.3E+04	5.8E+11
SUB-TOTAL		4.4E+14
TH-230	3.7E+03	3.1E+11
PO-210	3.7E+03	2.6E+11
CS-135	1.0E+05	2.6E+11
U--236	4.9E+03	1.9E+11
PA-233	2.4E+05	1.9E+11
RA-223	4.0E+02	1.5E+11
PU-242	3.9E+03	1.2E+11
U--234	2.8E+03	1.2E+11
PA-231	4.0E+02	7.5E+10
NB-93M	3.9E+05	7.2E+10
TC--99	1.8E+05	5.9E+10
ZR--93	3.9E+05	4.6E+10
U--238	9.9E+02	4.2E+10
AC-227	4.0E+02	3.4E+10
U--235	4.0E+02	1.6E+10
TH-227	4.0E+02	1.0E+10
BI-210	3.7E+03	1.0E+10
TH-234	9.9E+02	5.4E+09
PD-107	4.2E+04	2.6E+09
SB-126	2.1E+02	9.5E+08
SN-126	2.1E+02	9.6E+08
NI--59	1.7E+02	1.2E+08
RB--87	5.7E+00	5.9E+06
SE--79	2.9E+00	5.9E+05
TOTAL		4.5E+14

Table I-7

Ore Required to Produce 1E+04 GWe-year

Assumptions:

1) 38 tonnes of 3.1% U-235 fuel are required per GWe-year.*

2) Uranium concentration in the ore body is 0.2%.

3) Enrichment tailings are 0.0025% U-235.

Then:

$$1E{+}04 \text{ GWe-year} \times \frac{38 \text{ tonne fuel}}{\text{Gwe-year}} \times \frac{0.031}{0.0071-0.0025} \times \frac{1}{0.002}$$

= 1.3E+09 tonnes 0.2% ore required

One tonne of 0.2% U ore contains

$$1000 \text{ kg} \times 0.0002 \times 1000 \ \frac{\text{grams}}{\text{kg}} = 2000 \text{ grams U-238}$$

$$\frac{2000 \text{ grams}}{238} \times 6.03E{+}23 \times \frac{0.693}{4.5E{+}09 \times 3.15E{+}07} \times \frac{1}{3.7E{+}10}$$

= 6.7E-04 curie/tonne

At secular equilibrium there are then

6.7E-04 curie U-238/tonne of ore and similar quantity of each daughter.

1.3E+09 tonnes ore @ 6.7E-04 curie/tonne

= 8.6E+05 curies U-238 and each daughter.

---

* This value, used throughout DOE/EIS-0046D (ref 7), is believed to be high but has been adopted here to be consistent with the values given in Table I-2, which also came from 0046D.

Table I-8

RQ Values for Natural Ore Body

Nuclide	Inventory (Ci)	Receptor Annual Dose of: 5 mrem
Ra-226	8.60E+05	4.6E+16
Subtotal		4.6E+16
Th-230	8.60E+05	7.3E+13
U-238	8.60E+05	3.6E+13
U-234	8.60E+05	3.5E+13
Th-234	8.60E+05	4.7E+12
TOTAL		4.6E+16

Table I-9

Ratio of RQ and RQ_s Values for Spent Fuel and HLW to the Equivalent Ore Body Values

RQ Type	Form of Waste	RQ Value Ratioed to that of 0.2% Ore Body at: Year 2050	500 years	1000 years	5000 years	10,000 years	50,000 years	100,000 years	500,000 years	1,000,000 years
Overall	Spent Fuel	1.4E+02	9.6E-01	5.2E-01	1.6E-01	1.5E-01	2.8E-01	4.1E-01	3.7E-01	2.1E-01
Overall	HLW	1.2E+02	3.3E-01	1.7E-01	3.0E-02	2.4E-02	1.6E-02	2.2E-02	1.9E-02	9.8E-03
System$(RQ/RQ_e)=RQ_s$	Spent Fuel	1.1E+02	8.8E-01	4.3E-01	7.8E-02	8.9E-02	2.7E-01	4.1E-01	3.7E-01	2.1E-01
System$(RQ/RQ_e)=RQ_s$	HLW	1.0E+02	3.6E-01	1.7E-01	2.7E-02	2.0E-02	1.5E-02	2.2E-02	1.8E-02	9.2E-03

TABLE I-10

RQ VALUES -- ENVIRONMENTAL DILUTION -- FOR SPENT FUEL, HLW, OR ORE
(RIVER WATER USER)

NUCLIDE	RQ	NUCLIDE	RQ
H----3	2.6E+09	RA-225	5.0E+08
SB-125	1.3E+09	PR-144	4.7E+08
U--236	1.3E+09	AM-241	4.6E+08
U--235	1.3E+09	AM242M	4.6E+08
U--234	1.3E+09	AM-243	4.6E+08
U--232	1.3E+09	CM-244	4.6E+08
U--233	1.3E+09	CM-243	4.6E+08
U--238	1.3E+09	CM-242	4.6E+08
PU-241	1.3E+09	CM-245	4.6E+08
PU-238	1.3E+09	EU-155	4.6E+08
PU-239	1.3E+09	PM-147	4.6E+08
PU-242	1.3E+09	EU-152	4.6E+08
PU-240	1.3E+09	EU-154	4.5E+08
SB-126	1.3E+09	Y---90	4.5E+08
ZR--93	1.0E+09	SM-151	4.5E+08
SR--90	9.8E+08	AC-225	3.7E+08
I--129	8.9E+08	AC-227	3.7E+08
NP-239	8.0E+08	AG110M	2.2E+08
NP-237	8.0E+08	RU-106	2.2E+08
BI-210	8.0E+08	CD113M	1.9E+08
CO--60	7.3E+08	FE--55	1.6E+08
TH-228	6.7E+08	TC--99	1.0E+08
TH-227	6.6E+08	TE125M	8.1E+07
TH-234	6.6E+08	CS-134	7.6E+07
TH-230	6.6E+08	CS-137	7.6E+07
TH-229	6.6E+08	CS-135	7.5E+07
PB-210	6.5E+08	SE--79	6.1E+07
PA-231	6.3E+08	RB--87	6.0E+07
PA-233	6.3E+08	SN-126	4.9E+07
NI--59	5.5E+08	PD-107	4.5E+07
NI--63	5.5E+08	PO-210	3.1E+07
CE-144	5.0E+08	C---14	1.5E+07
RA-223	5.0E+08	MN--54	7.8E+06
RA-224	5.0E+08	NB-93M	5.6E+06
RA-226	5.0E+08		

TABLE I-11 page 1 of 5

PQ_s VALUES FOR SPENT FUEL (DOSE TO RIVER WATER USER 5 MREM/YEAR)

YEAR 2050

NUCLIDE	PQ_s
SR--90	6.2E+09
CS-137	3.6E+09
AM-241	1.2E+08
Y---90	1.1E+08
SUB-TOTAL	1.0E+10
CS-134	1.5E+07
PU-238	1.3E+07
CM-244	7.2E+06
PU-240	4.2E+06
PU-241	3.8E+06
PU-239	2.7E+06
EU-154	2.7E+06
NI--63	1.2E+06
AM-243	5.3E+05
AM242M	3.5E+05
SM-151	1.8E+05
CO--60	1.7E+05
FE--55	1.1E+05
PM-147	7.9E+04
TE125M	5.9E+04
CM-243	5.0E+04
RU-106	4.4E+04
SB-125	2.9E+04
CM-242	2.8E+04
NB-93M	2.1E+04
TC--99	1.6E+04
SN-126	1.5E+04
PU-242	1.4E+04
U--234	1.3E+04
NP-237	9.9E+03
CM-245	7.4E+03
NP-239	7.3E+03
C---14	4.3E+03
CE-144	4.0E+03
U--238	3.8E+03
CD113M	3.6E+03
CS-135	3.4E+03
U--236	2.5E+03
EU-152	1.8E+03
NI--59	1.4E+03
I--129	6.4E+02
TOTAL	1.0E+10

YEAR 2070

NUCLIDE	PQ_s
SR--90	3.8E+09
CS-137	2.3E+09
AM-241	1.3E+08
Y---90	6.7E+07
PU-238	1.1E+07
SUB-TOTAL	6.3E+09
PU-240	4.2E+06
CM-244	3.4E+06
PU-239	2.7E+06
PU-241	1.5E+06
EU-154	1.1E+06
NI--63	1.1E+06
AM-243	5.2E+05
AM242M	3.2E+05
SM-151	1.5E+05
CM-243	3.3E+04
CM-242	2.6E+04
NB-93M	2.1E+04
CS-134	1.8E+04
TC--99	1.6E+04
SN-126	1.5E+04
U--234	1.4E+04
PU-242	1.4E+04
CO--60	1.2E+04
NP-237	1.0E+04
CM-245	7.4E+03
NP-239	7.2E+03
C---14	4.3E+03
U--238	3.8E+03
CS-135	3.4E+03
U--236	2.5E+03
NI--59	1.4E+03
CD113M	1.4E+03
PA-234	1.1E+03
TH-234	9.8E+02
SB-125	6.5E+02
I--129	6.4E+02
U--232	6.0E+02
EU-152	5.9E+02
FE--55	5.1E+02
SE--79	4.3E+02
TOTAL	6.3E+09

RQ_s VALUES FOR SPENT FUEL
(DOSE TO RIVER WATER USER 5 MREM/YEAR)

500 YEARS AFTER 1975

NUCLIDE	RQ_s
AM-241	7.3E+07
PU-240	4.0E+06
PU-239	2.7E+06
PU-238	5.0E+05
AM-243	5.0E+05
CS-137	2.0E+05
SR--90	1.8E+05
AM242M	5.1E+04
NI--63	5.1E+04
SUB-TOTAL	8.1E+07
NB-93M	2.1E+04
NP-237	2.0E+04
U--234	1.9E+04
RA-226	1.8E+04
TC--99	1.6E+04
SN-126	1.5E+04
PU-242	1.4E+04
CM-245	7.2E+03
NP-239	7.0E+03
SM-151	6.2E+03
C---14	4.2E+03
CM-242	4.1E+03
U--238	3.8E+03
CS-135	3.4E+03
Y---90	3.2E+03
U--236	2.5E+03
NI--59	1.4E+03
TH-234	9.8E+02
SB-126	6.4E+02
I--129	6.4E+02
SE--79	4.7E+02
PO-210	3.6E+02
PA-233	3.2E+02
TH-230	2.7E+02
U--235	2.0E+02
PB-210	1.9E+02
ZR--93	7.1E+01
PD-107	5.2E+01
RA-223	5.0E+01
PU-241	3.1E+01
RA-224	2.3E+01
TOTAL	8.1E+07

1000 YEARS AFTER 1975

NUCLIDE	RQ_s
AM-241	3.3E+07
PU-240	3.8E+06
PU-239	2.7E+06
AM-243	4.8E+05
RA-226	9.2E+04
SUB-TOTAL	4.0E+07
NP-237	2.7E+04
NB-93M	2.1E+04
U--234	2.0E+04
TC--99	1.6E+04
SN-126	1.5E+04
PU-242	1.4E+04
PU-238	1.2E+04
CM-245	6.9E+03
NP-239	6.7E+03
AM242M	5.2E+03
C---14	3.9E+03
U--238	3.8E+03
CS-135	3.4E+03
U--236	2.6E+03
PO-210	2.0E+03
NI--59	1.4E+03
NI--63	1.2E+03
PB-210	1.0E+03
TH-234	9.8E+02
TH-230	6.4E+02
SB-126	6.4E+02
I--129	6.4E+02
PA-233	4.3E+02
SE--79	4.3E+02
CM-242	4.2E+02
U--235	2.0E+02
SM-151	1.1E+02
RA-223	1.1E+02
CM-243	1.0E+02
ZR--93	7.1E+01
PD-107	5.2E+01
PA-231	4.3E+01
RA-225	3.9E+01
AC-227	3.4E+01
U--233	3.4E+01
TOTAL	4.0E+07

RQ_s VALUES FOR SPENT FUEL
(DOSE TO RIVER WATER USER 5 MREM/YEAR)

5000 YEARS AFTER 1975

NUCLIDE	RQ_s
PU-240	2.5E+06
PU-239	2.4E+06
RA-226	1.7E+06
AM-243	3.3E+05
AM-241	5.8E+04
PO-210	3.5E+04
NP-237	3.2E+04
NB-93M	2.1E+04
U--234	2.0E+04
PB-210	1.9E+04
TC--99	1.6E+04
SN-126	1.5E+04
PU-242	1.4E+04
SUB-TOTAL	7.2E+06
CM-245	4.9E+03
NP-239	4.7E+03
U--238	3.8E+03
TH-230	3.4E+03
CS-135	3.4E+03
U--236	3.0E+03
C---14	2.4E+03
RA-225	1.4E+03
NI--59	1.4E+03
TH-234	9.8E+02
TH-229	7.8E+02
I--129	6.4E+02
SB-126	6.2E+02
PA-233	5.1E+02
RA-223	5.1E+02
SE--79	4.1E+02
U--233	2.5E+02
U--235	2.1E+02
PA-231	2.1E+02
AC-227	1.6E+02
AC-225	7.9E+01
ZR--93	7.1E+01
BI-210	5.5E+01
PD-107	5.2E+01
TH-227	2.9E+01
PU-241	2.1E+01
PA-229	1.3E+00
TOTAL	7.2E+06

10,000 YEARS AFTER 1975

NUCLIDE	RQ_s
RA-226	4.4E+06
PU-239	2.1E+06
PU-240	1.5E+06
AM-243	2.1E+05
PO-210	9.3E+04
PB-210	4.9E+04
NP-237	3.2E+04
NB-93M	2.1E+04
U--234	2.0E+04
TC--99	1.5E+04
SN-126	1.4E+04
PU-242	1.4E+04
TH-230	6.9E+03
RA-225	5.4E+03
SUB-TOTAL	8.5E+06
U--238	3.8E+03
CS-135	3.4E+03
U--236	3.4E+03
CM-245	3.2E+03
TH-229	3.1E+03
NP-239	3.0E+03
AM-241	2.9E+03
NI--59	1.3E+03
C---14	1.3E+03
RA-223	1.0E+03
TH-234	9.8E+02
I--129	6.7E+02
SB-126	6.0E+02
U--233	5.2E+02
PA-233	5.1E+02
PA-231	4.2E+02
SE--79	3.9E+02
AC-227	3.7E+02
AC-225	3.1E+02
U--235	2.3E+02
BI-210	1.4E+02
ZR--93	7.1E+01
TH-227	5.5E+01
PD-107	5.2E+01
PU-241	1.4E+01
PA-229	2.8E+00
TOTAL	8.5E+06

RQ_s VALUES FOR SPENT FUEL
(DOSE TO RIVER WATER USER 5 MREM/YEAR)

50,000 YEARS AFTER 1975		100,000 YEARS AFTER 1975	
NUCLIDE	RQ_s	NUCLIDE	RQ_s
RA-226	2.3E+07	RA-226	3.6E+07
PU-239	6.7E+05	PO-210	7.6E+05
PO-210	4.9E+05	PB-210	4.0E+05
PB-210	2.5E+05	PU-239	1.6E+05
RA-225	6.0E+04	RA-225	1.2E+05
		TH-229	6.9E+04
SUB-TOTAL	2.4E+07	SUB-TOTAL	3.8E+07
TH-229	3.3E+04	TH-230	4.7E+04
NP-237	3.2E+04	NP-237	3.1E+04
TH-230	2.8E+04	NB-93M	2.0E+04
PU-240	2.5E+04	U--234	1.6E+04
NB-93M	2.0E+04	PU-242	1.2E+04
U--234	1.8E+04	TC--99	1.1E+04
TC--99	1.3E+04	SN-126	7.7E+03
PU-242	1.3E+04	AC-225	7.0E+03
SN-126	1.1E+04	RA-223	6.3E+03
AM-243	5.7E+03	U--233	4.5E+03
RA-223	4.2E+03	U--236	3.9E+03
U--236	3.9E+03	U--238	3.8E+03
U--238	3.8E+03	CS-135	3.3E+03
AC-225	3.4E+03	PA-231	2.6E+03
CS-135	3.3E+03	AC-227	2.0E+03
U--233	2.5E+03	BI-210	1.2E+03
PA-231	1.7E+03	TH-234	9.8E+02
AC-227	1.3E+03	I--129	6.3E+02
TH-234	9.8E+02	NI--59	6.1E+02
NI--59	9.4E+02	PA-233	5.0E+02
BI-210	7.5E+02	TH-227	3.4E+02
I--129	6.3E+02	SB-126	3.2E+02
PA-233	5.1E+02	U--235	3.1E+02
SB-126	4.6E+02	PU-240	1.3E+02
U--235	2.9E+02	SE--79	1.5E+02
SE--79	2.5E+02	ZR--93	6.9E+01
TH-227	2.3E+02	AM-243	6.1E+01
CM-245	1.1E+02	PD-107	5.1E+01
AM-241	1.0E+02	RA-228	3.5E+01
NP-239	7.9E+01	CM-245	1.7E+00
ZR--93	6.9E+01	AM-241	1.5E+00
PD-107	5.1E+01	NP-239	8.5E-01
RA-228	1.7E+01	RA-224	1.6E-01
C---14	1.0E+01	C---14	2.5E-02
PU-241	4.8E-01		
TOTAL	2.5E+07	TOTAL	3.8E+07

RQ_s VALUES FOR SPENT FUEL
(DOSE TO RIVER WATER USER 5 MREM/YEAR)

500,000 YEARS AFTER 1975		1,000,000 YEARS AFTER 1975	
NUCLIDE	RQ_s	NUCLIDE	RQ_s
RA-226	3.2E+07	RA-226	1.8E+07
PO-210	6.8E+05	PO-210	3.8E+05
PB-210	3.6E+05	RA-225	3.0E+05
RA-225	3.1E+05	PB-210	2.0E+05
TH-229	1.8E+05	TH-229	1.7E+05
SUB-TOTAL	3.4E+07	SUB-TOTAL	1.9E+07
TH-230	3.9E+04	NP-237	2.3E+04
NP-237	2.7E+04	TH-230	2.1E+04
AC-225	1.8E+04	AC-225	1.7E+04
NB-93M	1.6E+04	NB-93M	1.3E+04
U--233	1.0E+04	U--233	1.0E+04
RA-223	7.7E+03	RA-223	7.7E+03
U--234	7.7E+03	U--234	4.7E+03
PU-242	5.6E+03	U--238	3.8E+03
SN-126	4.8E+03	U--236	3.8E+03
U--236	3.8E+03	PA-231	3.1E+03
U--238	3.8E+03	CS-135	2.7E+03
PA-231	3.1E+03	AC-227	2.5E+03
TC--99	3.0E+03	PU-242	2.3E+03
CS-135	3.0E+03	TH-234	9.8E+02
AC-227	2.5E+03	I--129	6.1E+02
BI-210	1.1E+03	BI-210	5.9E+02
TH-234	9.8E+02	TC--99	5.9E+02
I--129	6.2E+02	TH-227	4.2E+02
PA-233	4.4E+02	PA-233	3.7E+02
TH-227	4.2E+02	RA-228	3.5E+02
U--235	3.1E+02	U--235	3.1E+02
SB-126	2.0E+02	PD-107	4.7E+01
RA-228	1.8E+02	ZR--93	4.5E+01
ZR--93	5.6E+01	SN-126	1.8E+01
PD-107	4.9E+01	RA-224	1.7E+00
NI--59	1.9E+01	TH-232	7.4E-01
PU-239	1.9E+01	SB-126	6.7E-01
SE--79	2.1E+00	TH-228	4.4E-01
RA-224	8.4E-01	NI--59	2.5E-01
TH-232	3.7E-01	RB--87	1.1E-01
TH-228	2.2E-01	SE--79	1.0E-02
RB--87	1.1E-01		
TOTAL	3.4E+07	TOTAL	1.9E+07

TABLE I-12

RQ_s VALUES FOR HLW
(DOSE TO RIVER WATER USER 5 MREM/YEAR)

YEAR 2050

NUCLIDE	RQ_s
SR--90	5.4E+09
CS-137	3.6E+09
Y---90	9.5E+07
CM-244	8.3E+07
AM-241	5.6E+07
CS-134	1.5E+07
SUB-TOTAL	9.2E+09
EU-154	3.1E+06
AM242M	2.9E+06
AM-243	2.8E+06
NI--63	1.1E+06
PU-238	6.8E+05
PU-240	2.7E+05
CM-242	2.3E+05
SM-151	1.9E+05
CO--60	1.5E+05
CM-245	1.4E+05
CM-243	1.3E+05
PU-241	1.0E+05
FE--55	8.5E+04
PM-147	7.7E+04
TE125M	7.1E+04
RU-106	4.9E+04
PU-239	4.1E+04
NP-239	3.9E+04
SB-125	3.5E+04
NB-93M	2.1E+04
SN-126	1.8E+04
NP-237	1.6E+04
TC--99	1.6E+04
RA-224	1.1E+04
CD113M	7.5E+03
U--232	5.2E+03
CS-135	4.3E+03
C---14	4.1E+03
CE-144	3.8E+03
EU-152	3.1E+03
TH-229	2.9E+03
NI--59	1.2E+03
EU-155	8.9E+02
I--129	6.8E+02
TOTAL	9.3E+09

YEAR 2070

NUCLIDE	RQ_s
SR--90	3.3E+09
CS-137	2.2E+09
Y---90	5.8E+07
AM-241	5.5E+07
CM-244	3.9E+07
SUB-TOTAL	5.7E+09
AM-243	2.7E+06
AM242M	2.6E+06
EU-154	1.3E+06
NI--63	9.1E+05
PU-238	6.5E+05
PU-240	3.1E+05
CM-242	2.1E+05
SM-151	1.6E+05
CM-245	1.4E+05
CM-243	9.7E+04
PU-239	4.1E+04
PU-241	4.0E+04
NP-239	3.8E+04
NB-93M	2.1E+04
SN-126	1.8E+04
CS-134	1.8E+04
NP-237	1.7E+04
TC--99	1.6E+04
CO--60	1.1E+04
RA-224	8.1E+03
CS-135	4.3E+03
U--232	4.3E+03
C---14	4.1E+03
CD113M	2.9E+03
TH-229	2.1E+03
NI--59	1.2E+03
EU-152	9.9E+02
SB-126	7.5E+02
I--129	6.8E+02
PU-242	5.7E+02
TE125M	4.2E+02
FE--55	4.1E+02
SE--79	4.1E+02
PM-147	3.9E+02
PA-233	2.6E+02
TOTAL	5.7E+09

RQ$_s$ VALUES FOR HLW
(DOSE TO RIVER WATER USER 5 MREM/YEAR)

500 YEARS AFTER 1975

NUCLIDE	RQ$_s$
AM-241	2.9E+07
AM-243	2.7E+06
AM242M	4.2E+05
PU-240	3.4E+05
CS-137	2.0E+05
SR--90	1.6E+05
PU-238	1.5E+05
CM-245	1.4E+05
SUB-TOTAL	3.3E+07
PU-239	4.7E+04
NI--63	4.4E+04
NP-239	3.7E+04
CM-242	3.3E+04
NB-93M	2.1E+04
NP-237	2.0E+04
SN-126	1.8E+04
TC--99	1.6E+04
SM-151	6.6E+03
CS-135	4.3E+03
C---14	3.9E+03
Y---90	2.8E+03
NI--59	1.2E+03
RA-226	8.4E+02
SB-126	7.5E+02
U--234	7.5E+02
I--129	6.8E+02
PU-241	5.9E+02
PU-242	5.7E+02
SE--79	4.1E+02
PA-233	3.2E+02
RA-224	1.6E+02
U--232	8.8E+01
ZR--93	7.0E+01
PD-107	6.5E+01
TH-228	4.3E+01
U--236	3.3E+01
U--238	3.1E+01
PO-210	1.7E+01
CM-243	1.4E+01
U--233	1.3E+01
TH-230	1.1E+01
TOTAL	3.3E+07

1000 YEARS AFTER 1975

NUCLIDE	RQ$_s$
AM-241	1.3E+07
AM-243	2.5E+06
PU-240	3.3E+05
CM-245	1.3E+05
PU-239	5.6E+04
SUB-TOTAL	1.6E+07
AM242M	4.3E+04
NP-239	3.5E+04
NP-237	2.3E+04
NB-93M	2.1E+04
SN-126	1.8E+04
PU-238	1.8E+04
TC--99	1.6E+04
CS-135	4.3E+03
RA-226	3.9E+03
C---14	3.7E+03
CM-242	3.4E+03
NI--59	1.2E+03
NI--63	1.0E+03
U--234	9.4E+02
SB-126	7.5E+02
I--129	6.8E+02
PU-242	5.8E+02
PU-241	5.7E+02
SE--79	4.0E+02
PA-233	3.7E+02
SM-151	1.2E+02
PO-210	8.7E+01
ZR--93	7.0E+01
PD-107	6.5E+01
PB-210	4.3E+01
RA-225	4.1E+01
U--236	3.9E+01
U--233	3.3E+01
U--238	3.1E+01
TH-230	2.8E+01
TH-229	2.3E+01
RA-223	9.4E+00
TH-234	8.1E+00
PA-231	3.8E+00
AC-227	3.0E+00
TOTAL	1.6E+07

RQ_s VALUES FOR HLW
(DOSE TO RIVER WATER USER 5 MREM/YEAR)

5000 YEARS AFTER 1975		10,000 YEARS AFTER 1975	
NUCLIDE	RQ_s	NUCLIDE	RQ_s
AM-243	1.8E+06	AM-243	1.1E+06
PU-240	2.2E+05	RA-226	2.1E+05
AM-241	1.1E+05	PU-239	1.4E+05
PU-239	1.0E+05	PU-240	1.3E+05
CM-245	9.4E+04	CM-245	6.2E+04
RA-226	7.8E+04	AM-241	5.5E+04
NP-237	2.5E+04	NP-237	2.5E+04
NP-239	2.5E+04	NB-93M	2.0E+04
NB-93M	2.1E+04	SN-126	1.7E+04
SN-126	1.7E+04	NP-239	1.6E+04
TC--99	1.5E+04	TC--99	1.5E+04
SUB-TOTAL	2.5E+06	SUB-TOTAL	1.8E+06
CS-135	4.3E+03	PO-210	4.4E+03
C---14	2.3E+03	RA-225	4.4E+03
PO-210	1.6E+03	CS-135	4.3E+03
NI--59	1.2E+03	TH-229	2.4E+03
RA-225	1.2E+03	PB-210	2.3E+03
U--234	9.5E+02	C---14	1.2E+03
PB-210	8.6E+02	NI--59	1.1E+03
SB-126	7.3E+02	U--234	9.4E+02
I--129	6.8E+02	SB-126	7.1E+02
TH-229	6.6E+02	I--129	6.8E+02
PU-242	5.7E+02	PU-242	5.7E+02
PU-241	4.1E+02	U--233	4.1E+02
PA-233	4.0E+02	PA-233	4.0E+02
SE--79	3.9E+02	SE--79	3.7E+02
U--233	2.0E+02	TH-230	3.2E+02
TH-230	1.6E+02	PU-241	2.7E+02
U--236	7.6E+01	AC-225	2.5E+02
ZR--93	7.0E+01	U--236	1.0E+02
AC-225	6.7E+01	ZR--93	6.9E+01
PD-107	6.5E+01	PD-107	6.5E+01
U--238	3.1E+01	U--238	3.1E+01
RA-223	1.2E+01	RA-223	1.7E+01
TH-234	8.1E+00	TH-234	8.1E+00
PA-231	5.0E+00	PA-231	6.8E+00
AC-227	3.9E+00	BI-210	6.8E+00
BI-210	2.5E+00	AC-227	5.4E+00
U--235	2.0E+00	U--235	2.7E+00
TH-227	5.6E-01	TH-227	9.1E-01
RB--87	9.8E-02	RB--87	9.8E-02
TOTAL	2.5E+06	TOTAL	1.8E+06

RQ_s VALUES FOR HLW
(DOSE TO RIVER WATER USER = MREM/YEAR)

50,000 YEARS AFTER 1975		100,000 YEARS AFTER 1975	
NUCLIDE	RQ_s	NUCLIDE	RQ_s
RA-226	1.1E+06	RA-226	1.7E+06
PU-239	8.0E+04	RA-225	9.6E+04
RA-225	4.7E+04	TH-229	5.4E+04
AM-243	3.0E+04	PO-210	3.5E+04
TH-229	2.6E+04	NP-237	2.5E+04
NP-237	2.5E+04	PU-239	2.0E+04
PO-210	2.3E+04	NB-93M	2.0E+04
NB-93M	2.0E+04	PB-210	1.9E+04
TC--99	1.3E+04	TC--99	1.1E+04
SN-126	1.3E+04	SN-126	9.1E+03
PB-210	1.2E+04	AC-225	5.5E+03
SUB-TOTAL	1.4E+06	SUB-TOTAL	2.0E+06
CS-135	4.2E+03	CS-135	4.2E+03
AC-225	2.7E+03	U--233	3.5E+03
CM-245	2.2E+03	TH-230	2.[illegible]E+03
PU-240	2.1E+03	U--234	7.3E+02
U--233	2.0E+03	I--129	6.4E+02
AM-241	1.9E+03	NI--59	5.3E+02
TH-230	1.3E+03	PU-242	4.8E+02
U--234	8.4E+02	PA-233	4.0E+02
NI--59	8.1E+02	SB-126	3.8E+02
I--129	6.8E+02	AM-243	3.2E+02
SB-126	5.4E+02	RA-223	2.0E+02
PU-242	5.3E+02	U--236	1.5E+02
NP-239	4.2E+02	SE--79	1.4E+02
PA-233	4.0E+02	PA-231	8.1E+01
SE--79	2.4E+02	ZR--93	6.7E+01
U--236	1.5E+02	PD-107	6.4E+01
RA-223	9.5E+01	AC-227	6.4E+01
ZR--93	6.8E+01	BI-210	5.5E+01
PD-107	6.5E+01	CM-245	3.3E+01
PA-231	3.9E+01	U--238	3.1E+01
BI-210	3.6E+01	AM-241	2.9E+01
U--238	3.1E+01	PU-240	1.3E+01
AC-227	3.1E+01	U--235	1.1E+01
C---14	9.8E+00	TH-227	1.1E+01
PU-241	9.3E+00	TH-234	9.1E+00
U--235	8.4E+00	NP-239	4.5E+00
TH-234	8.1E+00	PU-241	1.4E-01
TH-227	5.2E+00	RB--87	9.8E-02
RB--87	9.8E-02	C---14	2.3E-02
TOTAL	1.4E+06	TOTAL	2.0E+06

RQ_s VALUES FOR HLW
(DOSE TO RIVER WATER USER 5 MREM/YEAR)

500,000 YEARS AFTER 1975

NUCLIDE	RQ_s
RA-226	1.2E+06
RA-225	2.5E+05
TH-229	1.4E+05
PO-210	2.6E+04
NP-237	2.2E+04
NB-93M	1.6E+04
AC-225	1.4E+04
PB-210	1.3E+04
U--233	8.1E+03
SUB-TOTAL	1.7E+06
CS-135	3.9E+03
TC--99	3.0E+03
TH-230	1.5E+03
I--129	6.6E+02
SN-126	5.7E+02
PA-233	3.5E+02
RA-223	2.9E+02
U--234	2.6E+02
PU-242	2.3E+02
U--236	1.5E+02
PA-231	1.2E+02
AC-227	9.4E+01
PD-107	6.2E+01
ZR--93	5.6E+01
BI-210	4.0E+01
U--238	3.1E+01
SB-126	2.4E+01
NI--59	1.6E+01
TH-227	1.6E+01
U--235	1.2E+01
TH-234	8.1E+00
SE--79	2.0E+00
PU-239	2.3E-01
RB--87	9.8E-02
TH-228	4.3E-12
PA-224	1.0E-12
TOTAL	1.7E+06

1,000,000 YEARS AFTER 1975

NUCLIDE	RQ_s
RA-226	4.0E+05
RA-225	2.4E+05
TH-229	1.3E+05
NP-237	1.9E+04
AC-225	1.4E+04
NB-93M	1.3E+04
PO-210	9.3E+03
U--233	7.9E+03
PB-210	4.4E+03
CS-135	3.4E+03
I--129	6.5E+02
TC--99	5.9E+02
SUB-TOTAL	8.4E+05
TH-230	4.7E+02
PA-233	3.0E+02
RA-223	2.9E+02
U--236	1.4E+02
PA-231	1.2E+02
AC-227	9.3E+01
PU-242	9.3E+01
U--234	8.7E+01
PD-107	5.9E+01
ZR--93	4.4E+01
U--238	3.1E+01
SN-126	1.8E+01
TH-227	1.6E+01
BI-210	1.7E+01
U--235	1.2E+01
TH-234	8.1E+00
SB-126	7.4E-01
NI--59	2.2E-01
RB--87	9.8E-02
SE--79	9.6E-03
TH-228	4.2E-12
PA-224	9.8E-13
TOTAL	8.4E+05

REFERENCES FOR SECTION I

1. U.S. Nuclear Regulatory Commission, In the Matter of Proposed Rulemaking on the Storage and Dispoal of Nuclear Waste, Statement of Position of the United States Department of Energy (DOE/NE - 0007 April 15, 1980).

2. U.S. Nuclear Regulatory Commission, In the Matter of Proposed Rulemaking on the Storage and Dispoal of Nuclear Waste, First Prehearing Conference Order, February 1, 1980.

3. U.S. Nuclear Regulatory Commission, In the Matter of Proposed Rulemaking on the Storage and Dispoal of Nuclear Waste, Summary of UWMG-EEI Positions Re Subjects Identified in Notice of Prehearing Conference (January 28, 1980).

4. Op cit, DOE (Ref. 1), p. I-2.

5. D. Godard, Report on Disposal of Spent Fuel, Oregon Department of Energy (November 1979).

6. W.A. Rodger, Critical Evaluation of the Limit of Transuranic Contamination of Low Level Waste, American Institute of Chemical Engineers Symposium Series (Vol. 72, No. 154).

7. Op cit, Godard (Ref. 5).

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

9. J.J. Cohen, et al, Determination of Performance Criteria for High-Level Solidified Nuclear Waste, Lawrence Livermore Laboratory for the U.S. Nuclear Regulatory Commission (NUREG-0279, July 1977), pp. 117-136.

10. B.L. Cohen, The Disposal of Radioactive Wastes from Fission Reactors, Scientific American (Vol. 236, No. 6, June 1977), p. 28.

11. J.B. Martin, Regulation of Geological Repositories, Presented at the Keystone Center for Continuing Education Conference in Keystone, Colorado (October 23, 1979), p. 6.

12. Op cit, DOE (Ref. 8).

13. Hearing on Nuclear Waste Disposal Before the Subcomm. on Energy, Nuclear Proliferation and Federal Services of the Senate Comm. on Governmental Affairs, 96th Cong., 1st Sess. (March 13, 1979) (statement of J.E. Mendel), p. 8.

REFERENCES FOR SECTION I (CONT)

14. M.D. Hill, The Hazards Associated with Vitrified High-Level Radioactive Waste (National Radiological Protection Board, Bull. No. 212, Oct. 1977).

15. H.C. Burkholder, et al, Incentives for Partitioning High Level Waste, Nuclear Technology (November 1976).

16. National Council on Radiation Protection and Measurements, Rpt. No. 45, Natural Background Radiation in the United States (November 1975).

17. Id.

18. D.T. Oakley, Natural Radiation Exposure in the United States (Ph.D. Dissertation, Harvard University School of Public Health; also printed by USEPA as ORP/SID 72-1, June 1972).

19. Op cit, NCRP (Ref. 16).

20. United Nations Scientific Committee on the Effects of Atomic Radiation, Ionizing Radiation: Levels and Effects 1 (1972).

21. EPA Interim Primary Drinking Water Regulations, 40 CFR Part 141 (1976).

22. Op cit, NCRP (Ref. 16).

23. Op cit, UN (Ref. 19).

24. Op cit, NCRP (Ref. 16).

25. Department of Health and Rehabilitation Services, Radiation Health Service, State of Florida, Draft Study of Radon Daughter Concentrations in Polk and Hillsborough Counties (November 1977).

26. Op cit, NCRP (Ref. 16).

27. D.B. Yeates, et al, Natural Radiation in the Urban Environment, Nuclear Safety (Vol. 13, No. 4, Jul.-Aug. 1972), pp. 275-286.

28. E.P. Radford, Jr. and V.R. Junt, Polonium-210: A Volatile Radio-Element in Cigarettes, Science (1964), pp. 143, 247.

29. Op cit, UN (Ref. 20).

30. C.J. Barton, et al, Contribution of Radon in Natural Gas to the Natural Radioactivity Dose in Homes, Oak Ridge National Laboratory (ORNL-TM-4154, 1973).

31. S.T. Windham, et al, The Effects of Home Ventiliation Systems on Indoor Radon-Radon Daughter Levels, Office of Radiation Programs, U.S. Environmental Protection Agency (1977).

REFERENCES FOR SECTION I (CONT)

32. C.C. Travis, et al, Natural and Technologically Enhanced Sources of Radon-222, Nuclear Safet--, (vol. 20-26, Nov.-Dec. 1979).

33. B. Shleien, et al, The Mean Active Bone Marrow Dose from Diagnostic Radiology, Bureau of Radiological Health, U.S. Food and Drug Administration (FDA-77-8013, April 1976).

34. J.N. Gitlin, Preliminary Dose Estimates from the U.S. Public Health Service 1970 X-Ray Exposure Study, Presented at 49th Annual Meeting of the American College of Radiology (April 1972).

35. Op cit, DOE (Ref. 1), p. II-6, II-9 through II-15.

36. NRC Standards for Protection Against Radiation, 10 CFR Part 20, Appendix B, "Concentration in Air and Water Above Natural Background" (1976).

37. R.J. Cholister, et al, Nuclear Fuel Cycle Closure Alternatives, Allied-General Nuclear Services (September 1975).

38. Regulatory Guide 1.109, Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I, U.S. Nuclear Regulatory Commission (March 1976).

39. Recommendations of the International Commission on Radiological Protection, Annuals of the ICRP (Vol. 1, No. 3), ICRP Publication 26 (1977).

40. Op cit, DOE (Ref. 8), Vol. 2 Tables A-14, A-15, A-38, A-39.

41. R.L. Frendberg, Computer Code Reference RQ GEN (January 1980).

42. Op cit, Cohen (Ref. 9).

43. Hearing on Impact Assessment of High Level Wastes Before the House Co[illegible] on Interior and Insular Affairs, 96th Cong., 1st Sess. (Statement of J. Martin, et al), p. 10.

44. Op cit, DOE (Ref. 8), pp. 3.1.9, 3.1.39.

45. Op cit, Mendel (Ref. 13), p. 12.

46. Id, (Ref. 13), pp. 6 and 13.

47. Op cit, DOE (Ref. 8), pp. 1.14, 3.1.32, 3.1.37.

48. Op cit, Mendel (Ref. 12), pp. 7, 8, and 12-13.

REFERENCES FOR SECTION I (CONT)

49. Hearing Before the Subcomm. on Energy Research and Production of the House Comm. on Science and Techr ogy, 96th Cong., 1st Sess. (May 16, 1979) (Statement of F.L. Parker), p. 28.

50. Op cit, DOE (Ref. 8), p. 3.1.23.

51. F. Gera, et al, Considerations in the Long-Term Management of High Level Radioactive Wastes, Oak Ridge National Laboratory (ORNL-4762, February 1972) p. 58.

52. Op cit, Mendel (Ref. 13), p. 14.

II DISPOSAL SYSTEM TECHNOLOGY ALTERNATIVES

In Section I deep geologic disposal was discussed as a waste disposal system in order to explain how the goal of protecting the public health, safety, and environment might be attained. In this section, we examine the history of conventional mined geologic disposal and explain the rationale of our choice of this technology as the only appropriate alternative to be assessed for the purpose of a finding of confidence.

A variety of system concepts for the disposal of high-level radioactive wastes have been identified.[1] They essentially all involve two major features--waste form (and package) and specific ultimate disposal environment.

It is recognized that because of current Administration policies concerning the reprocessing of spent fuel, this proceeding is focusing on spent fuel as a waste form for disposal. Nevertheless, it is important to note the existence of other potential waste forms, particularly solidified high-level wastes (HLW) prepared from high-activity liquid wastes evolved from reprocessing. As we discuss in Section III-B, if deep geologic disposal is acceptable for spent fuel, it obviously is also acceptable for HLW.

A DEEP GEOLOGIC REPOSITORIES

1) Historical Background

Recognition of the need for developing acceptable systems for disposal of high-level radioactive wastes from the nuclear power industry goes back more than 25 years. The first comprehensive scientific and technical discussion of the approaches to disposal of such wastes in geologic formations was at a conference requested by the Division of Reactor Development of the Atomic Energy Commission (AEC) held at Princeton University in September 1955 under the auspices of the Committee on Waste Disposal of the Division of Earth Sciences in the National Academy of Sciences/National Research Council (NAS/NRC).[2] A number of different approaches to disposal of high-level wastes were considered by the 65 experts in the earth sciences, chemistry, physics, engineering, biology, and medicine; included were disposal in deep, closed hydrologic basins, disposal in the oceans, and other variations. At that time the closure of the fuel cycle was to be accomplished by reprocessing of the spent fuel and disposal of the resulting HLW. The consensus was that disposal of solidified waste forms into specially selected deep salt formations provided the best possibility for technical feasibility, safety, and environmental protection. In the referenced report of the Committee a discussion entitled "Disposal of Radioactive Waste in Salt Cavities" was included as an appendix.

The advantageous properties of salt formations as a medium for waste disposal were pointed out by the Committee. They included:

a) Most importantly, salt formations, whose geologic age is about 250 million years, are isolated from and essentially impermeable to circulating groundwater.

b) The ability of salt to flow plastically under very high pressure is one of the ways in which this impermeability is maintained. Any fractures which may develop will close up or heal because of the ability of the salt to deform plastically.

c) Salt formations are geographically widespread. They underlie portions of 23 states and are not considered to be a scarce or highly valuable natural resource.

d) Salt, compared to other rock materials, has a relatively high thermal conductivity which enhances heat transfer and helps to limit the maximum temperatures resulting from the decay heat generated by the wastes.

e) Salt is readily mined, and salt mining is one of the most efficient underground extractive industries and is supported by well developed and widely used technology.

Following the Committee's recommendation that disposal in selected salt formations appeared to be the most promising approach, a detailed laboratory and field investigative program was initiated. This program culminated in the successful conclusion of Project Salt Vault, a test and demonstration program carried out in the Carey Salt Co mine at Lyons, Kansas in 1968. The results of this program have been extensively reported[3] and described.[4] As stated in ORNL-4555 "With the completion of this experiment, it can be concluded that most of the major technical problems pertinent to the disposal of highly radioactive waste in salt have been resolved. Project Salt Vault successfully demonstrated the feasibility and safety of handling highly radioactive materials in an underground environment. The stability of the salt under the effects of heat and radiation has been shown, as well as the capability of solving minor structural problems by standard mining techniques. The data obtained on the deformational characteristics of salt have made it possible to arrive at a suitable design for a mine disposal facility."

The successful completion of Project Salt Vault led to further consideration of the feasibility of establishing an operating repository at Lyons to meet the requirements for commercial waste managment. A conceptual design was developed by ORNL in 1969, and in 1970 the scope of the design was expanded to include a capability for disposal of transuranic (TRU) wastes.[5] The NAS/NRC Committee continued its oversight and endorsement of the program.[6]

Subsequently, more detailed investigation of the proposed repository location environs identified a number of exploratory oil and gas wells and a solution mining operation in the vicinity. This identification coupled with politico-institutional issues led to the conclusion that the Lyons site was not suitable for a Federal waste repository. It is emphasized that the circumstances which led to this conclusion were entirely related to man-made factors. The negative conclusion regarding the proposed site was in no way associated with any lack of viability of the deep geologic repository concept, nor to any basic questions regarding the feasibility of salt formations as a host formation for HLW disposal.[7]

Accordingly, following withdrawal of the proposal for a repository at the Lyons site a study was conducted of alternative locations for repository development. It was concluded from this study[8] that salt formations in the Delaware sub-basin of the Permian Basin near Carlsbad, New Mexico offered potential for a suitable repository site. Exploratory investigation and drilling in the Delaware sub-basin study area were carried out until mid-1974. At that time, however, the Federal program for a brief period was focused on the development of a Retrievable Surface Storage Facility (RSSF) rather than an ultimate disposal geologic repository. In April 1975 because of criticisms from a number of sources regarding the suitability of proceeding with an interim--rather than the ultimate--solution the newly established Energy Research and Development Administration (ERDA), which had assumed the AEC's waste management responsibilities, withdrew the RSSF program.[9]

Following this withdrawal, the Waste Isolation Pilot Plant (WIPP) project was, in effect, re-initiated in the Delaware sub-basin and aimed at the ultimate construction of a pilot transuranic waste disposal facility. It was stated by ERDA officials at that time (1975) that the objective of providing facilities to permanently dispose of commercial and ERDA TRU waste was achievable with proven, existing analytical capabilities and technology.[10] Since that time work has progressed on the WIPP project through the issuance of a Geological Characterization Report[11] and a Draft Environmental Impact Statement[12] and the initiation of repository design.

On February 12, 1980 the President in his policy statement on nuclear waste management[13] called for cancellation of the WIPP project (authorized for the disposal of defense TRU wastes) for institutional reasons, but stated that the site will continue to be evaluated along with other sites and, if qualified, will be reserved as one of several candidate sites for possible use as a repository for defense and commercial high-level wastes. Congress has not yet acted with respect to such cancellation.

With respect to the management of commercial reactor HLW, in the latter part of 1976 ERDA launched what was characterized as a major, expanded National Waste Terminal Storage (NWTS) program. The objectives of this program were stated to be:[14]

1) to provide terminal storage facilities for commercial radioactive waste at multiple geographic locations throughout the United States;
2) to establish such facilities in a time frame that assures nuclear power as a viable energy option; and
3) to assure that these facilities provide for the safe disposition of solidified commercial radioactive waste which must be delivered to the Federal Government for terminal storage.

A more detailed press release on the NWTS was issued on December 2, 1976.[15] This release stressed President Ford's emphasis of October 28, 1976[16] "to speed up the program to demonstrate all components of waste management technology by 1978 and to demonstrate a complete repository for commercial high-level nuclear wastes by 1985." A simultaneously released Fact Sheet[17] provided additional details on the program implementation plans, including a phased approach to investigate the geology in 36 states. These states were divided into three groups, based on available geologic information, and the potential in each state for locating suitable geologic formations for a nuclear waste repository were indicated. This expanded program continued to be based on the viability of the deep geologic repository and recognized the potential suitability of geologic formations in addition to rock salt, ie, crystalline rocks, including basalt and granite, and argillaceous formations. It also recognized the possibility that the waste form might be spent fuel and that the waste management system should accomodate solidified HLW or spent fuel.

With the advent of a new Administration in January 1977 ERDA was replaced by the DOE. During 1977, as part of the National Energy Plan a review of the US nuclear waste managment plan was directed. As a result, a task force internal to DOE was formed to carry out the review and published a draft report in February 1978.[18] The report concluded that "There appears to be a substantive consensus and valid technical basis for the view that present plans and actions should rely on geologic containment of waste <u>which can be achieved in a safe and environmentally acceptable manner</u>" (emphasis added). The report also noted that this view has been promulgated over a period of time by several independent assessments ranging from that of the National Academy of Sciences in 1957 (and subsequent reaffirmations by that body) to that of the American Physical Society[19] in 1977 and further pointed out that similar findings have been expressed in government-supported reviews in other countries (eg, West Germany, Sweden and Canada).

Following the issuance of the Task Force report, a broadly based governmental review of overall nuclear waste management, including high-level wastes, was directed by the President on March 13, 1978 to be carried out by an Interagency Nuclear Waste Management Task Force (subsequently identified as the Interagency Review Group on Nuclear Waste Management). This group comprised representatives of 14 Federal agencies. After receiving public comments on a draft report, the group issued its final report in March 1979.[20] Its principal conclusion was that "Present scientific and technological knowledge is adequate to identify potential repository sites for further investigation. No scientific or technical reason is known that would prevent identifying a site that is suitable for a repository provided that the systems view is utilized rigorously to evaluate the suitability of sites and designs, and in minimizing the influence of future human activities. A suitable site is one at which a repository would meet predetermined criteria and which would provide a high degree of assurance that radioactive waste can be successfully isolated from the biosphere for periods of thousands of years" (emphasis added).[21] Later on the report states, "Disposing of nuclear wastes in mined repositories is a highly promising approach to long-term isolation. While there is a possibility that such a technique could not be successfully employed, there is a high degree of confidence that a repository can be sited, designed, and operated so as to provide reasonable assurance of long-term isolation of radionuclides" (emphasis added).[22]

Subsequently, in the DEIS issued in April 1979, DOE once again arrived at positive conclusions regarding the viability and validity of the deep geologic repository concept. The DEIS in its summary states that "Based on the analysis presented here, and in the light of the greater depth of knowledge on geologic disposal, DOE proposes that (1) the disposal of radioactive wastes in geologic formations can likely be developed and applied with minimal environmental consequences, and (2) therefore the program emphasis should be on the establishment of mined repositories as the operative disposal technology."[23]

More recently still another independent, authoritative group, under the aegis of the National Academy of Sciences, addressed the subject of nuclear wastes in connection with its comprehensive, 4-year energy study. It concluded that "No insurmountable technical obstacles are foreseen to preclude safe disposal of nuclear wastes in geologic formations. All necessary process steps for immobilizing high- and low-level wastes have been developed, and there are no technical barriers to their implementation."[24]

Historically, it is clear that the basic, fundamental features of a viable waste management system have been identified as an appropriately selected deep geologic repository and a suitable waste form or package. From the initial consideration of the deep geologic repository concept, while it was recognized that a solid waste form would be required and that the specific characteristics of the waste form--or package--would have to be compatible with the characteristics of the host formation, the viability of disposal in deep geologic formations was essentially independent of the specific nature of the solid waste form. The waste form, in essence, was considered as a part of the overall system that represented another barrier to migration of nuclides to the biosphere. Accordingly, the consideration of disposal of spent fuel is quite consistent with the conclusions reached in all the evaluations relative to the repository concept.

Thus, successive Federal agencies, groups operating within the most prestigious scientific organizations in the land, independent scientific and technical groups, foreign governments, and others, using repeated analyses and evaluations and results of research, development, and field tests, have all, over the past generation, generally concluded that mined repositories in deep geologic formations are capable of containing and isolating high-level nuclear wastes, including spent fuel, in a safe and environmentally acceptable manner.

2) Geologic Formation Alternatives

Although currently the knowledge base is greatest for salt formations as the host rock for a repository, as indicated previously, other formations are also likely to be suitable. Factors which determine the suitability of a host rock include tectonic and mechanical stability, related hydrology and geochemistry, physical extent of host formation, and homogeneity and characteristics of surrounding geologic, hydrologic, and geochemical conditions. On an overall systems basis, host geologic formation suitability is also related to the waste form and package, ie, the waste form and package can be tailored to the characteristics of the repository formation as required to achieve overall system performance.

The potential suitability of several rock types as repository formations has been recognized in both the US NWTS program and foreign programs as well. In the US this recognition culminated recently in the President's message to

Congress[25] on the radioactive waste management program. In this message the President calls for locating and characterizing several suitable sites in a variety of geologic formations before selecting one or more sites for further investigation and development as a licensed repository.

Currently the geologic formation of rock type, besides salt, that is farthest along in its investigation and evaluation is granite, as a result of work in Sweden. This effort is described in more detail elsewhere in this report, but, in essence, it has been concluded[26,27] by the Swedish Government that granite is a suitable repository medium for the waste packages, including both spent fuel and vitrified HLW as waste forms, designed for disposal and even that such disposal is "absolutely safe." The major features of the proposed waste packages include 40-year cooling to reduce thermal loading, lead-filled copper canisters for spent fuel, lead and titanium canisters for vitrified HLW, and quartz-bentonite backfill. From a scientific standpoint probably the most significant issue concerning granite is the small flow of groundwater through fractures and, related to this point, the expected eventual saturation of the repository. In the Swedish approach this is compensated for by the design of the waste package.

B Alternative Disposal Systems

A number of disposal systems alternative to the mined deep geologic repository have been considered in the DEIS.[28] Included were:

- Subseabed disposal (see C infra)
- Very deep hole
- Rock melting
- Reverse well
- Space
- Ice sheet
- Island

A few brief comments follow about each of these alternatives, except subseabed disposal which is discussed in Section II-C, infra.

Very deep hole A potential alternative for nuclear waste management is to drill or sink a shaft to isolate high-level wastes in a very deep hole. This concept relies on using surrounding rock to contain the wastes and on the great depths to delay the release and reentry of nuclear material to the biosphere. The utility of the deep hole concept is affected by three principal factors, which depend upon the specific characteristics of the site and the size of the hole.

First, the geologic characteristics of the site, including hydrologic conditions, rock strength, and rock/waste interactions at great depths, are not well known. Hence, "how deep is deep enough" is not well defined. Very deep holes located in strong, unfractured rock, such as crystalline rock (which typically has low water content) or some deep sedimentary basins, would be a good selection for a deep hole site.

Secondly, the current capability to excavate a very deep hole has been established. Presently it is possible to drill a narrow deep hole to 10 km (35,000 ft) or to sink a wide shaft to about 4 km (15,000 ft). Whether or not the hole would have to be cased depends on the rock strength and confining pressures.

Third, the safe emplacement of wastes in this concept may present severe engineering problems. Lowering waste canisters 10 to 12 km on a wire through high-density muds could significantly increase short-term risks. Also, the number of holes (800-1300) required may be prohibitive.

The deep hole concept cannot be evaluated as a nuclear waste disposal alternative without more information on the deep groundwater system, rock strength under increased temperatures and pressure due to decay of wastes, and the sealing of the holes over long periods of time.

Rock melting The rock melting concept for geologic disposal of nuclear wastes calls for the direct emplacement of liquid waste in a deep undergound hole or cavity. Radioactive decay heat causes melting of the surrounding rock, which in turn dissolves the waste. In time the waste-rock solution refreezes, trapping the radioactive material in a relatively insoluble matrix deep underground. Presumed advantages of the method include ability to directly emplace high-level liquid waste without need for solidification or transportation, although presolidified waste or spent fuel can also be directly emplaced. Low- and intermediate-level TRU wastes would still probably have to be disposed of in a conventional geologic repository.

Extensive research to develop the concept includes gaining a thorough understanding of heat transfer and phase change phenomenology in rock. Also development of engineering methods for waste emplacement would be required.

Reverse well Reverse-well, or deep-well, disposal of radioactive wastes encompasses two distinctly different techniques:

1) injection of the waste in an acidic liquid form, essentially as received from the Purex process (referred to hereafter as "deep-well liquid injection"); or
2) injection of the waste in a slurry composed of neutralized liquid waste mixed with cement, clay, and other additives designed to ensure ultimate solidification after injection (referred to hereafter as "shale-grout injection")

The deep-well liquid injection concept draws largely upon existing disposal practices in the oil and gas industry. Shale-grout injections of intermediate-level wastes have been undertaken by Oak Ridge National Laboratory at Oak Ridge, Tennessee.

This concept is not suitable for the once-through cycle unless the practice of reprocessing only for disposal is utilized. Extensive technology development would be required in order to determine the viability of this alternative.

Space The dominant attraction of disposal of nuclear waste in space is the promise of permanent separation of selected wastes from the human environment. Some of the space-unique systems are in development (Space Shuttle and facilities) or are being planned for other requirements (orbital transfer vehicle). The major area of development is associated with the safety/environmental concerns and requires extensi additional study before a decision can be made on whether or not to proceed with the space disposal option. Space disposal of spent fuel is in any case not practical because of the large number of launches that would be required.

Ice sheet Disposal of nuclear waste in the Antarctic or Greenland ice sheets, if and when developed, could offer the advantages of remoteness from human activities and potential isolation from the biosphere. If conditions prevalent in these areas for the past 2 or 3 million years persist, the ice sheets could isolate the waste from human intrusion or accidental release for many thousands of years. The disadvantages of ice sheet disposal are the long transport distances, the high costs and difficulties of operations in the polar areas, and uncertainties in the long-range interactions between the ice sheets and the waste. Furthermore, ice sheet waste disposal would require new international initiatives, an amendment of the Antarctic Treaty, or a new treaty with Denmark.

An extensive research program would be required to develop this concept and to resolve the questions of possible interactions between the emplaced waste and the ice sheets and of long-range weather variations and trends.

Island disposal This concept involves emplacement of solidified wastes within deep stable geologic formations beneath an island. Salt deposits are unlikely to be available at island sites; the most probable disposal formation is crystalline rock. Conventional geologic disposal concepts for crystalline rock would therefore be directly applicable to the island disposal concept. Waste from any fuel cycle option could be handled.

The major difference between mainland and island disposal sites lies in the geohydrologic regime and the requirement of sea transportation.

Other technologies, which are really not disposal technologies, but may conceivably contribute to improved disposal systems, have also been considered. These include chemical synthesis (followed by geologic disposal) and partitioning and transmutation, also a process step prior to disposal.

While theoretically each of the above alternatives may be argued to have some advantageous attribute, eg, isolation in space, they need not be considered further in this proceeding because a) they do not appear to provide any significant increment to public health or environmental protection, b) major research, development, and testing extending well beyond the period of the repository program would be necessary, and c) each appears to have potentially significant technical or institutional disadvantages.

C SUBSEABED DISPOSAL

The subseabed concept has been under active investigation since 1973, albeit on a comparatively small scale in terms of allocated resources.[29] The concept involves the emplacement of spent fuel or solidified HLW in appropriate forms and packages into the red clay sediments in the middle of a tectonic plate, eg, the North Pacific Plate, under the center of a surface circular water mass called a gyre. The major potential advantages cited for subseabed disposal in the mid-plate/gyre (MPG) are as follows:

- Ability to make long-term predictions of stability and uniformity on the basis of samples of sediments that have been accumulating continuously for 70 million years
- Lack of resources in the areas of interest, reducing likelihood of human intrusion or future resource conflict in the disposal area
- Plastic nature of sediments, which will allow closure of any openings, man-made or natural
- Low-permeability and high-sorption qualities of the sediments
- Continuously depositional nature of the sediments, eliminating the risk of erosion down to or including buried waste
- Large size of the areas of interest, of which approximately only 0.006 percent would be used for waste disposal
- Remoteness of the areas of interest from normal human activities
- Lack of need for mining activities or waste-handling facilities at the site
- Lack of need to resolve questions about Federal-State relations and authority over disposal sites.

The geologic history of the MPG regions as obtained from short (up to 10 m long) and long (24.5 m x 11.4 cm) cores shows them to be remarkably uniform in extent (over areas of 6000 square nautical miles) with no indications of damaging perturbations for tens of millions of years. The formations are highly predictable, a highly advantageous attribute. The red clays also have very high sorption coefficients (K_d) and therefore represent a major barrier to radionuclide migration.

Emplacement technology is considered to be essentially available state-of-the-art,[30] including retrievability (of at least small numbers of canisters) and location monitoring, and has been demonstrated in deep ocean drilling activities. However, engineering of the transport and emplacement systems needs to be undertaken on a specific basis. The major technical issue in the subseabed concept appears to be in the area of near-field interactions between the sediments and the heat and, to some extent, the radiation from the waste. In situ field tests to investigate these interactions are planned for the early 1980s. Although the sediments have some structure they are somewhat thixotropic, and up-welling due to the thermal source could adversely affect the sediment barrier. However, because the thermal-associated phenomena are believed to be only near-field and the spacing between spent fuel--or HLW--canisters would be on the order of 100 m, it seems that these phenomena would not have significant adverse effects on the overall system. Based on 0.5 MT of 10-year-old spent fuel per canister and the 100-m spacing in a square array, the maximum estimated thermal flux at the sediment/water interface due to the spent fuel is of the same order of magnitude as the normal flux for the MPG area (1-2 microcalories/cm^2/sec).[31]

If additional resources are made available to accelerate the program, there is a reasonable basis for suggesting the subseabed concept as a technically feasible alternative disposal system for use by the late 1990s. A Program Activity Chart for the subseabed program indicates completion of detailed site characterization of an initial site in the northern oceans by 1985-86, completion of detailed system design and issuance of construction permit by 1994, and acquisition of an operating license in 1998-99.[32]

However, it should be recognized that from an institutional and public acceptance standpoint one can anticipate major international opposition to the concept. Indeed, it has been suggested that implementation of the subseabed disposal concept might be in violation of existing US law and inconsistent with international laws. Accordingly, although it is apparent that work on this alternative should continue, it cannot be relied upon at this time as a potentially usable system for waste disposal.

D REFERENCE SYSTEM FOR THIS REPORT

As stated previously, the major features of a HLW disposal system are the waste form, which for the purpose of this proceeding has been specified to be spent fuel, and the specific ultimate disposal environment. While a number of disposal technology alternatives have been identified--and may, indeed, prove to be technically viable--the advanced state of knowledge regarding deep, conventional mined geologic disposal and, in particular, salt, logically dictates consideration of this alternative. Accordingly, Section III identifies and discusses the elements comprising a mined geologic disposal system and, in particular, describes how these natural and engineered barriers combine to provide containment.

REFERENCES FOR SECTION II

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

2. National Academy of Sciences/National Research Council, Pub. No. 519, The Disposal of Radioactive Waste on Land (1957).

3. R. Bradshaw and W. McClain, Project Salt Vault: A Demonstration of the Disposal of High-Activity Solidified Wastes in Underground Salt Mines, Oak Ridge National Laboratory (ORNL-4555, April 1971).

4. W. McClain and R. Bradshaw, Status of Investigation of Salt Formations for Disposal of Highly Radioactive Power Reactor Wastes, Nuclear Safety (Vol. 11, No. 2, March-April 1970).

5. Radioactive Waste Repository Project Technical Status Report, Oak Ridge National Laboratory (ORNL-4751, 1971).

6. National Academy of Sciences/Nuclear Regulatory Commission, Disposal of Solid Radioactive Wastes in Bedded Salt Deposits (1970).

7. W. McClain, Background and Review of Investigations in the Salina Salt Basin, Oak Ridge National Laboratory presentation to Michigan Environmental Review Board (1976).

8. FY 1974, Authorization Hearings before Joint Congressional Committee on Atomic Energy (March 22, 1973) (Statement of F.K. Pittman).

9. ERDA Press Release No. 75-51, ERDA to Issue Expanded Environmental Impact Statement on Management of Commercial High-Level and Transuranium-Contaminated Wastes (April 11, 1975).

10. Hearings on U.S. Energy Research and Development Administration FY 1976 Authorization Before the Joint Comm. on Atomic Energy, 94th Cong., 1st Sess. (February 1975) (statement of F.K. Pittman).

11. D.W. Powers, et al, Geological Characterization Report -- Waste Isolation Pilot Project, Sandia Laboratories (SAND 78-1596, 1978).

12. U.S. Department of Energy, Draft Environmental Impact Statement on the Waste Isolation Pilot Project, (DOE/EIS-0076-D, 1979).

13. President's Message to Congress on Radioactive Waste Management, Weekly Comp. of Pres. Doc. 303 (February 12, 1980).

14. U.S. Energy Research and Development Administration, Overview: National Waste Terminal Storage Program of the United States Energy Research and Development Administration (Press Conference, December 1976).

III Status of Technology for Disposal of Spent Fuel

In this section the status of technology for carrying out the various tasks involved in establishing and operating a safe and environmentally acceptable geologic disposal system for spent fuel is discussed and evaluated. Integral components of an overall waste management system will also include capabilities and facilities for interim storage and transport of spent fuel. These matters are discussed in the companion document entitled "Interim Storage of Spent Fuel".

In discussing and evaluating the technology for accomplishing the disposal of waste the general approach used is first to characterize for each component of the waste disposal system the technology status as viewed by UNWMG-EEI based on results of work in the United States and abroad. Then we identify alleged "gaps and uncertainties" regarding the technology along with the status of on-going Federal programs to resolve these "gaps and uncertainties". This is followed by our assessment of the significance or relevance of these questions, consideration of mitigating measures which could be employed if deemed necessary, and our conclusions in relation to the confidence we believe is justified in the technology.

The overall waste disposal system will be made up of a number of independent but interrelated subsystems, each of which can contribute in varying degrees and at various times to the overall performance of the system. In the following subsections these subsystems are discussed:

- A Site Identification and Characterization
- B Waste Form and Package
- C Additional Engineered Barriers
- D Repository Design and Construction
- E Waste Emplacement
- F Repository Closure
- G Post-closure Monitoring and Prediction of Long-term Repository Performance

Subsection H summarizes our conclusion that these subsystems, taken together as a total system, provide confidence that the waste can be effectively contained by the system.

A SITE IDENTIFICATION AND CHARACTERIZATION

1) Status of Technology

Proper site selection will provide a repository regime which has highly favorable characteristics for containment of the waste and which can reasonably be expected to maintain these favorable chacteristics over long periods of time (See Barrier #1, figure I-8 and figure III-H-1)

Much of the required technology for site identification and characterization has been developed and proven in other applications; for example, geologic research programs, identification and exploitation of mineral resources, geologic hazards investigations for a variety of industrial and defense facilities, geotechnology of underground construction for mining and transportation, and related activities. As a result of these experiences, the status of existing technology is well advanced, and gaps or uncertainties are largely limited to specific problems that are the topics of ongoing or planned research and investigation. No new technological breakthrough is required to permit site selection.

Technology for site identification is particularly well advanced. The DOE-USGS Earth Sciences Technical Plan (ESTP)[1] indicates that most technical questions in this area are at a stage where application of the existing techniques is required to determine whether additional refinements are needed. Although criteria for repository siting and/or performance are in draft stages and subject to significant changes, there appears to be adequate agreement regarding the issues of concern in criteria and in associated technical specifications that have been proposed by ONWI,[2] NRC,[3] IAEA,[4] and others.[5,6,7] Site identification involves evaluations of present or short-term geologic conditions and processes by using techniques that are well established in the state-of-the-art and include the following:

1) depth, thickness, and lateral extent of host rock,
2) avoidance of surface water bodies and of adverse hydrologic conditions in the aquifers overlying the proposed host rock that could preclude construction of the repository,

3) avoidance of areas having significant levels of tectonic, seismic, or igneous activity,

4) identification of homogenous stratigraphic/structural systems that can be defined adequately to permit modeling of the repository system, and

5) general short-term compatibility of host rocks with waste-induced chemical, thermal, and radiological stresses.

Uncertainties that are perceived by some to exist regarding the proper approach to selection of sites for a repository generally are related to:

1) the forecasting of longer term geologic processes and conditions,

2) the perceived limitations of geologic exploration techniques, and

3) the earth sciences aspects of repository design, construction, operation, and long-term risk assessment.

These are discussed below.

a) Approaches and Methods for Site Identification Geologic studies for selecting and characterizing repository sites can be divided naturally into three phases:

- Regional reconnaissance
- Feasibility study
- Detailed engineering investigation

Each phase has specific objectives, activities, and milestones that are interrelated with project decisions and licensing stages. In addition, each phase focuses on progressively smaller geographic areas with progressively greater detail obtained for different purposes. In this manner, large, unsuitable areas can quickly be excluded from further consideration, allowing continuing efforts to address those areas most likely to contain favorable sites.

As recommended in the ESTP,[8] the regional reconnaissance for repository siting can be very broad and actually can be nationwide in scope. This is a significant advantage in that many potentially adverse conditions can be avoided on a regional basis. As a result, siting can concentrate on regions where there is virtually no potential for breach of containment as a direct consequence of tectonism, volcanism, erosion/dissolution/deposition, or naturally occurring, nontectonic rock deformation. Such regions in fact have been the focus of NWTS siting studies[9] and of previous programs such as WIPP[10] and the earlier Project Salt Vault.[11] Within these suitable regions, the main emphasis in siting studies is on locating host rock of adequate thickness and extent, identifying closed (or effectively closed) geohydrologic systems, and avoiding existing or potential man-made conditions that could compromise containment.[12,13,14,15] It is generally accepted that suitable host-rock configurations can be identified, existing man-made penetrations or other adverse conditions can be located, and potential resources or other factors that would encourage future detrimental activities by man can be determined with reasonable assurance.[16]

Even though adverse tectonic and other conditions can be avoided on a regional basis during siting studies, the ability to clearly demonstrate the absence of such conditions is an important factor in the licensing process. Areas having positive evidence of past long-term geologic stability, therefore, would be most desirable. For example, continuous and undeformed geologic features (eg, strata, terrace forms, or erosional surfaces) that were formed at a known ancient time provide positive evidence that an area has not experienced deformation since that time. These features can also provide a basis for estimating erosion rates and history of processes such as dissolution.[17] This method of demonstration is well established in the geological sciences, has proven successful in licensing of nuclear power plants,[18, 19, 20] and should also be acceptable for estimating rates of geologic processes for licensing of repositories.

Following identification of areas having positive evidence of geologic stability within regions that are accepted as being stable, feasibility of candidate sites can be evaluated by established techniques of field exploration and laboratory testing. Emphasis in this phase is on detailed characterization of subsurface geology and hydrology in order to confirm site stability and to establish that the geohydrologic system would provide containment. Containment would be assured in a geohydrologic system with flow paths from the candidate repository to the biosphere that are sufficiently long to prevent releases of radionuclides for an adequate period of time.[21] Moreover, sites can be identified where flow paths are not toward the biosphere at all or where gradients are so low that groundwater is effectively immobile. Geochemical analyses can establish the length of time that groundwater has been isolated from the outside hydrosphere and thereby provide positive evidence for long residence time.[22]

Upon completion of the first two phases of investigation, project and licensing decisions can be made on the safety of the site with respect to geologic hazards and the adequacy of containment provided by the geologic environment to the overall system. The influence of all far-field phenomena should be determined by this time, with the issues of construction/operational safety to personnel and design details remaining for the third phase.

Detailed engineering studies involve extensive exploration, laboratory testing, and in situ tests at sites that have been found to be feasible on the basis of the previously described work. The design studies are concerned chiefly with determining the parameters and cost for construction and operation of the underground workings. Because of the detailed quantitative information required, results of studies may vary considerably over an area as large as that for a repository. However, the consequences of this variability will not significantly affect containment safety, but rather the economics of construction. Recent experience in underground construction for many purposes has demonstrated the ability to make adjustments in design to accommodate differences in conditions discovered during detailed investigations.[23,24,25]

b) <u>Status of Technology for Exploration and Characterization</u> Exploration techniques that would be used to identify and characterize sites for spent-fuel repositories are intended to determine qualitative and quantitative characteristics of four main environmental elements:

1) stratigraphy and geologic structure,
2) geologic and tectonic hazards,
3) geohydrologic systems, and
4) geotechnical properties (soil and rock mechanics).

The applicable techniques have been developed and proven for other geosciences applications, leading DOE and USGS to conclude that existing methods can provide adequate confidence for site characterization.[26] Some techniques are applicable to all elements, while others are of use only to small parts of a single element.

A single technique may be applied at various scales yielding different degrees of resolution. For most subsurface exploration programs a suite of techniques is chosen that will optimally address all objectives at appropriate stages. A well planned program will use combinations of techniques that are optimum and redundant at the right times and for the right purpose with the appropriate degree of detail. In this manner the limitations of any single method or combination of methods can be overcome, and the objectives satisfactorily reached with minimal uncertainty.

For example, seismic-reflection profiling can rapidly and relatively inexpensively (compared to drill holes) interrogate strata to great depth and lateral extent, describing the general location and configuration of major horizons and their larger discontinuities. However, this geophysical technique cannot detect all the thin beds, minor faults, or fractures that may be of concern. Detection and evaluation of such features require arrays of drill holes that provide cores of the formation rock for examination and testing and that permit downhole logging by a suite of radiometric and electric geophysical techniques. On the other hand, each drill hole interrogates only a very small area of the proposed repository environment and cannot begin to describe larger areas and features without many holes drilled at close spacing--undesirable intrusions of the repository environment. Thus, a combination of geophysical and drill-hole techniques would be appropriate, each one contributing data of unique quality and dimension in complementary fashion.

Exploration programs commonly progress in phases from larger areas, which are examined for large-scale features, to smaller areas which are scrutinized in greater detail for smaller features and also for collection of specific quantitative data. In this manner, large unsuitable areas can quickly be excluded from further consideration, allowing continuing efforts to address those areas most likely to be favorable where detailed data necessary for design can be justifiably sought. Such programs seek to completely avoid major geologic and tectonic hazards and to identify the general characteristics of geohydrologic systems at a regional to subregional scale. Later studies at specific sites and site-vicinities seek to confirm earlier finds; identify smaller hazards, such as shear zones, salt anticlines, breccia pipes, and evidence of dissolution; and evaluate geotechnical data on physico-chemical parameters of repository rocks that will influence design. At each succeeding step the likelihood of discovering major hazards or fatal flaws is greatly reduced, and those flaws that are found are very likely to be small and capable of mitigation.

Thus, the objectives of exploration change considerably at each phase, permitting major project and licensing decisions at each step without the need to wait until all the details are available for the very last step. For example, decisions on adequacy of a site with respect to far-field hazards (ie, faults, earthquakes, igneous activity, etc) and general geohydrologic conditions could be made in early phases. Later phases would develop quantitative parameters for design and construction engineering and would not contribute significantly to the basis for decisions on far-field hazards.

Collective confirmation of all geologic conditions will be made throughout the exploration and operation programs. Consequently, there are many opportunities (eg, mapping of underground openings, in situ testing, and long-term monitoring) to confirm and supplement exploration data[27] and thereby reduce the significance of limitations of any particular technique.

c) <u>Related Experience</u> The extensive experience of the electrical utility industry in selecting and licensing sites for nuclear power plants[28,29] can be applied to site selection for repositories. The lessons learned in these activities add to the confidence that geologic stability can be established to meet licensing criteria at properly selected sites. Although spent fuel repositories require assurance of containment for long periods of time, and investigation of deep hydrologic and geochemical systems must be made in greater detail, similarities with nuclear power plant siting do exist. For example, both types of facilities are sensitive to geologic processes and both are subject to intense regulatory reviews. Furthermore, siting studies for both nuclear power plants and spent fuel repositories begin with systematic regional surveys to select potentially favorable candidate sites by evaluating a number of tectonic and geologic factors. Because of these similarities, power plant experience is applicable to the search for repository sites, thus adding to our confidence that suitable sites can be identified and that their acceptability can be demonstrated in the licensing process.

Dealing with geologic issues in the licensing process has been a most important aspect of industry experience with nuclear power plants. For example, reactor license applicants have learned how geologic facts can be determined and presented to licensing agencies with a high degree of reliability (see, for example, ref 20). They have also learned to recognize geologic issues that cannot be resolved with adequate assurance for the licensing process and should, therefore, be avoided in site selection.

It should be noted that the greater freedom available in seeking repository sites provides additional confidence that such sites can be identified and licensed. Power plant siting is usually constrained by the service area of a utility, the need to limit transmission distances, and the plant's requirements for large amounts of cooling water. In comparison, repository sites can be selected from much larger areas and can avoid geologic or hydrologic conditions that may be potentially adverse or difficult to resolve with confidence. This ability to avoid problem areas greatly adds to confidence that demonstrably suitable sites can be identified.

Many types of underground facilities (eg, mines, transportation tunnels, power plants, defense facilities) have been successfully constructed and operated despite limitations of exploration techniques.[30] It should be kept in mind that most of these facilities were constructed without much choice of location and certainly without the requirements for uncomplicated geologic configurations and exceptionally high rock quality that characterize spent fuel repository siting. In addition, the stable geologic and tectonic environments that are preferred for repository siting minimize the potential for fracturing, deformation, or other undesirable rock conditions. The worldwide history of underground construction has demonstrated the adequacy of exploration techniques to develop subsurface data and the ability of design and construction techniques to accommodate the uncertainty inherent in underground conditions. It follows that the geologic environment for a repository will be inherently much more favorable than for other underground facilities. Therefore, adjustments during construction for unforeseen geologic-geotechnical conditions will be fewer and easier to accomplish.

d) Ability to Forecast Long-term Geologic Stability In the context of evaluating site suitability for a proposed repository, it is well to remember that in Section I of this report we have shown that the degree of containment required of a repository system significantly exceeds that required for an ore body for only a few hundred years. Therefore the finding of confidence does not depend to a great extent on the ability to predict geologic stability far into the future. Even so, our capability to do so is significant. The geologic processes that could influence containment may be classified as:

- Tectonic movements
- Igneous activity
- Nontectonic rock deformations (diapirism, salt flow)
- Erosion/dissolution/deposition
- Groundwater movement
- Climate and related changes

A study performed for EPRI[31] has compiled available data on the occurrence and rates of movement, change, or propagation of these processes, and has applied these data to estimate the effect of each process operating over periods of 1000 to 1,000,000 years. Rate data compiled in that study are summarized in table III-A-1. These data indicate that most of the natural geologic processes operate at rates such that containment in a mined repository would not be reduced significantly over periods of several hundred thousand to a million years or more. For example, erosion at the rate which has occurred in the Grand Canyon region of the Colorado River would require much more than a million years to reach the depths that typically are proposed for a mined repository.

Results of the EPRI study also indicate that regions can be identified where the relevant geologic processes have operated very slowly and without significant change in rate for periods so much longer than the time of concern for the repository that the uncertainty of the required continued performance is very small. In such areas there is virtually no potential for loss of containment as a direct result of tectonism, igneous activity, nontectonic rock deformation, or erosion/dissolution/deposition over periods up to more than a million years. Because we expect repository sites to be located in such areas, and to be deeply buried, the first four processes no longer continue to be significant issues after the siting phase.

Groundwater movement is the most probable and most rapid mechanism for potential transport of radionuclides from a repository.[32,33,34,35] However,

Table III-A-1
Summary of Rates for Geologic Processes

<u>Tectonic Activity:</u>

Plate tectonic movements (Relative motion of plates occurring across plate boundaries)-- 1.5 to 16 cm/yr - 2.6 cm/yr average (ref 1)* (Rates averaged over millions of years.)

Rates of slip on individual strike-slip faults, averaged over thousands to millions of years - less than 0.1 cm/yr to 6.6 cm/yr. (ref 2)

Uplift a) To 0.08 cm/yr over periods of 120,000 to 450,000 yr in large areas of Southern California (ref 3 & 4)
b) 1 cm/yr over last 45,000 years in Ventura area (ref 3)
c) 1.8 to 3.6 cm/yr in Transverse Ranges over periods of few years: 0.43 to 0.48 cm/yr average over last 100 years (ref 5)

<u>Climate Changes:</u>

Sea level: Fluctuates over range of 85 to 140 meters (average range estimated at 100 meters) with change occurring at rate of 0.1 to 1 cm/yr (ref 8)

<u>Surface Deformations:</u>

Salt Dispirism: Less than 0.003 to 0.2 cm/yr averaged over millions of years (ref 6)

Compaction from withdrawal of water or oil: 1 to 33 cm/yr for periods up to about 35 years (ref 7)

<u>Surface Processes:</u>

Erosion: (Lowering of land surface) - 0 to 0.04 cm/yr (810 m in 2.1E+06/yr. on lower Colorado River) (ref 9)

0.0002 to 0.08 cm/yr (based on historic sediment loads in various streams in the US) (ref 10)

Deposition: (Increased height of land surface) - 0.01 cm/yr in Western US desert basins, averaged over 3 m y (ref 11) to 0.2 cm/yr Offshore of Gulf Cost, averaged over estimated 2.5 m y *ref 12)

Salt Dissolution: - 0.005 to 0.016 cm/yr of dissolution (vertical component) averaged over drainage basins in southeast New Mexico, based on present-day dissolved salts (ref 13)

- 0.01 cm/yr at solution-depression in southeast New Mexico, averaged over 600,000 years (ref 14)

<u>Groundwater Movement:</u>

16,000 cm/yr	(poor confining formation)
6.3 cm/yr	(good confining formation)
0.3 cm/yr	(highly confining formation)

Rates are calculated on basis of representative permeabilities and assumed hydraulic gradients (ref 15); actual conditions will vary.

* References for Table III-A-1 are listed separately at the end of Section III-A.

areas can be identified where rates of movement and lengths of flowpaths to the biosphere are suitable to contain radionuclides for periods of thousands to millions of years.[36,37] Although changes in a groundwater regime may occur in response to changes in other geologic processes, a location that is stable with regard to other processes also can be expected to be stable with regard to groundwater hydrology.

Movement of groundwater is controlled by:

1) hydraulic conductivity,
2) hydraulic gradient, and
3) flow paths.

Of these parameters, hydraulic conductivity and flow paths are characteristics of the host medium, and they would remain constant unless altered by tectonic or other deformations. As discussed previously, regions can be identified where any significant deformations would be extremely unlikely to occur for periods of a million years or longer.[38]

In geologically stable areas, climate is the main natural variable influencing long-term groundwater movement. Precipitation rates may alter surface hydrology in source areas, and gradients may change because of climate-related changes in base levels. However, the geologic record over Quaternary time (approximately the last 1.8 million years) indicates the ranges in climate-related conditions that may occur during the next several hundred thousand years or more,[39,40,41] and these should provide a reasonable basis for analyzing groundwater movement over time periods of importance to spent-fuel repositories.

We believe the technology is totally in hand to select sites where geologic stability can be assured for time periods well in excess of that required. Nonetheless a number of alleged gaps and uncertainties have been raised by some. These are considered in the following subsection.

2) Alleged "Gaps and Uncertainties"

Gaps and uncertainties are perceived by some earth scientists to exist in the present status of technology for identifying and characterizing candidate repository sites, at least to the extent that the solutions for some problems are not uniformly recognized in the technical community. In general, these issues are related to the forecasting of long-term future geologic conditions and processes, the perceived limitation of the available techniques for geologic exploration, and incomplete knowledge of candidate host-rock characteristics. The perceived geologic uncertainties relating to site identification and characterization may be classified as:

1) Capability of geologic, hydrologic, and geotechnical exploration techniques to determine subsurface conditions with adequate reliability to assure feasibility of constructing and operating a repository and containment of radioactive materials.
2) Ability of geoscientists to forecast geologic and hydrologic conditions as much as hundreds of thousands of years in the future with adequate confidence to assure continued containment of the radioactive materials.
3) Incomplete knowledge of the pertinent properties of candidate host rocks and other rock types along potential pathways from a repository to the biosphere.
4) Potentials for reduced containment resulting from disruptions to the geologic environment by exploration, construction, and operation of a repository and by processes (such as convection) that may be induced by the storage of radioactive material.

While we recognize that the existing state of knowledge may be incomplete in each of these categories, we consider it to be fully adequate for identifying potential repository sites and for preliminary site characterization and evaluation. Existing knowledge is also adequate for preliminary engineering analyses, as long as conservative bounding estimates are used for parameters that may be uncertain. We consider this approach to be sound engineering practice for all types of civil construction and not unique to the disposal of radioactive materials. Factors pertinent to each of the alleged gaps and uncertainties are discussed in the following paragraphs.

a) Capability of Exploration Techniques Many authorities consider present technology to be adequate for site characterization;[42] nevertheless, questions have been raised regarding the ability to detect fractures and other discontinuities that may be potential flow paths.[43] While such features might influence construction and operation of the repository, their main significance is in relation to long-term containment. The fact that numerous mines and other underground facilities have been developed successfully by use of existing technology (or less-advanced technology in many cases)[44] provides assurance that a repository can be identified and characterized using available exploration methods. The possible existence of undetected fractures would be unlikely to influence near-term containment because effects can be mitigated easily by selecting sites that do not have potential for unacceptable water flows under present hydrologic conditions. In view of these factors, the most important implication of undetected fractures or other discontinuities is that they conceivably may facilitate groundwater flow under stresses induced by repository construction or operation or by changed hydrologic conditions in the future.

The NWTS program[45] includes research to improve exploration techniques and methods of analysis and risk assessment. The exploration research emphasizes development of geophysical methods that would improve recognition of small-scale features without excessive boreholes (ref 1, tasks 2.1.1, 2, and 3). Research on analysis and risk assessment chiefly involves fracture flow, fluid migration and transport, and systems analysis.[46]

We believe any present uncertainties can be largely overcome by conservatism in site selection and design of facilities and by improving techniques through the additional research being conducted by the NWTS program. We confidently expect that the results of research and additional site-specific experience will further reduce uncertainties in this area. Limitations of available exploration technology are therefore not a significant deterent to successful accomplishment of the NWTS program.

b) Long-term Forecasting of Geologic and Hydrologic Stability A number of studies have questioned the ability of geoscientists to forecast conditions hundreds of thousands of years in the future.[47,48,49,50] The implication of these uncertainties is that future tectonic or igneous events, climate changes, or other phenomena could reduce containment to unacceptable levels. As concluded in the preceding discussion of status of technology in long-term forecasting, available data on rates of geologic processes indicate that such concerns are unfounded. Instead, the data indicate that geologic processes operate at rates that would not compromise containment over periods of hundreds of thousands to a million years or more and that regions can be identified wherein tectonic processes, igneous activity, erosion, and other processes would not be a hazard to a repository.[51]

There are a number of tasks in the NWTS program that address earthsciences aspects of long-term risk analysis (ref 1, WBS 4.0). It is particularly significant that most tasks in this area address specific sites. We agree that long-term geologic stability can and should be assessed on a site-specific basis.

Forecasting of geologic conditions over the time period during which a high degree of containment is required is a significant factor in demonstrating site suitability. However, the weight of data from geologic research programs,[52] investigations for the WIPP site,[53] and for nuclear power plants in various parts of the Continental United States (see, for example, ref 54, 55, 56) provides confidence that areas of past geologic stability can be identified and that these can provide assurance of adequate future stability, and as we have shown in Section I (supra) the time period over which a degree of containment higher than that of naturally occurring ore bodies is required only for a short time (few hundred years).

c) Incomplete Understanding of Rock Characteristics While understanding of rock properties is fundamental to site identification and characterization, the largest part of this work can be done most effectively on a site-specific basis and should not be a reason for deferring site-identification activities. Because of the inherent variability of stratigraphy and lithology among different sites, there is no amount of generic information that could greatly reduce the need for detailed studies at specific sites. As a result, it would be most effective to proceed with site identification and to conduct the required testing on candidate host rocks. In the event that the sites were found acceptable, such testing would expedite site characterization. If not, it would contribute to the generic data base. Nevertheless, the ESTP[57] notes that major research efforts are in progress in this area, including both generic studies and testing on candidate host rocks from specific sites.

d) Potentials for Induced Reductions in Containment Concerns have been expressed that exploration, construction, and operation of a repository may disrupt the geologic environment to the extent that containment would be compromised. Questions have also been raised as to whether the thermal and radiation effects from the stored radioactive material could induce processes such as convection or fluid migration that would reduce containment unacceptably.[58]

Possible disruptions include the effects of exploratory boreholes, construction excavations, and subsidence of the rocks overlying the mined cavities. Concerns have been expressed that fracturing induced by these disruptions may facilitate groundwater movement through the repository and that incomplete sealing of penetrations may provide a relatively open pathway to the biosphere. Thermal loading is thought to have potential for inducing fracturing in brittle rocks and for increasing rates of plastic flow in salt or shale. In general, these concerns relate to rock mechanics, geochemistry, fluid flow, and waste-media interactions.

At present, the largest part of the NWTS program addresses the earth science aspects of repository design, construction, and operation[59] and is attempting to resolve these concerns. Because of the diversity of issues that are involved in potential disruptions to the environment and induced processes, research and testing in many areas will be applicable to this concern. In addition, modeling efforts will provide a basis for predicting the effects on containment.

To overcome the uncertainties of induced reductions in containment, site-specific investigation and modeling will be needed to quantify the problems, and conservatism will be needed in site selection and repository design. Because analyses of these effects demand site-specific information, the uncertainties in this area require that site identification proceed in a timely manner, rather than being a reason for delay.

The nature and significance of the induced effects will be determined by engineering design of the repository in relation to site-specific conditions. Engineering analyses, modeling, and design approaches will maintain these effects within conservative bounds. For example, analyses for the WIPP[60] determined that the maximum temperature increase resulting from the storage of spent fuel would be 17 C in the floor of the emplacement room and would occur about 25 years after emplacement. A temperature increase of this magnitude would seem unlikely to cause drastic changes in rock properties.

The geologic conditions of a carefully selected repository site will inherently minimize the possible reductions in containment resulting from induced processes and will facilitate modeling of their effects. Specifically, a desirable repository site would contain uniform host rock of significant thickness and lateral extent, uncomplicated geologic structure, and generally would be in a dry environment.[61,62] The nature and extent of the host rock are intended to contain and buffer effects of repository construction and operation, while uniformity and uncomplicated geologic structure would simplify modeling. The generally dry environment would tend to assure that unacceptable migration of radionuclides would not occur even if the geologic barriers were to be disrupted. In combination, these factors provide a high level of confidence that integrity of the repository environment can be preserved throughout exploration, construction, and operation of a repository.

For the reasons discussed above we conclude that these perceived gaps and uncertainties do not impede the selection and characterization of a suitable site now. Moreover, as discussed in the following paragraphs much work in these areas is underway here and abroad.

3) <u>The NWTS Program</u>

The status and organization of the NWTS program are described in detail in the DOE Statement of Position,[63] and earth science aspects have also been summarized previously in the ESTP.[64] Following the Work Breakdown Structure (WBS) in the ESTP, activities in the earth sciences are classified as:

- o Identification and Evaluation of Potential Geologic Environments
- o Site Characterization
- o Earth Science Aspects of Facility Design, Construction, and Operational Risk
- o Earth Science Aspects of Long-term Risk Analysis
- o Coordination and Review.

At present, the greatest levels of effort are in identification and evaluation of environments on a regional scale and in testing for the earth science aspects of design, construction, and operation. Much of the testing is on generic materials or at test sites, rather than being directly for candidate repository sites. As potential sites are identified, we would expect the level of effort in site characterization to increase and become by far the largest part of the program.

The schedule for the NWTS program (ref 63, figure III-2) calls for identification and qualification of sites in various regions and media over the next 5 years and selection of site(s) for repository construction in 1987. Earliest repository operation is scheduled for 1997. This schedule seems readily achievable.

4) International Activities and Experience

A number of countries outside the US have significant efforts underway for identification and characterization of potential repository sites. Table III-A-2 is a summary of international activities largely based on information compiled by Battelle Pacific Northwest Laboratories[65] and the IRG.[66] While several countries have conducted general studies for feasibility and site identification, the most significant work has been done by Canada, Federal Republic of Germany, and Sweden.

An overview study conducted for the Canadian Government[67] indicated confidence that nuclear waste can be disposed of safely. Preliminary analyses, using representative parameters for hard rock, identified no insurmountable construction problems. It was considered likely that sites with good containment properties could be found in the Canadian Shield, but site-specific data would be needed for reliable analyses of long-term containment.

Federal Republic of Germany (W Germany) has been one of the leading nations in disposal of radioactive waste.[68] Significant experience has been obtained from waste disposal and experiments at the Asse Salt Mine. This facility has been used as a repository for low-level waste since 1967 and for intermediate level wastes since 1972; since 1976 it has been used as a test facility for storage of high-level waste.[69]

The Swedish experience in evaluating feasibility of nuclear waste disposal is particularly significant. A new law, the "Stipulations Law," required the

Table III-A-2

Summary of International Activity in Site Identification and Characterization

Country	Activity
Belgium	- evaluating storage in Boom Clay formations at Mol; results of preliminary tests indicate general feasibility[65] - planning heater tests and construction of test chamber[65]
Canada	- results of over-view study[67] indicate confidence that acceptable sites can be identified in Canadian Shield region - have identified many potential sites in granite and some in salt[65] - have established test site in granite[65]
Finland	- evaluating feasibility of repository in crystalline rock[67]
France	- presently studying salt and crystalline rock with emphasis on the latter, although several promising areas in salt also have been found[65,66] - have evaluated site in granite at La Hague[65]
Germany (Fed Rep)	- one of the leading nations in radioactive waste disposal[66] - have been storing waste with low TRU content in salt mine at Asse on routine basis[66] - evaluating other sites in salt[65]
India	- surveying igneous and sedimentary rocks for potential sites[65,66]
Italy	- evaluating clayey sediments near Trisaia Centre in Southern Italy[65] - performing in situ tests of thermal and radiation effects[65]
Japan	- does not permit land burial of waste at present[66] - have identified potential sites in granite and zeolite formations and planning additional studies[65]
Netherlands	- evaluating salt dome disposal[65] - have selected sites for exploratory drilling[65]
Spain	- have stored non-HLW in abandoned iron mine for several years[65] - conducting search additional sites[65]

Table III-A-2 (continued)

Country	Activity
Sweden	- KBS report recommending disposal in granite has been reviewed extensively and accepted by the Swedish Government as ade- quate to demonstrate that spent fuel can be disposed of safely[65] - work is continuing to qualify a repository site[65] - R&D studies have been conducted at Stripa Test Station, Studsvik Energy Centre, and Chalmers University[65]
Switzerland	- potential site has been located in granite at Olten[65]
United Kingdom	- has ongoing program evaluating disposal in clay and granite[66] - have identified a number of sites (in crystalline rock, in clay and in evaporite beds) for drilling in situ testing[65] - have developed conceptual design for repository in granite
USSR	- evaluating geological disposal in salt[65] - presently emphasizing surface storage[66]

owners of nuclear reactors to demonstrate that spent fuel could be disposed of safely before the government would permit fuel to be loaded and operations to commence.[70] In response to this law, a group of four Swedish nuclear power utilities formed the Swedish Nuclear Fuel Safety Project (KBS), which issued a report proposing geologic disposal in granite. After extensive evaluation, this report was accepted by the Swedish Government as adequate to satisfy the law and allow fuel loading and reactor operations to commence; site-qualification studies are continuing.[71]

In summary, evaluations of the relevant work by other nations support the findings by groups in the United States that suitable sites can be identified and characterized adequately and that spent fuel and other radioactive materials can be disposed of safely. While specific results of technical studies are not completely available, the number of countries that have identified potential sites (table III-A-2) suggests that a very large number of favorable sites exist throughout the world and further supports confidence in safe disposal.

5) Conclusions

The UNWMG-EEI has carefully reviewed the status of technology, the alleged gaps and uncertainties, and the national and international programs for identifying and characterizing sites for spent-fuel and HLW repositories. In our view a high level of confidence exists that sites for repositories can be identified and characterized in a timely manner and with adequate assurances of safe construction, operation, and long-term containment. Proven methods and technology are available for the largest part of the required work and, most significantly, for the preliminary needs of site identification. The perceived gaps and uncertainties relate chiefly to the misunderstanding of the period of time over which containment significantly greater than that provided by natural ore bodies is needed and to the current lack of site specific data. The NWTS program for site selection and characterization is highly likely to identify favorable candidate sites.

REFERENCES FOR SECTION III-A

1. Earth Science Technical Plan for Disposal of Radioactive Waste in a Mined Repository. Office of Nuclear Waste Management, US Department of Energy, and US Geological Survey, Department of Interior (DOE/TIC-11033 (Draft), April 1980).

2. NWTS Criteria for the Geologic Disposal of Nuclear Waste: Site-Qualification Criteria. (Draft) (ONWI-33 (2), 1980).

3. 10CFR60. Disposal of High Level Radioactive Wastes in Geologic Repositories, Parts A-D (Draft) Federal Register (December 6, 1979): Parts E - I drafts 7 and 10, unpublished.

4. Site Selection Factors for Repositories of Solid, High-Level and Alpha-Bearing Wastes in Geological Formations. IAEA Technical Report No. 177 (1977).

5. Geological Criteria for Radioactive Waste Repositories, G.D. Brunton and W.C. McClain (Y/OWI/TM-47, 1977).

6. Geological Criteria for Repositories for High-Level Radioactive Wastes. National Academy of Sciences - National Research Council (1978).

7. Implementation of Long-Term Environmental Radiation Standards: The Issue of Verification. National Academy of Sciences - National Research Council (1979).

8. Op cit, DOE, USGS (Ref. 1).

9. M. Kehnemuyi, The Geologic Repository - Where, How, When? (ONWI-62, p 7-8, 1979).

10. Draft Environmental Impact Statement, Waste Isolation Pilot Plant. (DOE/EIS - 0026-D, 1979).

11. R.L. Bradshaw and W.C. McClain, Project Salt Vault: A Demonstration of the Disposal of High-Activity Solidified Wastes in Underground Salt Mines. (ORNL-4555, 1971).

12. G.D. Brunton, R.B. Laughon, and W.C. McClain, Screening Specifications for Bedded Salt, Salina Basin, New York and Ohio. (Y/OWI/TM-54, 1978).

13. G.B. Griswold, Site Selection and Evaluation Studies of the Waste Isolation Pilot Plant (WIPP), Los Medanos, Eddy County, New Mexico. Sandia Laboratories Report (SAND-77-0946, 1977).

14. F.R. Conwell, Regional Geological Screening Studies for Waste-Repository Siting in the Paradox Basin. (ONWI-62, p 70-72, 1979).

15. C.O. Durham, Jr, A Review of Gulf Coast Salt-Dome Evaluations. (ONWI-62, 1979) p. 77-78.

16. Isolation of Radioactive Wastes in Geological Repositories: Status of Scientific and Technical Knowledge (Draft). Office of Science and Technology Policy, Subgroup for Alternate Technology Strategies, Working Paper for the Review Group on Nuclear Waste Management (1978).

17. G.O. Bachman, Geologic Processes and Cenozoic History Related to Salt Dissolution in Southeastern New Mexico. US Geol. Survey Open-File Report 74-194 (1974).

18. J.L. Smith and J.D. Scott, A Systematic Approach to Nuclear Power Project Siting (paper presented at the ANS/INEN meeting in Mexico city) (September 1975).

19. E. Van Brunt, Jr, L.T. Klein, and J.L. Smith, Site Selection for the Arizona Nuclear Power Project (paper presented at the annual meeting of the Atomic Industrial Forum) (1974).

20. Geology in the Siting of Nuclear Power Plants. Geological Society of America, Engineering Geology Volume IV (1979).

21. State of Geological Knowledge Regarding Potential Transport of High-Level Radioactive Waste from Deep Continental Repositories. Report of Ad Hoc Panel of Earth Scientists (EPA/520/4-78-004, 1978).

22. Op cit, OSTP (Ref. 16).

23. M. Dolcetta, "Problems with Large Underground Stations in Italy". ASCE Symposium on Underground Rock Chambers, (1971) p. 243-286.

24. R.R. Rising and G.A. Erickson, "Design of Underground Pumping Chamber at Lake Mead." ASCE Symposium on Underground Rock Chambers, (1971) p. 381-405.

25. J.R. Swisgood and R.E. Versae, "Geotechnical Investigations for Mine Shafts." Mining Engineering, v 24, no. 6 (1974).

26. Op cit, DOE, USGS (Ref. 1).

27. Op cit, J.R. Swisgood (Ref. 24).

28. Op cit, J.L. Smith (Ref. 18).

29. Op cit, E. Van Brunt, Jr. (Ref. 19).

REFERENCES FOR SECTION III-A (CONT)

30. T.A. Lang, "Underground Rock Structures Challenge the Engineer." ASCE Symposium on Underground Rock Chambers, (1971) p. 1-20.

31. J.L. Smith and M.A. Balderman, Geologic Processes and Site Evaluation for Radioactive Waste Repositories (Draft). Report prepared for Electric Power Research Institute (1980).

32. Op cit, IAEA Technical Report No. 177 (Ref. 4).

33. Op cit, NAS/NRC (Ref. 6).

34. Op cit, EPA/520/4-78-004 (Ref. 21).

35. Op cit, Office of Science and Technology Policy (Ref. 22).

36. Op cit, EPA/520/4-78-004 (Ref. 21).

37. G. de Marsily, E. Ledoux, A. Barbreau, and J. Margat, Nuclear Waste Disposal: Can the Geologist Guarantee Isolation? Science, v. 197 (1977) p. 519-527.

38. Op cit, J.L. Smith (Ref. 31).

39. C. Emiliani, Quarternary Hypsithermals. Quaternary Research, v. 2 (1972) p. 270-273.

40. R.W. Fairbridge, "Climatology of a Glacial Cycle." Quaternary Research, v. 2 (1972) p. 283-302.

41. J. Kukla, R.K. Matthews, and J.M. Mitchell, Jr., "The End of the Present Interglacial." Quaternary Research, v. 1 (1972) p. 261-269.

42. Op cit, DOE, USGS (Ref. 1).

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

44. Op cit, T.A. Lung (Ref. 30).

45. Op cit, DOE USGS (Ref. 1).

46. Id.

47. Op cit, DOE (Ref. 43).

48. Op cit, DOE USGS (Ref. 1).

49. "Report to the President by the Interagency Review Group on Nuclear Waste Management" (TID-29442, March 1979).

REFERENCES FOR SECTION III-A (CONT)

50. J.D. Bredehoeff and others, "Geologic Disposal of High-Level Radioactive Waste--Earth Science Perspectives." US Geological Survey Circular - 779 (1978).

51. Op cit, J.L. Smith (Ref. 31).

52. Id.

53. Op cit, DOE (Ref. 10).

54. Boston Edison Company, "Preliminary Safety Analysis Report, Pilgrim Nuclear Plant Unit 2."

55. Washington Public Power Supply System, "Preliminary Safety Analysis Report, WPPSS Nuclear Project Unit 1."

56. San Diego Gas and Electric Company, "Preliminary Safety Analysis Report, Sundesert Nuclear Plant, Units 1 and 2."

57. Op cit, DOE, USGS (Ref. 1).

58. Op cit, DOE (Ref. 43).

59. Op cit, DOE, USGS (Ref. 1).

60. Op cit, DOE (Ref. 10).

61. Op cit, ONWI-32(2) (Ref. 2).

62. Op cit, 10CFR60 (Ref. 3).

63. U.S. Nuclear Regulatory Commission, In the Matter of Proposed Rulemaking on the Storage and Disposal of Nuclear Waste, Statement of Position of the United States Department of Energy (DOE/NE-0007 April 15, 1980).

64. Op cit, DOE, USGS (Ref. 1).

65. K.M. Harmon, Summary of National and International Waste Management Programs. Battelle Pacific Northwest Laboratory (PNL-2941, 1979).

66. Interagency Review Group, Report to the President by the Interagency Review Group on Nuclear Waste Management, Appendix G (TID-29442, March 1979).

67. Acres Consulting Services Limited, Radioactive Waste Repository Study Part III, Summary (ACLS-6188-3, 1978).

68. Op cit, Interagency Review Group (Ref. 66).

REFERENCES FOR SECTION III-A (CONT)

69. T. English, et al, An Analysis of High-Level Radioactive Waste and Spent Fuel Management Systems, Jet Propulsion Laboratory (JPL-77-69, 1977).

70. Op cit, Interagency Review Group (Ref 66).

71. Op cit, Harmon (Ref. 65).

REFERENCES FOR TABLE III-A-1

1. B. Isacks, et al, Seismology and the New Global Tectonics, J. Geophys. Res. (Vol. 73, 1968), pp. 5855-5899.

   X. LePichon, Seafloor Spreading and Continental Drift, J. Geophys. Res. (Vol. 73, 1968), pp. 3661-3697.

2. Woodward-Clyde Consultants, Report of the Evaluation of Maximum Earthquake and Site Ground Motion Parameters Associated with the Offshore Zone of Deformation, San Onofre Nuclear Generating Station, Report to the Southern California Edison Company (1979).

3. K.R. Lajoie, et al, "Quarternary Marine Shorelines and Crustal Deformation, San Diego to Santa Barbara, California," in Geological Excursions in the Southern California Area (P.L. Abbott ed. 1979), pp.3-16.

4. P.J. Kern, Origin and History of Upper Pleistocene Marine Terraces, San Diego, California, Geol. Soc. America Bull. (Vol. 88, 1977), pp. 1553-1566.

5. D.A. Rodgers, Vertical Deformation, Stress Accumulation and Secondary Faulting in the Vicinity of the Transverse Ranges of Southern California, California Division of Mines and Geology, Bull. No. 203 (1979).

6. F. Gera, Review of Salt Tectonics in Relation to the Disposal of Radioactive Wastes in Salt Formations, Geol. Soc. America Bull. (Vol. 83, 1972), pp. 3551-3574.

   J.D. Martinez, Tectonic and Hydrologic Stability of Louisiana Salt Domes, Proceedings of the National Waste Terminal Storage Program Information Meeting (ONWI-62, 1979), pp. 80-83.

7. J.F. Poland, "Subsidence and Its Control," in Underground Waste Management and Environmental Implications (T.D. Cood ed. 1972), pp. 50-71.

8. R.F. Flint, Glacial and Quarternary Geology, John Wiley & Sons, New York, N.Y. (1971).

   R.D. Steiner, et al, Eustatic Low Stand of Sea Level Between 125,000 and 105,000 B.P.: Evidence from the Subsurface of Barbados, West Indies, Geol. Soc. America Bull. (Vol. 83, 1973), pp. 63-70.

   N.J. Shackleton and N.D. Opdyke, Oxygen Isotope and Palaeomagnetic Stratigraphy of Equatorial Pacific Core V28-238; Oxygen Isotope Temperatures and Ice Volumes on a 100,000 and 1,000,000 Year Scale, Quarternary Research (Vol. 3, 1973), pp. 39-55.

REFERENCES FOR TABLE III-A-1 (cont.)

9. P.E. Damon, et al, Revised Chronology for Critical Stages in Evolution of the Lower Colorado River, Geol. Soc. America Abstracts with Programs (Vol. 10, 1978), pp. 101-102.

10. R.W. Fairbridge, The Encyclopedia of Geomorphology, Rheinhold Publ. Co., New York, N.Y. (1968).

11. A.J. Eardley, et al, Lake Cycles in the Bonnevile Basin, Utah, Geol. Soc. America Bull. (Vol. 84, 1973), pp. 211-216.

12. B.R. Sider, et al, "Late Pleistocene Geologic History of Texas Outer Continental Shelf and Upper Continental Slopes," in Framework, Facies, Oil-Trapping Characteristics of the Upper Continental Margin (A.H. Bouma, G.T. Moore, J.M. Coleman eds. 1978), pp. 243-266.

13. G.O. Bachman and R.B. Johnson, Stability of Salt in Permian Salt Basin of Kansas, Oklahoma, Texas, and New Mexico, U.S. Geological Survey (Open File Report No. 4339-4, 1973).

14. G.O. Bachman, Geologic Processes and Genozoic History Related to Salt Dissolution in Southeastern New Mexico, U.S. Geological Survey (Open File Report No. 74-194, 1974).

15. G. de Marsily, Nuclear Waste Disposal: Can the Geologist Guarantee Isolation?, Science (Vol. 197, 1977), pp. 519-527.

16. Op cit, Le Pichon (Ref. 1).

B WASTE FORM AND PACKAGE

The waste form represents Barrier #2 and the package, Barrier #3 (see Figure I-8 and Figure III-H-1). The waste package (canister) provides important containment during emplacement and any retrieval period deemed necessary. It can also be expected to provide a high degree of containment in the important early years when fission product content is controlling. The waste form can be expected to provide containment in the form of a low leach rate, should water ever reach the waste, for very long time periods.

1) Status of Technology

a) Spent Fuel as a Solid Waste Form As noted earlier in this section and elsewhere, it is recognized that the focus of these hearings is on spent fuel as a waste form. Since extensive work has been carried out on glass (vitrified HLW) waste forms, it is deemed appropriate to discuss spent fuel as a waste form on the basis of its comparative stability to a reference borosilicate glass waste form.

The overall physical and chemical characteristics of spent LWR fuels have been investigated extensively.[1] Current LWR fuels use Zircaloy tubes (cladding) which are filled with UO_2 pellets. During operation of the reactor various physical and chemical changes take place in the UO_2 fuel and cladding because of the neutronic and thermal actions resulting from the fission process. Radioactive fission products are formed and migrate within and are released from the fuel pellets. Fuel pellets crack and interact with the Zircaloy cladding, which in turn may embrittle and grow. These physical and chemical characteristics of spent fuel are described in summary fashion by Houston.[2] The solution chemistry of UO_2 (uraninite), and associated compounds depending on oxidation state, is also described, in summary, by Holland and Brush.[3] They observe, for example, that the solubility of UO_2 at 25 C in water equilibrated with air is about 2E-05 $g/(cm^2)$(day), somewhat higher than that of a reference borosilicate waste glass.

Actual fuel leaching studies have been under way for several years and have been accelerated with the Administration's decision to defer spent fuel reprocessing and the consideration of disposing of spent fuel in a geologic repository. Much of this work in the United States has been done at the Pacific Northwest Laboratory and is described in reports from that installation.[4,5]

Leach tests of fuel fragments having a nominal dimension of about 0.6 cm taken from actual power reactor fuels with burnups ranging from 9 to 54 MWd/kgU were carried out for periods up to about 3 years at 25 C with de-ionized water, sodium chloride solution, calcium chloride solution, sodium bicarbonate solution, and WIPP "B" brine (a reference solution developed by workers at the Waste Isolation Pilot Plant). Comparisons between spent fuel and borosilicate waste glass were made. In sodium bicarbonate solution the leach rates were nearly equal; the glass was more resistant in the calcium chloride solution, and the difference rose progressively in sodium chloride solution, WIPP "B" brine, and de-ionized water in which the waste glass was two to three orders of magnitude more leach resistant than the spent fuel. Fuel burnup showed no discernible effect on leach rate. Table III-B-1 shows the comparison between glass and spent fuel based on release of Pu-239 and Cm-244.

In summary, it appears that spent fuel can be acceptable as a waste form in anticipated repository environments even though its long-term stability and leach resistance under certain conditions is not as great as that of borosilicate glass or other potential waste forms.

b) <u>Alternate Solid Waste Forms</u> Research and development on immobilization of HLW has been carried out since the 1950s. As part of the Waste Solidification Engineering Prototype (WSEP) program a variety of solidified waste forms were investigated. These included calcine, phosphate glass, phosphate ceramics, and borosilicate glass.[6] Earlier, work was done at Brookhaven National Laboratory on fixation of radionuclides in montmorrilonite clay and, on a small scale, at The Johns Hopkins University on the incorporation of radionuclides in synthetic feldspars (alumino-silicates).[7] As a result of these early investigations, and taking into account the process development aspects of waste solidification, borosilicate glass was selected as the most promising form for further development.[8] Although the program has emphasized

Table III-B-1

Comparison of Leach Rates at 25 C

Leach Solutions	^{239}Pu Leach Rates (a) g/(cm)^2(day) 76-68 Glass	Spent Fuel	^{244}Cm Leach Rates (b) g/(cm)^2(day) 76-68 Glass	Spent Fuel
Deionized Water	5×10^{-8}	2×10^{-5}	4×10^{-7}	2×10^{-5}
WIPP "B" Brine	2×10^{-8}	2×10^{-6}	1×10^{-8}	3×10^{-6}
NaCl	7×10^{-8}	3×10^{-6}	2×10^{-7}	4×10^{-6}
$CaCl_2$	2×10^{-8}	3×10^{-6}	1×10^{-7}	9×10^{-8}
$NaHCO_3$	2×10^{-7}	1×10^{-7}	2×10^{-7}	2×10^{-7}

(a) 151 d

(b) 454 d

From PNL SA 7734, p 19

the development of borosilicate glass, a substantial part of the effort has been directed to other waste form alternatives.[9] These include development of a crystalline waste form called "supercalcine", multibarrier or coated waste forms, glass-ceramics, ceramic cermets, and synthetic rocks (Synroc) incorporating the natural minerals hollandite, zirconalite, and carnallite.[10]

Extensive investigation of leaching of various waste forms has been, and is being, carried out.[11] Leaching tests have included variations of groundwater composition and pH, temperature, pressure, and waste form surface-area-to-volume ratios.

Canada has field tested the storage of nuclear wastes in glass.[12] In 1960 at Chalk River, 25 glass samples, each about 5 inches in diameter, containing 1100 curies of fission products, were placed in the ground below the water table and completely, without any barriers, exposed to water flowing at 7 inches/day. The loadings of Sr-90 in the Canadian experiments were kept low (5×10^{-3} Ci/g) to eliminate temperature as a significant variable in their experiment.[13] Total fission product loadings of 1 to 20 Ci/g are usually considered for commercial HLW, and for 10-year-old waste Sr-90 represents about 20 percent of the fission product activity. Measurements of Sr-90 concentration in the water have been carried out for 18 years and have shown that the Sr-90 concentration 3 feet downstream is less than the maximum permissible concentration for unrestricted areas in the federal regulations (10 CFR 20).

Although at the scientific level reasonable arguments have been advanced, supported by preliminary laboratory data, that waste forms superior to borosilicate glass are possible, independent authoritative conclusions have been reached with respect to the positive suitability of glass as a waste form. For example, the Panel of Waste Solidification of the National Academy of Sciences Committee on Radioactive Waste Management concluded in its report[14] that "The Panel finds that many solid forms are likely to be satisfactory for use in an appropriately designed system. Furthermore, at least one form--glass--because of an extensive development effort, is currently adequate for use in a first demonstration system consisting of solidification, transportation, and disposal."

Also a Peer Review Panel composed of eight scientists and engineers representing independent, non-DOE laboratories from government, industry, and universities and the disciplines of materials science, ceramics, glass, metallurgy, and geology reviewed the relative merits and potential of eleven alternative waste forms under consideration. In terms of present scientific merit, current least risk use, and present and potential engineering practicality glass was ranked highest. Top research priority was given to multibarrier forms, Synroc, and stuffed glass. The report[15] of the panel, among other things, "concluded that the scientific methodology exists for predicting the long term (>1000-year) performance of glass waste forms and confirming the prediction by use of natural analog materials covering the same time regime as long as the surface temperature of storage is maintained at less than 95 C. Control of waste loading factors, decay prior to incorporation in the solid, or forced cooled interim storage can be used to maintain the < 95 C surface temperature. The rapid radioactive decay of Cs and Sr isotopes during the first 300-500 years provides assurance that a <95 C surface temperature can be maintained over geologic time regimes if the short term temperature is controlled. Maintenance of surface temperatures at <95 C also provides assurance that devitrification of the glass during geologic storage times can be avoided. Extrapolation of hydration rates of natural glass analogs at temperatures < 95 C show a predictable kinetics behavior over times out to 10^6 year. The mechanisms of diffusion controlled surface reactions of glass waste form show equivalent kinetics which provides confidence in extrapolating long term performance."

Processes for the solidification of high-level waste have been demonstrated both in the United States and abroad. Several canisters of vitrified HLW have been produced using a pilot plant at the Battelle Pacific Northwest Laboratories. In France, a commercial vitrification facility has been in operation at Marcoule since 1978. A summary of foreign solidification programs is presented in Table III-B-2.

c) Primary Canister Materials In a multibarrier system, there is usually a canister to contain the spent fuel assembly or solidified HLW. The purpose of the canister is to facilitate handling and emplacement operations and to provide extra protection during the initial period after disposal. One benefit of the canister can be the delay of contact between water, if present, and the solid waste form until the high-level waste form has undergone some years of

Table III-B-2

Highlights of International Activities in HLW Solidification
March 1979 - March 1980

Country	Highlights
Australia	Professor A E Ringwood's Synroc development program is being supported by the IAEA ($50 K during the first year for equipment) and the AAEC (funding for a staff of three). The AAEC is to cooperate in the program with scale-up of fabrication techniques and irradiation testing of the product.
Belgium	The waste vitrification situation at Mol is as follows. Eurochemic is still responsible for solidifying the high-level wastes generated through 1974 by the Eurochemic fuel reprocessing plant. There are two types of waste: 67 m^3 of Low Enriched Waste Concentrate (LEWC), typical of Purex wastes from reprocessing LWR fuels; and about 800 m^3 of High Enriched Waste Concentrate (HEWC), high-aluminum wastes from reprocessing MTR fuels. Two waste vitrification plants are to be built on the Eurochemic site at Mol. An AVM-type plant, to vitrify the 800 m^3 of HEWC is to be built by the French for the Belgian Government (Belgoprocess) with the limitation that no information or know-how is to be turned over to the other Eurochemic partners. The cost of the plant is to be funded through normal contributions to Eurochemic from the member states. Operation of the AVB at Mol is to be limited to Belgian personnel. A Pamela process pilot plant is to be engineered and paid for by the Federal Republic of Germany. Eighty percent of the funds are to be provided by the Ministry of Research and Development and twenty percent by DWK. The Pamela process is being redesigned to produce a borosilicate glass and either glass beads or monolithic blocks, using existing Eurochemic LEWC. Startup of this plant is not expected until 1986.

Table III B-2 (cont)

Country	Highlights
Belgium (cont)	Present plans are that two vitrification plants at Mol will be totally independent of each other, with no shared utilities, services or buildings.
Federal Republic of Germany	See Belgium
United Kingdom	Harwell researchers have successfully tested the use of microwave power in a continuous HLW vitrification process. The technique also uses a continuously replaced glass-former off-gas filter. Harwell is also starting design work on a joule-heated ceramic melter with plans to install one.

From: PNL-3333

radioactive decay and is, therefore, at a lower temperature; this, in turn, improves the leach resistance of the solid waste form. The best developed of the canister designs uses metals, such as steel, copper, and titanium.

Rockwell Hanford Operations has conducted studies of passive, near-surface, interim storage concepts and has recently shifted emphasis to investigations of waste forms and packages for retrievable storage or permanent disposal in geologic repositories. During this past year, major activities within the program (reconstructed, expanded, and presently designated Commercial Waste and Spent Fuel Packaging (CWSFP) Program) have included several experimental investigations involving simulated and actual spent fuel, as well as completion of the conceptual design of a Spent Fuel Receiving and Packaging Facility (SFRPF).[16]

Unmodified spent fuel assemblies, packaged in relatively simple steel containers along with inert gas or some solid medium for improved thermal and/or mechanical properties, have indicated potential as an acceptable waste form if stored in a suitable repository. The simplicity of processing this type of package configuration is clearly advantageous from the standpoints of cost and safety. As a result the primary thrust of analytical and experimental spent fuel activities in the CWSFP Program thus far has been based upon "simple" packages comprising steel cylinders containing one or more fuel assemblies and backfilled with helium prior to final closure. Stabiizer materials other than helium (metals, ceramics, cements, graphite, sand, crushed rock) have also been considered.[17-22]

The principal spent fuel activities conducted within the SWSFP Program are:

- Simulated near-surface (drywell) storage demonstrations at Hanford and the Nevada Test Site (NTS)
- Surface (sealed storage cask--SSC) and drywell storage demonstrations at the NTS
- Spent fuel receiving and packaging facility conceptual design

These investigations are described in the following paragraphs.

The SSC and drywell storage concepts were selected for experimental verification on the basis of studies conducted previously. The canisters designed and fabricated for the Engine Maintenance and Disassembly (EMAD) demonstrations are not prototypical. Packaging and emplacement of 3 PWR fuel assemblies, one into an SSC and two into drywells, were successfully completed during January 1979.

The fourth unit at EMAD is being tested in a hot cell to evaluate effects of external temperature on fuel pin temperatures. Electric heaters surround the assembly to provide control of the boundary temperature and permit measurement of interior pin temperatures as a function of the surrounding thermal environment. The data obtained in this effort will support analyses of close spacing in a repository or burial in a particularly insulative medium.

For spent unreprocessed nuclear fuel, KBS of Sweden has proposed to use a canister of pure metallic copper, so-called Oxygen-Free High-Conductivity Copper.[23] The fuel in the canister is imbedded in lead in order to prevent deformation of the canister by external overpressure. The thickness of the copper reduces the radiolysis of groundwater.

Before encapsulation the spent fuel assemblies are dismounted and the fuel rods are placed in special copper racks. Each rack can take 496 BWR-rods or 636 PWR-rods. The rack is then placed in a copper container. The empty space between the fuel rods is backfilled with lead in an oven at 380-400 C temperature. After cooling, the copper canister is sealed with three consecutive lids. The lids are electron beam welded. Welding tests have been made with good results. An ultrasonic test of the welds has been experimentally checked and does not involve any particular problems. The welds will also be helium-leak tested.

As the copper is thermodynamically stable in pure water, corrosion can only be sustained by chemical substances which are dissolved in the groundwater. Investigations have shown that the only substances of importance are free oxygen and sulfides.

Chemical analysis shows that the groundwater is practically free from oxygen at some hundred meters depth. This is also evident from the presence of iron(II) (Fe^{2+}) in the rock minerals. It has been conservatively assumed that the groundwater will contain 0.1 ppm of free oxygen. Another oxygen source is the air enclosed in the buffer material when the repository is sealed. A third source is the oxygen formed by radiolysis. Using upper limit values it has been calculated that these free oxygen sources can corrode about 4.5 kg of copper in 1 million years.

For calculating the corrosion from sulfides it has been conservatively assumed that the groundwater contains 5 mg sulfides per liter. This is the highest value observed in any measurement and is much higher than can be expected in the chemical environment close to a repository. Sulfides could also be formed by microbiological reduction of sulfates. However, the sulfate-reducing bacteria require organic material for their activity. It has been conservatively assumed that all organic material available, ie, less than 12.5 mg per liter groundwater and less than 200 mg per kg buffer material, will be suitable for such bacteria. With these assumptions the maximum corrosion attack from sulfides would be about 55 kg copper per canister in 1 million years.

Thus in total the copper corrosion would be at most 60 kg per canister after 1 million years. This is equivalent to 0.5 mm average corrosion depth on the 200 mm thick canister. If it is assumed that the oxidants coming from the tunnel attack only the upper 10 percent of the canister surface, the average corrosion depth there would be about 2.4 mm in 1 million years.

A study of the KBS work by a subcommittee of the National Academy of Sciences recently concluded[24] "In summary, the Subcommittee agrees that the KBS authors have established a technically sound basis for concluding that the copper canisters would be both mechanically stable and highly resistant to corrosion for times on the order of one million years, even if conditions in the repository would be changed by remotely possible rock movement or by the incursion of seawater."

As an alternative to the copper canister, KBS has supported development work at the ASEA Company's High Pressure Laboratory on a ceramic canister.[25] Alumina of the α-type such as the minerals called corundum or sapphire has many properties which makes it very attractive as material for high-level waste canisters.

Next to diamond, corundum and sapphire are among the hardest minerals occurring in nature, and deposits of these in the form of weathered materials in river beds and shingle on seashores exhibit a very high mechanical and chemical resistance over long periods of time.

The method used for producing the canister, hot isostatic pressing, is one of several high-pressure methods which can be accomplished with ASEA's QUINTUS presses. A container and a lid are produced separately by pressing alumina powder at 100 MPa and 1350 C. The alumina container is then examined by non-destructive testing methods. Ultrasonic examination and proof testing are methods which can be used.

Before the encapsulation the spent fuel assemblies are dismounted, upper and lower tie plates are removed, and the fuel rods are withdrawn from the spacer grids. The length of the fuel rods is reduced by a specially developed method. Today's equipment does not allow fabrication of canisters larger than about 3 m length, whereas ordinary LWR rods are close to 4 m in length.

The rods are rolled to spiral rolls and stacked in a steel container which is then enclosed in the alumina canister. The proposed canister is 3 m long and has 0.5 m outer diameter and 10 cm wall thickness. The weight would be about 2 metric tons. Each canister can take 144 BWR rods or 174 PWR rods.

During the initial stage of the project it was considered unsuitable from the geological viewpoint to encapsulate full-length fuel rods in alumina canisters. It was surmised that rock movements might give rise to unacceptable stresses in such a long (approximately 5 m) alumina canister. Through increased knowledge of earth movements and a suitable arrangement of the storage procedure, the need to reduce the length of the fuel rods has decreased in importance. Nevertheless, it is considered preferable to continue the development work on an encapsulation process which permits the containment of spent fuel rods in alumina canisters about 3 m long. This means that QUINTUS presses of the necessary size are available today. However, there is nothing in the process to prevent the production of alumina canisters of a size sufficient for the encapsulation of full-length fuel rods.

In the final repository, the canister will be subjected to the action of the surrounding groundwater. However, this takes place at a very slow rate. Tests are currently under way to establish the durability of the hot-isostatically compacted aluminum oxide in the types of groundwater in question. These tests are being conducted at elevated temperatures, 100-350 C, in order to obtain measurable effects. In the groundwater in question, however, deposition on the surface of the canister is more likely than erosion of the surface.

Tests performed to date have shown that a canister of aluminium oxide with 100-mm thick walls can withstand the action of the groundwater for hundreds of thousands of years with an ample safety margin. The durability of the canister is only slightly affected by the surrounding geological environment.

Both theoretical and experimental studies concerning the strength and thermal properties of large ceramic bodies have also been carried out. The results show that the canisters can resist the forces to which they can be subjected in the final repository with adequate margin.

During the first months of the KBS project glass ceramic materials were considered as strong potential candidates for waste canisters.[26] Preliminary studies were started on glass-ceramic code 9617 at Corning Glass Works in Corning, NY. This material was chosen because it was known from other applications to be fairly tolerant of different chemical environments. Further, it seemed to be reasonably simple to develop a fabrication technique for large canisters. The approach would be to fabricate a container and a cover of the base material and then seal the final joint by a frit that would be ceramed at lower temperature than the base material.

The preliminary studies were mainly concentrated on corrosion studies. The code 9617 glass-ceramic is a composite of several phases, beta spodumene (Li_2O . Al_2O_3 . X SiO_2; X = 5-7; 90 percent) rutile (TiO_2; $\sim$3 percent) spinel (Mg or Zn - Al_2O_4; $\sim$ 3 percent) and residual glass ($\sim$5 percent). Because of this, one would expect the composite material to exhibit different localized corrosion rates. This appears to be the case. Scanning electron microscopy of the glass-ceramic surface eroded in water showed the glass phase to be removed leaving the crystalline phases relatively untouched. The preliminary data obtained indicated that the glass-ceramic would probably meet the requirements on a high-level waste canister. It was, however, concluded that this could not be verified within the relatively short time schedule of the KBS project. These types of materials should, however, be considered as an economically attractive alternative for long-time resistant waste canisters. Their development and verification would, however, require some years of work.

All of the above canister materials, such as copper, aluminium, and glass-ceramics would also be used for vitrified high-level waste.

In the Swedish design for disposal of high-level waste in granite, one canister system, which uses a layer of lead 4-inches thick surrounded by 1/4 inch of titanium,[27] has been devised to provide at least 500 years of retrievability; the use of other metals, such as copper, and various ceramics is also being considered as part of the US waste disposal system.

In the case of bedded salt, the experience from research in systems to handle geothermal brines indicates that titanium alloys might be a suitable choice for a canister material.[28] Research at Sandia Laboratories to date tends to confirm this initial conclusion.[29]

d) Overpack In addition, another canister (or "overpack") could be placed around the primary canister, if it were deemed necessary for overall system performance. Many of the canister materials discussed above could be used, in various combinations, as overpacks.

A number of waste form and package options are now or will soon be available for the disposal of spent fuel or solidified HLW. These include a variety of metallic and nonmetallic canister/overpack materials and both glass and ceramic forms for the solidification of high-level waste. Sufficient study has now been accomplished to allow at least a provisional conclusion that, as a waste form itself or as embedded in a metallic matrix, spent fuel will be adequately leach resistant to meet waste package performance requirements. Nevertheless, as in the case of other system elements, some alleged gaps and uncertainties have been raised with respect to waste form and canisters. These are considered below.

2) Alleged "Gaps and Uncertainties"

Concerns about the durability of the waste form and canister/overpack have most frequently been raised in conjunction with solidified HLW forms (particularly glass) and have centered about the following few areas:

- Corrosion of the solid form and canister/overpack at higher operating temperatures
- The effect of the repository environment including the pH of the water, gases (eg, oxygen) and minerals such as sulphites dissolved in the water, or the action of brines in the core of salt disposal
- The effects of radiation, particularly the effects of radiolysis
- The validity of extrapolating laboratory accelerated-life testing to long future time periods

a) The Effect of Higher Operating Temperatures There is ample evidence from both laboratory experiments and nature that many materials are less durable at high temperatures. The debate on the use of glass as a solid waste form has centered primarily on this issue.

Although a recent study by the National Academy of Sciences/National Research Council concluded that glass is an adequate solid waste form,[30] some scientists who have criticized the selection of glass for fixing nuclear wastes base their concerns on devitrification data and leach data at temperatures of 200 to 300 C. These high temperatures will only be reached with the relatively high fission product concentrations (about 25 percent) being considered as an option in the United States. By reducing the fission product content before emplacement in the repository, the maximum temperature at the surface of the glass in the proposed Swedish waste storage system is 65 C.[31] A reduction of the US waste from 25 percent to about 5 percent will result in the same maximum temperature of 65 C at the surface with less than 10 years' decay.

Examples of the effect of temperature in glasses can be found in nature. For perlite, one type of natural glass, the rate of devitrification (the rate the glass crystallizes) is 5000 times higher at 200 C than at 100 C.[32] Of course, devitrification doesn't really destroy the value of the glass solid waste form as a barrier. A devitrified glass usually has a leach resistance only about 10 times worse than if it had remained a glass.[33]

Tests conducted in France to measure the leach resistance of borosilicate glass show that the leach rate at 100 C is 3.5 times higher than the leach rate at 70 C and 35 times higher than the leach rate measured at room temperature--about 25 C.[34] The low temperature of the Canadian nepheline-syenite glass, which is due to the very low loadings of activity (about 5×10^{-3} Ci/g of Sr-90), contributes significantly to the low leach rates observed in the Canadian tests.[35]

A number of strategies are available to control the effects of temperature in the waste disposal system including:

1) longer interim storage (eg, 50 years) of the spent fuel or solidified high-level waste prior to disposal,
2) lower loadings, in curies/gram or curies/liter, of radionuclides to low watts/canister, and
3) wider spacing between canisters in the repository.

These may be employed to fully control repository temperature distributions.

b) The Effect of the Repository Environment It is essential that the tests being performed in the laboratory match as closely as possible the probable conditions in the repository. For example, leach rates are often several orders of magnitude lower in groundwater than in distilled water. Shifts to a highly acidic or highly alkaline environment can adversely effect the chemical durability of the solid waste form. It has been pointed out that sulfides and dissolved oxygen in the water would accelerate the corrosion of the KBS-designed copper canister.[36] Concern has also been expressed about the possible corrosive effects of brines which may be found in a salt repository.[37] There are several approaches to mitigating the effects of the repository environment. First, it is wise to be conservative in the design of the canister to allow for some of these accelerated corrosion mechanisms during the life of the canister. Also, it is possible to match the materials to the environment; for example, both nickel and titanium alloys have shown an excellent resistance to corrosion even in geothermal (hot) brines.[38] The control of temperature, discussed earlier, can help reduce the effect of an adverse chemical environment. The choice of an appropriate overpack can also be of assistance.

In the case of brine migration, the technological concern is not with the phenomenon, but with avoiding its potentially deleterious consequences.[39] The phenomenon of brine migration has been one of the most thoroughly investigated concerns relating to the safe disposal of radioactive wastes in bedded or dome salt formations.

In most evaporite formations, small inclusions of liquid are distributed throughout the evaporite rock. In salt beds these inclusions can provide as much as 1 percent of the volume of the medium as brine.[40] (The volume percentage of brine inclusions in domed salt is likely to be considerably smaller than this.) The introduction of a heat source in the salt, as with the emplacement of heat generating waste canisters in a salt repository, can induce a thermal gradient along which these inclusions are known to migrate.[41] Under some conditions the brine can migrate toward the heat source. This possibility presents several concerns, including the accumulation of liquid brine in the emplacement hole and the generation of gases in the repository as a result of radiolysis of the brine.[42] In situ tests,[43,44] laboratory experiments,[45,46] and analyses show that the amount of brine that is predicted to enter a canister emplacement hole is quite small. For example, model calculations calibrated with laboratory experiments show that for repository concepts presently contemplated, for even the worst conditions, less than 10 liters of brine might enter a HLW canister hole in the first 100 years after waste emplacement. For spent fuel repository designs less than 2 liters of brine will enter a canister hole in the first 100 years after canister emplacement.[47] Further, in both the HLW and spent fuel cases, the brine inflow rate is even lower after 100 years. It is anticipated that virtually all of the water entering the HLW or spent fuel emplacement hole will vaporize and leave the emplacement hole because of the expected temperatures. Even where the emplacement hole is sealed the vapor will not produce pressures exceeding 100 atmospheres, well below the lithostatic pressure of the surrounding rock.[48] Even under these extreme conditions, with proper waste package design, canister corrosion rates are predicted to be insufficient to degrade canisters within several hundred years after emplacement.[49]

Thus, the only remaining task is to design and test a waste package that will provide protection against the rather small quantities of brine possibly expected to enter an emplacement hole. This task is relatively simple--dessicant between the canister and the edge of the emplacement hole will very likely suffice to mitigate corrosive effects of migrated brine.

On the basis of the foregoing information the UNWMG-EEI concludes that:

1) brine migration is a phenomenon that has been sufficiently studied and understood to show that its perturbation of a given in situ environment of a repository will be negligible,
2) technical knowledge is readily available to completely mitigate any deleterious consequences of brine migration, and
3) brine migration is a phenomenon which is sufficiently unimportant to merit dismissal in consideration of confidence in mined geologic disposal.

c) <u>The Effect of Radiation</u> Radiation from the spent fuel or solidified HLW can either cause radiation damage to the materials themselves or, by radiolysis (eg, radiation splitting water into hydrogen and oxygen), adversely affect the near-field environment. The solution to this problem adopted by the Swedes has been to use lead shielding.[50] It should be kept in mind that the production of a high radiation field is a relatively short lived phenomena. After a few years, even a plain steel container is sufficient to shield the near-field environment from radiolysis. As pointed out in previous sections, all solid waste forms and candidate canister/overpack materials are extensively tested for susceptibility to radiation damage. In the case of glass, the equivalent of 250,000 years of radiation damage has been found to have negligible effects.[51] In the case of spent fuel as a waste form, the total integrated radiation it will be subjected to during the storage period will be small compared to that to which it was subjected during its use as a fuel.

d) Extrapolation of Short-term Tests to Long-term Phenomena This question centers on the validity of accelerated life testing, for example, by doing tests at high temperatures. Accelerated life testing is a technique which is used in many disciplines on a routine basis. It is true that the degree of extrapolation required is large. However, careful planning and execution of the experimentation coupled with a conservative approach to design allows for these uncertainties. Also, there are extensive Federal programs now underway to reduce these uncertainties,[52,53] and widespread material testing programs are being conducted at the various national laboratories as well as by private contractors.

3) Conclusions

A large number of options are now or will soon be available for the disposal of spent fuel or solidified high-level waste. These include a variety of metallic and nonmetallic canister/overpack materials and both glass and ceramic forms for the solidification of HLW. The next appropriate step beyond laboratory studies will be in situ tests on a meaningful scale. These studies should include both spent fuel and solidified high-level waste. Sufficient study has been accomplished to allow the provisional conclusion that as a waste form itself or as embedded in a metallic matrix, spent fuel will be adequately leach resistant to meet waste package performance requirements. A multiplicity of possible stabilizers are available and, depending on the nature of the waste form and of the remainder of the waste package, can be selected to maximize waste package performance. In summary, there is presently adequate information to conclude with confidence that the waste form and package will perform as required to contribute to the assurance of containment.

REFERENCES FOR SECTION III-B

1. Office of Nuclear Waste Isolation, Battelle Memorial Institute, Topical Report: The Disposal of Spent Nuclear Fuel (ONWI-59, December 1979)

2. M.D. Houston, Properties of Spent Fuel, Proceedings of the Conference on High-Level Radioactive Solid Waste Forms, Denver, Colorado (NUREG/CP-0005, December 1978).

3. H.D. Holland and L.H. Brush, Uranium Oxides in Ores and Spent Fuels, Proceedings of the Conference on High-Level Radioactive Solid Waste Forms, Denver, Colorado (NUREG/CP-0005, December 1978).

4. Y.B. Katayama, Spent LWR Fuel Leach Tests, Battelle Pacific Northwest Laboratory (PNL-2982, April 1979).

5. Y.B. Katayama and D.J. Bradley, Long-Term Leaching of Irradiated Spent Fuel, Battelle Pacific Northwest Laboratory (PNL-SA-7734, 1979).

6. J.L. McElroy et al, Waste Solidification Program Summary Report, Evaluation of WSFP High-Level Waste Solidification Processes (BNWL-1677 July 1972).

7. The Johns Hopkins University, Fixation of Radioactive Material in Stable, Solid Media (TID-7550, June 1957).

8. W.F. Bonner, The High-Level Waste Immobilization Program: An Overview, Battelle Pacific Northwest Laboratory (PNL-3094, September 1979).

9. Proceedings of the Conference on High-Level Radioactive Solid Waste Forms, Denver, Colorado (NUREG/CP-0005, December 1978).

10. J. Tewey, Distribution of Radioactive Waste Elements in Synroc Minerals, Waste Management '80, Tucson, Arizona (March 1980).

11. Op cit, Bonner (Ref. 8).

12. W.F. Merritt, High-Level Waste Glass: Field Leach Test, Nuclear Technology (January 1977).

13. D.J. Cameron and G.G. Strathder, "Materials Aspects of Nuclear Waste Management in Canada, Ceramics in Nuclear Waste Management, Cincinnati, Ohio (CONF-790420, April 1979).

14. National Academy of Engineering/National Academy of Sciences, Solidification of High-Level Radioactive Wastes (NUREG/CR-0895, July 1979).

REFERENCES FOR SECTION III-B (cont)

15. L.L. Hench, et al, The Evaluation of Review of Alternative Waste Forms for Immobilization of High-Level Radioactive Wastes, Alternative Waste Form Peer Review Panel (August 1979).

16. M.S. Bensky, et al, Spent Fuel Storage and Isolation, Waste Management '79, Tucson, Arizona (February 1979).

17. J.W. Braithwaite and M.A. Molecke, Nuclear Waste Canister Corrosion Studies Pertinent to Geologic Isolation, Sandia National Laboratories (SAND 79-1935J, 1979).

18. R.E. Westerman et al, Preliminary Conceptual Designs for Advanced Packages for the Geologic Disposal of Spent Fuel, Battelle Pacific Northwest Laboratory (PNL-2990, April 1979).

19. N.J. Magnani and J.W. Braithewaite, Corrosion Resistant Metallic Canisters: An Important Contributer to the Multibarrier Concept for Nuclear Waste Disposal, Sandia National Laboratories (SAND79-1429C, 1979).

20. E.J. Nowak, The Backfill as an Engineered Barrier for Nuclear Waste Managment, Sandia National Laboratories (SAND 79-0990C, 1979).

21. I. Neretnieks, Retardation of Escaping Nuclides from a Final Repository, KBS Tochnisk Rapport 30 (Stockholm, Sweden, 1977).

22. W.J. Mecham, Method for Storage of Solid Waste, U.S. Patent Office (U.S. Patent No. 3,983,050; (1976).

23. P. Ahlstrom, Ceramic and Pure-Metal Canisters in Buffer Material, Proceedings of the Conference on High-Level Radioactive Solid Waste Forms, Denver, Colorado (NURFG/CP-0005, December 1978).

24. Subcommittee for Review of the KBS II Plan, Committee of Radioactive Waste Management, National Academy of Sciences, A Review of the Swedish KBS II Plan for Disposal of Spent Nuclear Fuel (1980).

25. Op cit, Ahlstrom (Ref. 23).

26. Id.

27. Karn-Bransle-Sakerhet, Handling of Spent Nuclear Fuel and Final Storage of Vitrified High-Level Reprocessing Waste, Vol. IV (Stockholm, Sweden, November 1977).

28. J.P. Carter and S.D. Cramer, "Corrosion Resistance of Some Commercially Available Metals and Alloys to Geological Brines," Corrosion Problems in Energy Conversion and Generation, The Electrochemical Society (1974).

29. J.W. Braithwaite and M.A. Molecke, High-Level Waste Canister Corrosion Studies Pertinent to Geologic Isolation, Presented at the Conference on High-Level Radioactive Solid Waste Forms, Denver, Colorado (December 1978).

REFERENCES FOR SECTION III-B (cont)

30. National Academy of Sciences/ National Research Council, Solidification of High-Level Wastes, Report of the Panel on Waste Solidification (December 1978).

31. Op cit, KBS (Ref. 27).

32. R.R. Marshall, Devitrification of Natural Glass, Geol. Soc. America Bull (Vol. 72, October 1961).

33. J.E. Mandel, The Storage and Disposal of Radioactive Waste as Glass Canisters, Battelle Pacific Northwest Laboratory (PNL-2764, December 1978).

34. Op cit, NAS (Ref. 30).

35. Op cit, Merritt (Ref. 12).

36. Op cit, Ahlstrom (Ref. 23).

37. G.J. McCarthy et al, "Hydrothermal Reactivity of Simulated Nuclear Waste Forms and Water Catalysed Waste-Rock Interactions," Scientific Basis for Nuclear Waste Management, Vol 1, Plenum Press, New York, N.Y (1979) p. 329.

38. Op cit, Carter (Ref. 28).

39. J.D. Bredehoeft, et al, Geologic Disposal of High-Level Radioactive Wastes - Earth Sciences Perspectives (U.S. Geological Survey Circular No. 779, 1978).

40. G.H. Jenks, Effects of Temperature, Temperature Gradients, Stress and Irradiation on Migration of Brine Inclusions in a Salt Repository, Oak Ridge National Laboratory (ORNL-5526, July 1979).

41. W.B. Krause, et al, In situ Brine Migration Experiments at the Avery Island Salt Mine, RE/SPEC Inc (to be published in 1980).

42 Op cit, Bredehoeft (Ref. 39).

43. Op cit, Krause (Ref. 41).

44. R.L. Bradshaw and W.C. McClain, Project Salt Vault - A Demonstration of the Disposal of High-Activity Solidified Wastes in Underground Salt Mines, Oak Ridge National Laboratory (ORNL-4555, 1971).

45. Op cit, Jenks (Ref. 40).

46. J.J. Hohlfelder and G.R. Hadley, "Laboratory Studies of Water Transport in Rock Salt," Letters in Heat & Mass Transfer (Vol. 4, 1979), p. 271.

47. H.C. Claiborne, et al, Expected Repository Conditions - Salt, Oak Ridge National Laboratory (ORNL-TM-7201, to be published).

REFERENCES FOR SECTION III-B (cont)

48. G.H. Jenks, Radiolysis & Hydrolysis in Salt-Mine Brine, Oak Ridge National Laboratory (ORNL-TM-3717, March 1972)

49. J.W. Braethwaite and M.A. Malecke, High-Level Waste Canister Corrosion Studies Pertinent to Geologic Isolation, Oak Ridge National Laboratory (ORNL-TM-7201, to be published).

50. Op cit, KBS (Ref. 27).

51. Op cit, Bonner (Ref. 8).

52. Office of Nuclear Waste Isolation, Battelle Pacific Northwest Laboratory, Technical Program Plan (ONWI-19, December 1979).

53. U.S. Department of Energy/U.S. Geological Survey, Earth Science Technical Plan for Disposal of Radioactive Waste in a Mined Repository (Draft DOE/TIC-11033, January 1980).

C ADDITIONAL ENGINEERED BARRIERS

In the previous section we described the waste package as being the system of barriers which includes everything placed in the emplacement hole with the waste by the transporter. Thus the waste package is a combination of waste form, canister (which might include stabilizers within the canister), canister overpack, and fillers. Anything else which might be added to the system, eg, emplacement hole liners or sleeves, coatings, grouts, or backfill materials which modify the chemical and physical environment within the repository, are defined as "additional engineered barriers"[1] and are described in this section. These engineered barriers represent Barrier #4 (see Figure I-8 and Figure III-H-1).

1) Status of Technology

In any repository the mined cavity will be backfilled, and this backfill in itself is an additional engineered barrier. The backfill material may, if deemed necessary, be modified to adjust physical and chemical conditions relative to specific radioisotopes. In addition some thought has been given to special sleeves or cement grouting beyond the canister/overpack or to the need for a concrete vault in the repository. The major engineered barrier under consideration beyond the canister/overpack is the placement of special materials in the space between the canister/overpack and the geologic medium. In addition to retardation of radionuclide migration with an appropriate canister design or inert coating of the waste form, certain materials can be used to absorb or otherwise slow radionuclide migration from the package and the repository. Retardation mechanisms include surface absorption, ion exchange, coprecipitation, and redox effects. Materials could also be selected to minimize the contact between the waste package and any water which might enter the repository.

The sorption material should be mechanically, thermally, and chemically stable in the repository environment. Also, it must be dry when in contact with the waste package and in the waste form radiation field to prevent accelerated canister corrosion. Good heat-conducting properties and low permeability to groundwater also are desirable sorption material characteristics. If the material is used for repository backfilling, it should have sufficient load-bearing capacity to prevent cavern roof collapse. The organic content of the filling material should be low to avoid radionuclide complexing and possibly enhanced migration rates. Materials may be added, if necessary, to cause oxidation-reduction changes that retard radionuclide migration.

Candidate sleeve materials include many of the same materials which may be useful for canister/overpacks (see Section III-B). Candidate material selection will be based largely on the results of corrosion tests as a function of temperature, radiation, and groundwater chemistry (eg, pH, eH, composition, and ionic strength), that are typical of the water in various media of interest, ie, basalt, granite, salt, and tuff.

Research sponsored by the Office of Nuclear Waste Isolation (ONWI) is determining sorption coefficients of many minerals and rocks that may be of interest for sorption barrier use.[2] Swedish and Canadian workers also have ongoing programs to investigate sorption of radionuclides in clays and rocks. Current candidate materials are listed below along with references to studies and assessments of their applicability:

Metals:[2-7]

- Ti alloys
- Zr alloys
- Ni alloys
- Pb, Pb alloys
- Fe, Fe alloys
- Cu, Cu alloys
- Aluminum alloys

Ceramics:[2,8-11]

- Al_2O_3 (alumina)
- $2Al_2O_3 \cdot SiO_2$ (mullite)
- Al_2O_3-ZrO_2-SiO_2 (fused refractory
- $CaTiO_3$ (perovskite)
- $CaO \cdot ZrO_2 \cdot TiO_2$ (zirconolite)
- TiO_2 (rutile)
- ZrO_2 (zirconia)
- $ZrSiO_4$ (zircon)

Carbides:[2,12,13]

- TiC
- SiC
- TaC

Glasses:[2,11,14-20]

- Wide variety

Carbon:[2,12,13]	Cements:[21-26]
Impervious graphite	High-alumina cements
Glassy carbon	Specially tailored cements, grouting
Pyrolytic graphite	Compounds, or chemical binders

Test programs on these candidate materials now include the assessment of general corrosion rates, pitting and crevice corrosion susceptibilities, stress corrosion cracking, effects of oxygen concentration, solution volume to solid surface area ratio, and possible effects from radiolysis products.[3,4,5]

Currently available data on emplacement hole backfill barrier performance,[27-30] particularly in regard to radionuclide sorptive characterisitcs, indicate that backfill materials can effectively contribute to the isolation of radioactive wastes in a geologic repository in the presence of brines and other groundwaters. The following list shows the backfill candidate materials that have received attention in the last several years through studies in the United States and Sweden:[27,28,31-35]

Sand	Attapulgite
Bentonite	Peat
Borates	Other clay minerals
Zeolites	Gypsum
Iron	Al_2O_3
CaO	Carbon
MgO	$CaCl_2$
Tachyhydrite	Crushed host rock
Anhydrite	Mixtures of the above
Apatite	

Backfill materials are being tested for selective nuclide sorption capacities (for fission products and actinides) to eliminate or significantly reduce radionuclide migration through the backfill barriers. The ability to prevent or delay groundwater flow through the backfill is also being determined. Other properties of interest being evaluated[27,28,33] are thermal conductivity, mechanical support strength, swelling, plastic flow, and forms and methods for emplacements.

Recent studies have focused on the testing and development of smectite clay and sand barriers in the presence of several brine compositions for utilization in a salt repository.[36] The actinides Pu and Am sorb very well

(K_d values about 2000), and Eu, Cs, and Sr sorb moderately (K_d values about 200). Existing data suggest that a properly chosen, 1-foot-thick backfill barrier surrounding a waste container could delay the breakthrough of Pu and Am (defined as 1 percent of original concentration) through the barrier for periods of 10,000-100,000 years, depending upon the interstitial brine flow rate. Concurrently, the breakthrough of Cs, Sr, and Eu could be delayed for 1000-10,000 years, which is sufficient time for nearly complete decay of those radionuclides. A wetted clay/sand mix also swells appreciably, yielding a nearly impermeable groundwater barrier. Backfill studies in Sweden[37,38] on bentonite clay, clay/sand mixtures, and barriers of the zeolite mineral clinoptilolite yield similar results: a 0.2 m backfill barrier of clinoptilolite could delay the breakthrough of Cs and Sr in groundwater for about 10,000 years; a 1-m-thick clay/sand mixture could delay the release of Pu and Np for about 2,000,000 years.

2) Alleged "Gaps and Uncertainties"

The primary uncertainty about engineered barrier systems is the long-term behavior of candidate materials in the repository environment. This will depend on the geological medium, repository design, and waste package design. In situ testing will be required when these parameters are fixed. However, the available in situ data,[39,40,41] along with the results of other studies[42,43-47] and conclusions reached in studies of individual barrier components, indicate that there is a sufficient understanding of the interactions of the proposed storage media and waste package material.

3) International Activities and Experience

The Swedish studies[48,49] on engineered barriers represent the major international activity in this area. These studies concluded that for spent fuel disposal, a thick copper canister in conjuction with a bentonite clay could prevent the release of radionuclides to the host rock in the presence of granitic groundwater for thousands to hundreds of thousands of years. In this system the bentonite barrier chemically conditions the groundwater, reducing its corrosiveness on the copper canister. As discussed in Section III-B the Swedes have also done extensive studies[50,51] on other long-lived package concepts, such as thick aluminum oxide canisters and titanium-clad lead canisters, for spent fuel as well as high-level waste. All proved to have an effective lifetime in granite of several thousand years.

4) Conclusions

If additional engineered barriers are deemed necessary, numerous materials are available to accomplish specific physical or chemical functions to assure that overall repository performance requirements are met.

REFERENCES FOR SECTION III-C

1. Lester, et al, System Study on Engineered Barriers, Task 1, Interim Report, Science Applications, Inc. for Battelle Pacific Northwest Laboratory (December 1979).

2. R.E. Westerman, et al, Preliminary Conceptual Designs for Advanced Packages for the Geologic Disposal of Spent Fuel, Battelle Pacific Northwest Laboratory (PNL-299, April 1979).

3. J.W. Braithwaite and M.S. Molecke, Nuclear Waste Canister Corrosion Studies Pertinent to Geologic Isolation, Sandia National Laboratories (SAND 79-1935J, 1979).

4. N.J. Magnani and J.S. Braithwaite, Corrosion Resistant Metallic Canisters: An Important Contributer to the Multibarrier Concept for Nuclear Waste Disposal, Sandia National Laboratories (SAND 79-1429C, 1979).

5. J.W. Braithwaite, et al, Titanium Alloy Corrosion in Nuclear Waste Environments, Sandia National Laboratories (SAND 79-2023E, 1979).

6. D.W. Shannon, Corrosion of Iron-Base Alloys Versus Alternate Materials in Geothermal Brines, Battelle Pacific Northwest Laboratory (PNL-2456, November 1977).

7. D.W. Shannon, Geochemical Engineering Program - Progress Report for the Period Ending December 1977, Battelle Pacific Northwest Laboratory (PNL-2736, September 1978).

8. Kircher and Bowman, eds., Effect of Radiation on Materials Components, Rheinhold Publ. Co., New York, N.Y. (1964), p. 42.

9. Office of Nuclear Waste Isolation, Battelle Memorial Institute, Waste Package Design, Development, and Test Plan (ONWI Draft 1979).

10. E. Mattsson, "Corrosion Resistance of Canisters for Final Disposal of Spent Nuclear Fuel," Scientific Basis for Nuclear Waste Management, Vol. 1, Plenum Press, New York, N.Y. (1979), p. 271.

11. J.D. Sundquist, "Preliminary Corrosion Test of a Glass-Ceramic Candidate for a Nuclear Waste Canister," Scientific Basis for Nuclear Waste Management, Vol. 1, Plenum Press, New York, N.Y. (1979), p. 289.

12. Op cit, Kircher (Ref. 3), p. 330.

13. J.M. Rusin, et al, "Development of Multibarrier Nuclear Waste Forms," Scientific Basis for Nuclear Waste Management, Vol. 1, Plenum Press, New York, N.Y. (1979), p. 169.

REFERENCES FOR SECTION III-C (CONT.)

14. Op cit, Kircher (Ref. 8), p. 44.

15. F.P. Robert, et al, Radiation Effects in Solidified High-Level Wastes, Part I: Stored Energy, Battelle Pacific Northwest Laboratory (BNWL-1944, January 1976).

16. R.P. Turcotte, Radiation Effects in Solidified High-Level Wastes, Part II: Helium Behavior, Battelle Pacific Northwest Laboratory (BNWL-2051, May 1976).

17. J.E. Mendel, et al, Annual Report on the Characterization of High-Level Waste Glasses, Battelle Pacific Northwest Laboratory (BNWL-2252, June 1977).

18. W.A.Ross, et al, Annual Report on the Characterization of High-Level Waste Glasses, Battelle Pacific Northwest Laboratory (BNWL-2625, June 1978).

19. W.J. Weber, et al, "Radiation Effects in Vitreous and Devitrified Simulated Waste Glass," in Proceedings of the International Symposium on Ceramics in Nuclear Waste Management in Cincinnati, Ohio, April 30 - May 2, 1979 (P.O. Chikalla and J.E. Mandel, eds. 1979).

20. P. Ahlstrom, Ceramic and Pure Metal Canisters in Buffer Material, Proceedings of the Conference on High-Level Radioactive Solid Waste Forms, Denver, Colorado (NUREG/CP-0005 December 1978).

21. E.L. White, "Permeability Measurements on Cementitious Materials for Nuclear Waste Isolation," Scientific Basis for Nuclear Waste Management, Vol. 1, Plenum Press, New York, N.Y. (1979), p. 471.

22. J.A. Stone, "Studies of Concrete as a Host for Savannah River Plant Radioactive Waste, Scientific Basis for Nuclear Waste Management, Vol. 1, Plenum Press, New York, N.Y. (1979), p. 443.

23. G. Rudolph and R. Koster, "Immobilization of Strontium and Cesium in Intermediate-Level Liquid Wastes by Solidification in Cements," Scientific Basis for Nuclear Waste Management, Vol. 1, Plenum Press, New York, N.Y. (1979), p. 467.

24. R.O. Lokken, A Review of Radioactive Waste Immobilization in Concrete, Battelle Pacific Northwest Laboratory (PNL-2654, June 1978).

25. J.G. Moore, et al, Radioactive Waste Fixation in FUETAP (Formed Under Elevated Temperature and Pressures) Concretes - Experimental Program and Initial Results, Oak Ridge National Laboratory (ORNL/TM-6375, 1979).

26. D.M. Roy and G.R. Gouda, High-Level Radioactive Waste Incorporation into Special Cements, Nuclear Technology (Vol. 40, 1978), p. 214.

27. E.J. Nowak, The Backfill as an Engineered Barrier for Nuclear Waste Management, Sandia National Laboratories (SAND 79-099C, 1979).

REFERENCES FOR SECTION III-C (CONT.)

28. I. Neretnieks, Retardation of Escaping Nuclides from a Final Repository, KBS Tochnisk Rapport 30 (Stockholm, Sweden, 1977).

29. E.J. Nowak, The Backfill Barrier as a Component in a Multibarrier Nuclear Waste Isolation System, Sandia National Laboratories (Draft SAND 79-1109, 1980).

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

31. Karn-Bransle-Sakerhet, Handling and Final Storage of Unreprocessed Spent Nuclear Fuel, Vol. II, Technical Chapter 5 (Stockholm, Sweden, 1978).

32. Op cit, Ahlstrom (Ref. 20).

33. Sandia National Laboratories, Summary of Research and Development Activities in Support of Waste Acceptance Criteria for the Waste Isolation Pilot Plant (SAND 79-1305, 1979).

34. S. Fried and A. Friedman, "Kinetic Effects in Migration," in Radioactive Waste in Geologic Storage, American Chemical Society Symposium Series 100 (1979).

35. Op cit, Nowak (Ref. 30).

36. E.J. Nowak, The Backfill as an Engineered Barrier for Nuclear Waste Management, Sandia National Laboratories (SAND 79-0990C, 1979).

37. P. Ahlstrom, Ceramic and Pure-Metal Canisters in Buffer Material, Proceedings of the Conference on High-Level Radioactive Solid Waste Forms, Denver, Colorado (December 1978).

. Karn-Bransle-Sakerhet, Handling of Spent Nuclear Fuel and Final Storage of Vitrified High-Level Reprocessing Waste, Vol. IV (Stockholm, Sweden, November 1977).

39. R. Bradshaw and W. McClain, Project Salt Vault: A Demonstration of the Disposal of High-Activity Solidified Wastes in Underground Salt Mines, Oak Ridge National Laboratory (ORNL-4555, April 1971).

40. Krumhansl, Final Report: Conasanga Near-Surface Heater Experiment, Sandia National Laboratories (SAND 79-1855, November 1979).

41. G.H. Jenks, Effects of Temperature, Temperature Gradients, Stress, and Irradiation on Migration of Brine Inclusions in a Salt Repository, Oak Ridge National Laboratory (ORNL-5526, July 1979).

42. Op cit, Magnani (Ref. 4).

REFERENCES FOR SECTION III-C (CONT.)

43. Office of Nuclear Waste Isolation, Battelle Memorial Institute, Quarterly Report (ONWI-9(4), 1979).

44. M.A. Molecke, PNL-Sandia Joint High-Level Waste Total System Test, Sandia National Laboratories (Memorandum to distribution, January 1980).

45. G.J. McCarthy, et al, "Hydrothermal Reactivity of Simulated Nuclear Waste Forms and Water Catalysed Waste-Rock Interactions," Scientific Basis for Nuclear Waste Management, Vol. 1, Plenum Press, New York, N.Y. (1979), p. 329.

46. G.J. McCarthy, et al, "Hydrothermal Stability of Spent Fuel and High-Level Waste Ceramics in the Geological Repository Environment," in Proceedings of the International Symposium on Ceramics in Nuclear Waste Management (T.D. Chikalla and J.E. Mendel, eds., April 1979).

47. W. White, paper presented at Materials Research Society Meeting, Boston, Massachusetts (November 1979).

48. Op cit, KBS (Ref. 31).

49. Karn-Bransle-Sakerhet, Handling of Spent Fuel and Final Storage of Vitrified High-Level Reprocessing Waste, Vol. III: Facilities (Stockholm, Sweden, 1978).

50. Op cit, KBS (Ref. 31).

51. Op cit, KBS (Ref. 49).

D REPOSITORY DESIGN AND CONSTRUCTION

The proper design and construction of a geologic repository are required to assure that all the barriers associated with the disposal system perform their necessary functions.

1) Status of Technology

Conceptual designs for geologic repositories for spent fuel and HLW have been completed.[1,2,3] A typical commercial-scale repository would consist of several chambers excavated deep within a suitable geologic formation, along with access shafts and necessary surface structures. A large repository might ultimately occupy around 2000 acres of subterranean area . Surface facilities at the repository, which would occupy approximately 200 to 300 acres of land, would accommodate the receipt and unloading of the waste shipping containers, the transfer of the waste containers to inspection facilities, and the preparation of the waste for emplacement in the drilled cavities in the repository.

Initially, the repository would probably be operated so that the waste, in the form of packaged spent fuel, would be readily retrievable; that is, the design would allow for all emplaced waste to be removed from the repository at about the same rate and with about the same effort as emplacement. After satisfactory repository operation for a suitable period of time, operations performed to maintain this retrievability would be terminated. Once the repository was considered full, surface facilities would be decommissioned and dismantled, all storage and access shafts to the mine would be sealed, and the area marked for the future.

During the operacion of Project Salt Vault[4,5] (described in Section II) the following tests were performed and information gathered:

- Fourteen irradiated fuel elements were emplaced and removed to demonstrate equipment and techniques

- o Electrical heaters were used to raise the temperature of large quantities of salt to determine in situ structural capabilities
- o Four million curies of fission products in 21 containers were emplaced and removed safely
- o Geohydrologic and geophysical factors relating specifically to natural containment of wastes were investigated and information developed
- o Thermal effects were evaluated and maximum heat loadings (150 kW/acre) were established setting canister spacing and number
- o Rock mechanics analysis and testing for design, construction, and operation of a repository were completed
- o Mechanisms to cause radionuclide movement were identified and determined to be essentially negligible
- o Radiation impacts (energy storage and release in salt were found to be self healing at temperatures above 150 C)
- o Brine content was found to be small in volume and of minor consequence even though corrosive
- o Environmental assessment bases and methods were developed, and a safety analysis performed

A more recent preliminary design for bedded salt is the Waste Isolation Pilot Plant (WIPP) planned for southeastern New Mexico.[6] The WIPP reference repository has been designed to receive, inspect, overpack when necessary, and permanently dispose of radioactive wastes in bedded salt. It could be a repository for defense TRU waste, demonstration of the disposal of spent power reactor fuel, and an experimental facility for in situ tests of proposed techniques for the disposal of spent fuel and HLW. Since the advanced status of the WIPP design demonstrat current capability, a brief description is provided in Appendix III-D-1.

From an engineering design and construction standpoint a geologic repository for spent fuel or HLW can be characterized as an underground civil structure conceptually similar to other mined cavities which, for centuries, have been constructed and operated over extended periods of time. Even though some of these underground workings were not built for long-term stability, they have remained open to the present. With a repository, however, there is spent fuel or waste producing thermal loading of the rock formations and radiation introducing thermal, mechanical, and possibly chemical effects that add another dimension to the repository design. This dimension must be considered in connection with the operational period of the repository and for some extended period following repository closure. This extended period of time, identified as the "thermal" period, is related to the major heat pulse associated with the decay of the fission products.

Thus the major kinds of information needed for the design of a repository are:

- Physical and mechanical properties of the host rock
- Temperature profiles and thermal stresses
- Induced stresses due to excavation

These information requirements are discussed below.

a) Physical and Mechanical Properties of the Host Rock In order to properly design and construct a repository in a specific rock formation it is obviously necessary that appropriate information on the properties of the rock and their variation in the volume affected by the repository be obtained. We are addressing here the site-specific geotechnical information required to provide the bases for repository design at a specific site. Current exploration and investigative techniques utilizing geophysical techniques, drill holes, and down-hole logging by a variety of radiometric, electric, and electronic methods, as noted elsewhere in this report, are deemed capable of locating potentially suitable sites. These methods and techniques are also useful in determining to a considerable extent the degree of uniformity of the formation under consideration.

A wide variety of techniques and a vast amount of data and information are available regarding the determination of rock properties pertinent to repository design and construction. The types of such information and data are shown by the investigative efforts carried out in connection with WIPP.[7] The three major areas of investigation of the thermophysical properties of rock salt at the WIPP site comprise:

1) petrography related to physical and mechanical properties,
2) general physical properties (density, moisture content, etc), and
3) thermal-mechanical properties.

Included in the first are petrographic analyses to identify rock failure mechanisms. Physical property measurements for core samples include density, moisture content, porosity, permeability (gas and brine), electrical resistivity, ultrasonic velocity, and thermal conductivity. Included in the measured mechanical properties are uniaxial compressive strength, tensile strength, stress-strain behavior under triaxial compression, elastic modulus, yield stress, and creep rates. The effect of temperature and pressure on these properties is an integral part of the continuing investigative program. From the comprehensive description and discussion of this investigative effort at WIPP it can be concluded that rock properties required for repository design and construction can adequately be determined.[8]

b) Temperature Profiles and Thermal Stresses Central to any consideration of any thermal effect is the thermal source term. This is so whether the effects be in the very near-field (canister dimension scale), near-field (repository room dimension scale), or far-field (beyond 1-2 times overall repository dimension scale and up to the earth's surface). The question is whether the thermal source term is large enough to adversely affect rock mechanics, thermo-chemical effects, or hydrologic effects. It is important to recognize that the thermal source term, in the case of spent fuel, is a function of the age of the spent fuel at time of emplacement and the spacing of the canisters or areal thermal loading density in the repository medium. With solidified high-level waste the thermal source term is also dependent on the type of waste and its concentration in the solid waste form. All of these factors are specifically quantifiable and, more importantly, subject to direct control as a basis for repository design.[9] Thermal decay over time for spent fuel and HLW are shown in figure III-D-1.

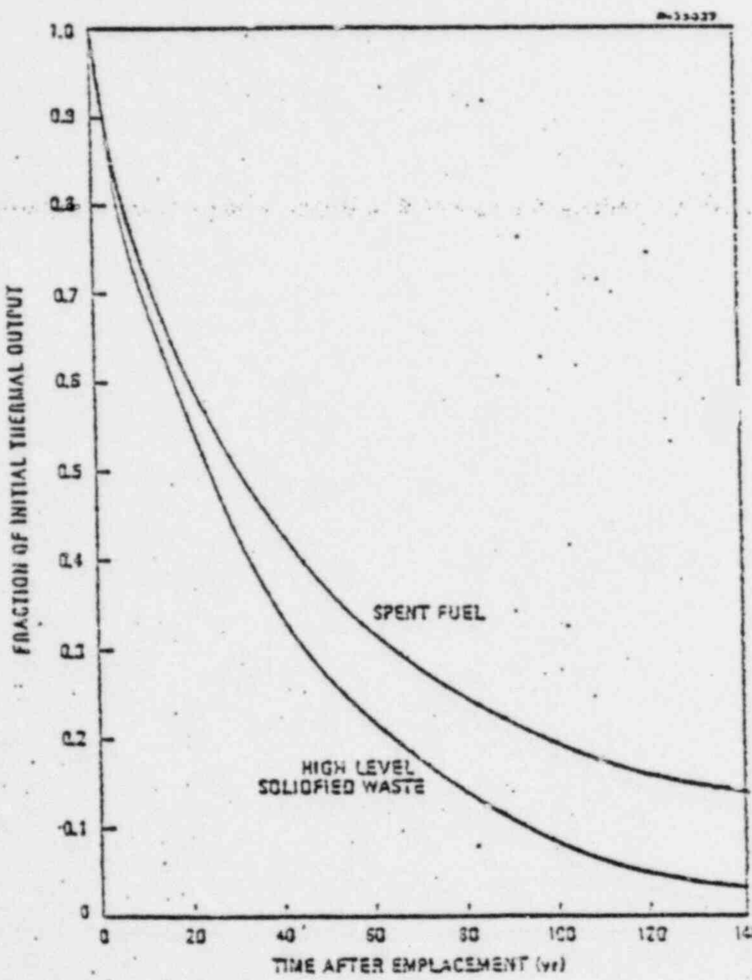
It is also pointed out that the amount of temperature increase and its distribution in the repository can be modified by specific repository operations, eg, mechanical ventilation. This latter point is illustrated in figures III-D-2 and III-D-3. In these illustrations, repository (in salt) conditions are:

thermal conductivity 2.2 $Btu/hr\text{-}ft^2\text{-}F$

density 135 lb/ft^3

Figure III-D-1

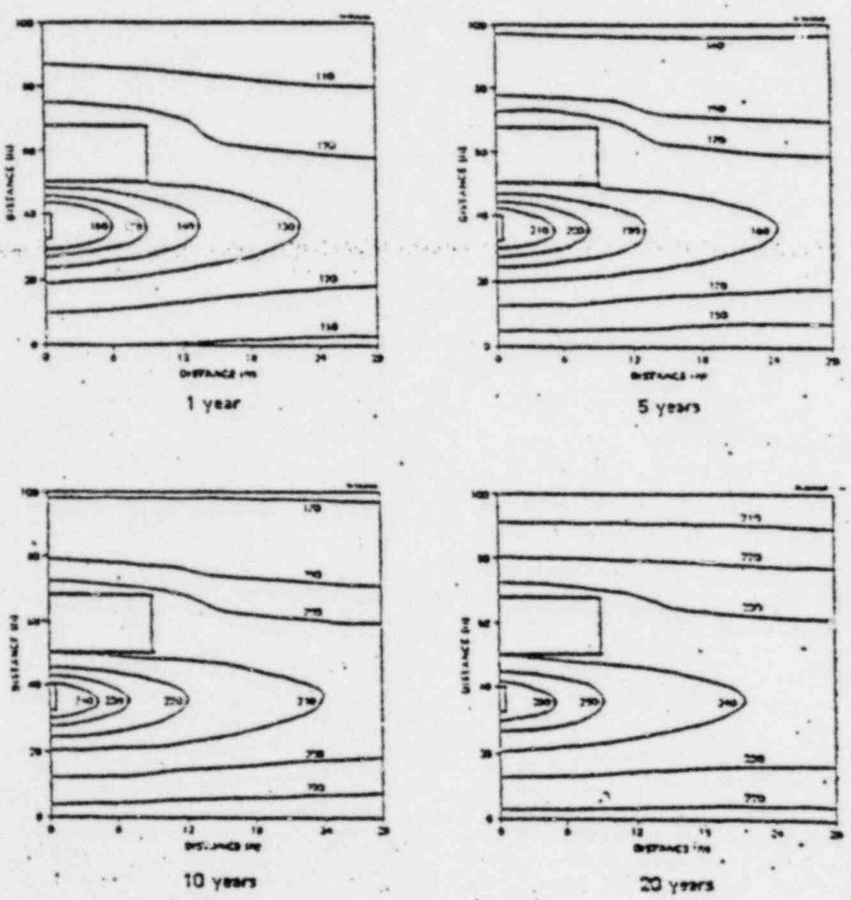
Thermal Decay Curves for High-level Solidified Waste and Spent Fuel



Reference: NUREG/CR-0495

Figure III-D-2

Thermal Contours for the Unventilated Repository



Pillar Width = 40 ft

Conductivity of Salt = 2.2 Btu/ft-hr-°F

Note: The Horizontal and Vertical Scales are Unequal

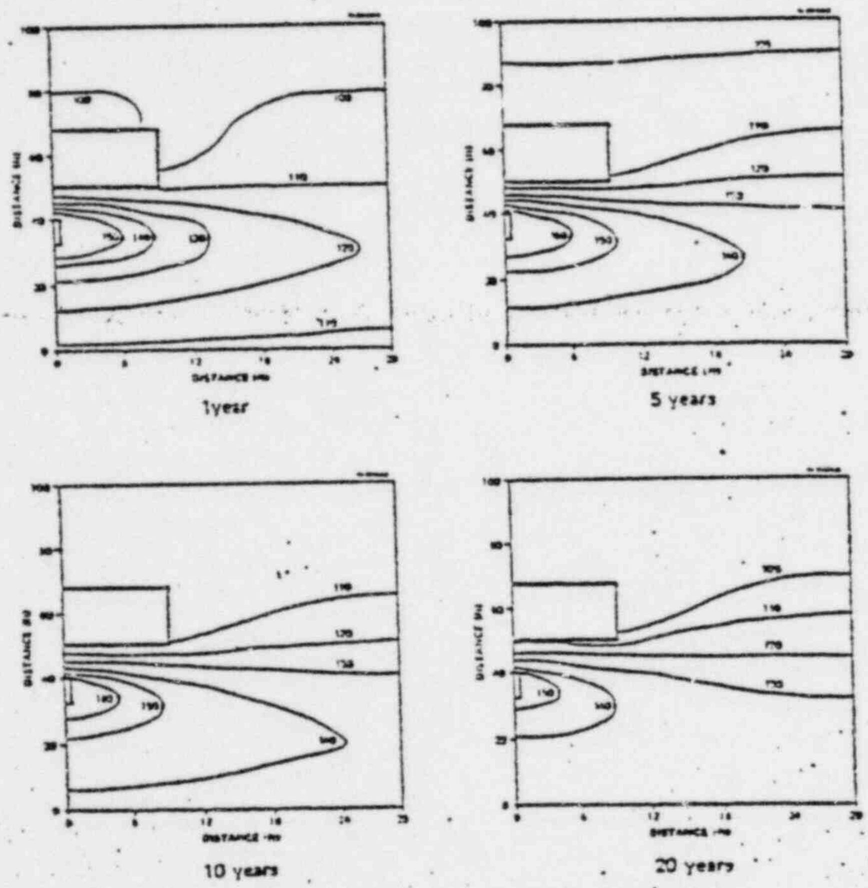
Unventilated

Temperatures in °F

Reference: NUREG/CR-0495

Figure III-D-3

Thermal Contours for a Ventilated Repository



Pillar Width = 40 ft

Conductivity of Salt = 2.2 Btu/ft-hr-°F

Note: The Horizontal and Vertical Scales are Unequal

Ventilated: Inlet Air Temperature = 70 °F

Temperatures in °F

Reference: NUREG/CR-0495

specific heat capacity 0.21 Btu/lb-F

initial temperature 100 F

HLW canister initial thermal output 3.5 kW

bottom of canister 18 ft below room floor

canister 10 ft x 1 ft (8 ft for HLW)

storage rooms 560 ft x 18 ft x 18 ft

areal thermal loading 100 kW/acre

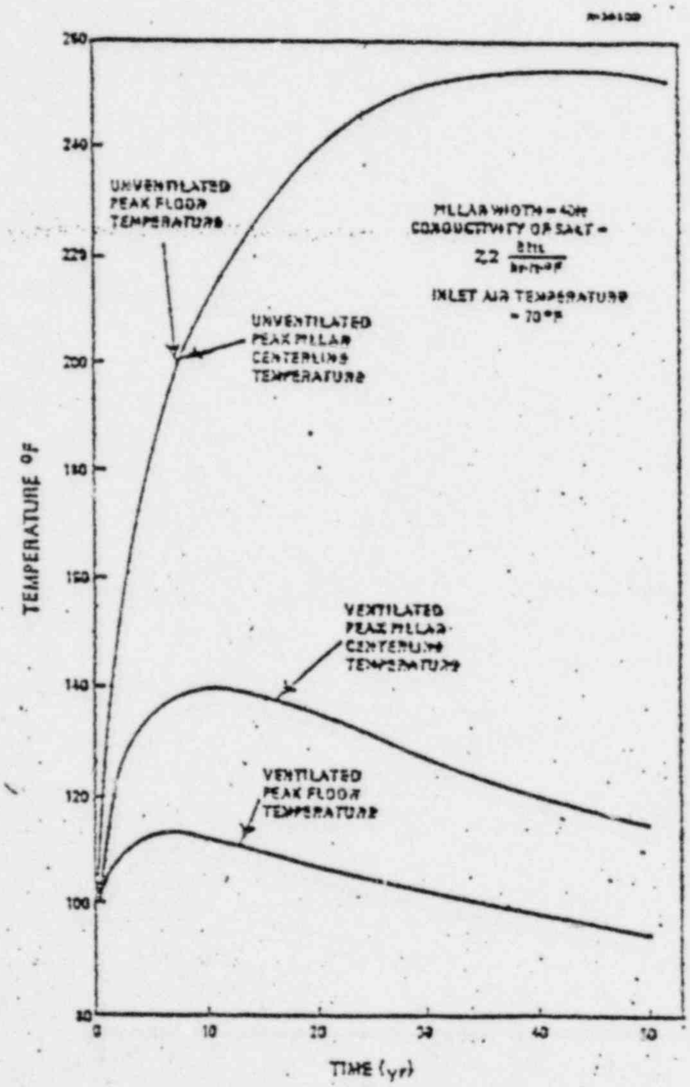
ventilation case 9700 ft^3/min, 70 F inlet

The variation in peak temperature with time is shown for these same repository conditions in figure III-D-4.

An example of the conservatism associated with a very near-field design consideration is that related to salt decrepitation, or fracture temperature of salt due to the differential thermal expansion of brine inclusions that may be found in the salt and the salt crystals themselves. McLain and Bradshaw[10] noted that the temperature at which decrepitation was detected in some 48 samples of bedded salt from seven different locations averaged about 280 C with none being less than about 250 C. Several samples showed no signs of decrepitation at the maximum test temperature of 400 C. None of a number of samples of domal salt decrepitated. Cheverton and Turner[11] showed that with thermal loading of 150 kW/acre and 5.1 kW/canister no more than 1 percent of the salt volume in a unit cell surrounding the canister would exceed 250 C and no more than 25 percent of the salt volume would exceed 200 C. However, as has been noted by Llewellyn[12] and others the thermal source strength of a 10-year old HLW canister is only about 2.1 kW (a spent fuel canister of the same age would have a thermal source strength of less than 1 kW). Very near-field design temperature constraints are therefore not limiting. (As noted in Section III-B, considerations of waste form integrity could dictate lower temperatures.) It is emphasized again that a) this particular factor is of a site-specific nature that is subject to specific determination as a basis for specific repository design requirements, and b) the thermal strength of the source is subject to direct control. Similar statements can be made with respect to thermal effects in the near-field, eg, structural stability of room openings and pillars and relation to extraction ratios, and to far-field considerations such as up-lift and the potential effect on the integrity of overlying stratigraphy and surface or aquifer temperature increases.

Figure III-D-4

Temporal Variation of Repository Temperature



Reference: NUREG/CR-0495

The basic objective of the repository design process is a stable underground structure (excavation) that fulfills the requirements for emplacement of spent fuel or solidified HLW. Stability must be assured for the period of waste replacement as well as for thermal loading and natural forces during the thermal period and natural forces beyond that period. The design process consists essentially of optimizing, on a conservative basis, the excavation size, shape, and orientation with respect to the in situ stress field.[13] Analytical and numerical (eg, finite element and finite difference methods) solution methods exist, and are being further developed, for the design problems involved.[14] A number of rock thermomechanical computer codes have been developed, and although none have been completely verified, results so far have been good. For example, the thermophysical behavior of rock salt has been modeled by various workers.[15] Of specific interest and pertinence is the analytic reproduction of actual Project Salt Vault field test results by two independent groups and analytic methods.[16,17]

c) Stresses Induced by Excavation The two major concerns about excavating are the development of fractures on the perimeter of the excavated area as a result of drilling or blasting and the subsidence of the overlying rock when rooms or tunnels are dug out.

These questions obviously relate to the nature of the formation under investigation. In the case of salt, excavation can be done by mechanical ripping with a continuous mining machine which should have much less impact than drilling or blasting. In the case of hard rock, it may be possible to use tunnel-boring machines. If blasting and drilling are necessary, fracturing can be limited by controlling configuration of drill holes, size and type of charge, and sequence of detonation.[18] In situ tests are in progress to confirm the suitability of such controls to waste repositories. These questions have been addressed in detail, and it was concluded that the excavation process can be satisfactorily accomplished using existing mining techniques.[19]

The nature of the induced stress depends on the geometry of the excavation and the extraction ratio. This ratio for a repository will be about 10-20 percent, compared to up to 90 percent for a conventional mine. This low ratio will minimize this problem. The stability of storage rooms can be further improved by mining the roof of the excavation to a shape that minimizes the effect of geologic structure and by installing artificial supports.[20]

Another factor that should be considered in the design of a repository is the ability to retrieve the waste if necessary. It should be emphasized that the retrieval of wastes is considered to be an unlikely event.

The NWTS program has structured a repository safety verification program consisting of modeling, analyses, laboratory testing, field testing, and in situ testing to gain confidence in the ability of the repository (engineered and natural features) to safely isolate waste from the biosphere.

Much of this program is to be conducted prior to the emplacement of waste at a repository site and will corroborate a high level of confidence in repository performance prior to waste emplacement. In addition to this extensive program to assure safe performance, the NWTS program has gone even further to assure program safety by establishing the design concept that the waste will be retrievable for an appropriate period of time.

Retrieval of spent fuel in stainless steel canisters emplaced in salt was demonstrated on a small scale in Project Salt Vault.[21] Retrieval is, in essence, the reverse of the emplacement process; the degree of difficulty of retrieval depends mainly on the degree to which the emplacement process has been carried through. Retrieval would be easiest when the canisters were first emplaced, relatively loose in their holes, and with ventilated rooms which were not backfilled. The most difficult conditions of retrieval would be encountered just prior to decommissioning the facility, when canisters may be tight in their holes and rooms and corridors may be backfilled and unventilated. The feasibility of removing backfill from the rooms,[22] ventilating rooms to attain operational temperatures,[23] and retrieval or recovery of the waste[24,25] have been established through engineering studies.

UNWMG-EEI believes that the length of the retrievability period should be based on whether sufficient additional confidence in repository safety is gained from gathering in situ data to warrant the delay in establishing the designed long-term configuration (backfilled, sealed, etc) and the continuing additional expense of maintaining retrievability.

2) Alleged "Gaps and Uncertainties"

The principal alleged gaps and uncertainties that have been raised concerning information required for repository design and construction are:

- Questions regarding the representativeness of rock mass involved in the repository structure. A corollary question concerns the representativeness of the rock properties as determined from samples with respect to the in situ properties of the rock mass.
- The adequacy of thermomechanical models for the various potentially suitable geologic repository media under certain thermal impacts has been questioned. In addition, predictions of thermomechanical response of groundwater flow require further analysis and model development.

These areas are being addressed through in situ testing activities[26] and the Earth Sciences Technical Program.[27] Although as noted previously Project Salt Vault field results were generally reproduced by two independent modeling analyses (one using the finite difference and the other the finite element method) further development work is being continued. For example, in a dome salt formation (Avery Island, LA) heated canisters have been emplaced to obtain confirmatory thermomechanical response data and information.[28]

The extent or significance of questions concerning the representativeness of rock samples is obviously dependent on the nature of the formation under investigation. Rock properties are not as important in, for example, dome salt or suitable salt beds as they might be in igneous rocks. Nevertheless, they are most properly addressed by site-specific subsurface investigations and tests.

This requires the systematic collection and evaluation of structural geologic data, which is accomplished by essentially two general approaches or methods. The first is what amounts to a statistical analysis of formation discontinuities from information obtained by geophysical and drill hole methods where direct access to the excavation location is not initially available. The second is a deterministic analysis of discontinuities where actual discontinuities can be examined through access by exploratory shafts or adits.[29] These methods are, of course, an integral part of any engineering undertaking involving geotechnology.

Further, with respect to thermomechanical questions, work is well underway, and answers should be available by the time the site selection process is complete. In any event, the technology is sufficiently developed so that design can proceed in stages, relying on the results of in situ testing after initial excavation of at least a portion of the repository.

In summary, the alleged gaps and uncertainties do not represent an obstacle to repository design and construction.

3) <u>International Activities and Experience</u>

An experimental waste disposal facility has been operating in Germany for the past 13 years. This facility uses an existing abandoned mine in the Asse salt dome with renovated corridors and shafts to permanently store low- and intermediate-level wastes.[30] In 1967, West Germany started disposing of significant numbers of low-level drummed wastes at Asse, and from August 1972 to June 1976, intermediate-level waste drums were also placed in the mine in shielded transport casks. No difficulties with equipment, radiation exposure, or safety have been encountered. West Germany is planning to put AVR (a small high-temperature gas-cooled reactor) elements into the mine; however, the characteristics of these elements are such that essentially only radiation impact will be tested. However, West Germany is also conducting tests with electric heating elements in Asse to obtain design data for the thermal effect on the larger Gorleben design.[31] Testing the stability of underground openings at ambient temperature, including measurements of closure in both old and newly mined rooms, is also in progress. The United States is currently negotiating an agreement with Germany to continue to exchange information on waste management and to initiate joint field testing programs in the Asse facility.

A cooperative US/Swedish experimental program is being conducted at the Stripa mine in Sweden involving in situ testing in a granite formation.[32] This program is the first comprehensive set of in situ tests to evaluate hard crystalline rock as a medium for disposing of radioactive waste. Experiments with emplaced heaters have shown that existing computer codes can accurately calculate temperature profiles in the rock and that changes in stress as a result of heating can also be calculated. Calculation of deformation resulting from heating did not agree with measured values, and this is still under investigation.

Design and safety analyses for repositories have been performed in several countries--Canada, Denmark, Germany, Italy, Sweden, and the United Kingdom.[33-38] All rely on traditional deep-mining techniques to excavate into rock formation.

4) Conclusions

With respect to repository design and construction, a variety of methods are available to measure physical and mechanical properties of rock, and a combination of laboratory and in situ testing can adequately characterize rock properties. Induced stresses due to excavation can be predicted and controlled by proper choice of excavation method, extraction ratio, and room and corridor design. Temperature profiles can be accurately calculated, and a number of models are available for prediction of thermal stresses. Although continued testing of these models is still in progress, results so far, particularly those based on operating experience in Project Salt Vault, indicate that the models are adequate. Furthermore the thermal source term can be controlled by specifying the age of the fuel and canister spacing.

There is therefore no reason why a conservatively designed repository could not be built based on current knowledge. Furthermore, there are no substantial problems associated with the technology for retrievability for whatever period deemed required.

In sum, as stated by the IRG subgroup on Alternate Technology Strategies,[39] "By employing appropriate engineering conservatism and by careful evaluation of existing data and analyses, current knowledge with respect to rock mechanics is adequate to design repositories in salt successfully." Based on the current state of knowledge and in the light of ongoing investigative efforts we conclude that this expression of confidence can also be applied to other formations at appropriate sites.

APPENDIX III-D-1

The WIPP as currently designed consists of both surface and underground facilities, including a waste-handling building for receiving and preparing radioactive waste for transfer underground, an underground personnel building to support underground construction, a storage-exhaust-filtration building, an administration building, four shafts to the underground area, two mined underground levels for the storage of contact-handled (CH) and remotely handled (RH) wastes, and various support structures: a warehouse and workshops, an emergency power plant, a suspect-waste and laundry building, a vehicle-maintenance building, a sewage-treatment plant, and a water supply system. In addition, there would be a mined-rock pile and an evaporation pond for sewage-treatment effluents. A construction spoils disposal area and a sanitary landfill are also included in the design.

The plant would be constructed in accordance with the general design criteria of DOE Manual Appendix 6301, Part 1. Surface buildings that will contain radioactive materials are designed to withstand the effects of credible earthquakes, accidents, and tornadoes to insure that both public health and safety and the environment are protected. The surface structures consist of eight major buildings in an area of about 50 acres. Underground structures consist of four shafts and two waste-storage areas about 2100 and 2700 feet below the surface. Approximately 2000 acres will be used for underground storage.

Surface structures

The surface facilities would support the waste-storage operations. The major surface structure would be the waste-handling building, which is centrally located and equipped to handle both CH and RH waste from the time they are unloaded until they are lowered through the waste shaft for placement underground. In the current design, it is about 230 feet wide, 550 feet long, and 50 feet high (except for a 115-foot-high bay area). The building has separate areas for the receipt, inventory, inspection, and transfer of wastes through separate airlocks to a common waste shaft.

Separate areas are provided for handling CH and RH wastes. The larger portion of the building would be used for CH-waste unloading and loading, inventory, and preparation. A room is provided for overpacking and repairing CH-waste containers. A decontamination area, a cooldown-and-preparation room, and a hot cell are provided for RH wastes. Two independent airlocks would be installed at the shaft entrance for wastes entering from the CH and RH areas. Filtration equipment for the waste-handling area, a laboratory, change rooms, and offices are also located in the waste-handling building. Facilities for CH waste include a rail and truck shipping-and-receiving area, a receiving-and-inspection area, an inventory-and-preparation area, and overpack-and-repair rooms for damaged containers. For RH wastes there is a separate shipping-and-receiving area, an area for shipping-cask preparation and decontamination, a cask-unloading area, and a hot cell for waste-canister storage, overpacking, or decontamination. The waste-handling building also contains offices, change rooms, a health-physics laboratory, and ventilation-and-filtration equipment. Safety equipment and radiation-exposure control measures are included in the design of the waste-handling building.

Other surface structures in this design include the administration building (about 36,000 square feet), the storage-exhaust-filtration building (about 10,000 square feet), the vehicle-maintenance building (about 2300 square feet), a warehouse and shops (about 18,000 square feet), the emergency-power plant (about 10,000 square feet), the sewage-treatment plant, and the suspect-waste and laundry building.

A 30-acre area east of the plant contains the mined-rock pile, which will store the rock, principally salt, excavated from the repository. The maximum height of the pile is 80 feet.

Contact-handled waste would be shipped to the plant in approved shipping containers by rail or truck. It would then be unloaded with an overhead crane in the waste-handling building, through airlocks that control the movement of air during the unloading operations. The air in the waste-handling building would be maintained below atmospheric pressure to prevent contaminants from leaking to the outside air, even though no contaminants are expected to become airborne in significant amounts.

The CH waste would be received in 55-gallon drums, special boxes, or bins that have been transported in shipping containers. Once the shipping containers have been unloaded and the waste removed, the empty containers would be reloaded onto vehicles leaving the plant; the CH-waste containers would be inspected. If found to be acceptable, they would be moved to the CH inventory-and-preparation area and then underground. If a container is found to be externally contaminated or damaged, it would be sent to the overpack-and-repair room, where it can be decontaminated, repacked or recoated, and returned to the CH inventory-and-preparation area for transfer underground.

Remotely handled waste would arrive in special shielded shipping casks, by rail or truck. On arrival, each shipping cask, which may contain one or more canisters of waste, would be inspected and unloaded from the railcar or truck in the cask-unloading-and-receiving area of the waste-handling building. If the railcar or truck is found to be contaminated, it can be cleaned and decontaminated in the transporter wash station outside the building. From there the cask would be moved to the cask-preparation-and-decontamination area, where any special operations such as cask cleaning or attachment of handling equipment can be performed. Remotely handled waste would be handled from behind shielding and/or with remote-handling equipment. The RH-waste canisters will be unloaded from their shipping casks into the hot cell. After appropriate treatment, the shipping cask will be checked for contamination, decontaminated if necessary, and returned to the shipper for reuse. Canisters will be removed from the hot cell and loaded into the facility cask for transfer underground.

The environmental control system would maintain a controlled environment for plant personnel and limit the discharge of radioactivity to the atmosphere. Included in it are heating, ventilating, and air-conditioning systems; air-cleaning and final discharge systems; and all related subsystems.

Plant personnel will work upstream from areas with higher potential for contamination. Access to these areas will be restricted. Pressure differences, maintained between separated areas in the plant and between these areas and the outside air, will insure air flow in the proper direction. To confine radioactive material, the air-cleaning system will pass the air through banks of high-efficiency particulate air filters. Monitors will warn of the presence of radioactivity in the airstream.

Underground facilities

The underground waste facilities in a current design consist of the waste shaft, the waste-shaft hoist-cage system, ventilation shaft, and all facilities in the waste-storage areas. The storage areas in the current WIPP design will be on two levels. The upper level, 2100 feet below the surface, will receive CH waste for storage; it will cover about 170 acres when first developed. The lower level, 2700 feet below the surface, will contain three areas: one (10 acres) for the disposal of RH TRU waste, one (20 acres) for demonstrations with high-level waste, and one (20 acres) for spent fuel disposal demonstrations.

Underground workshops, warehouses, and equipment-storage areas are provided in the design for the various pieces of mining and salt-transport equipment used in construction. An underground ventilation system supplies air to both the construction and the waste-storage areas; separate exhausts are installed for each area. Restrooms and other personnel facilities are also provided. To insure the safety of underground operations, safety equipment and radiation-exposure-control measures are included in the design of the underground facilities.

Both CH and RH wastes would be moved underground through the waste shaft in the waste-handling building. The other accessways to the underground storage areas are the ventilation-supply and service shaft for ventilation and movement of personnel and equipment and a construction-exhaust and salt-handling shaft to remove mined salt and exhaust air from the waste-storage area at each level.

The waste shaft transfers CH and RH waste from the waste-handling building to the underground storage areas. The waste-shaft hoist cage will accommodate the RH-waste facility cask and the CH-waste containers to be handled at the plant. The hoist cage can handle a loaded pallet weighing about 30 tons.

The waste shaft in this design is about 19 feet in diameter and 2700 feet deep. It extends 2100 feet from the surface to the CH-waste level and 600 feet from the CH-waste to the RH-waste level.

The upper end of the waste shaft is in the waste-handling building. After a pallet is loaded, it would be transferred to the hoist cage, which would be lowered through the waste shaft to the underground CH-waste-receiving station. The hoist cage is a fully enclosed steel cage that is guided in its descent and ascent. At the CH-waste-receiving station, an opening, about 20 feet high by

40 feet wide allows access to the shaft. The pallet and the waste containers would be unloaded from the hoist cage onto a diesel-powered transporter for transfer to the CH-waste storage area. A decontamination and radiation-safety check station is located near the waste shaft on the CH-waste level.

The CH-waste storage area as designed consists of four access tunnels and a number of storage rooms. Not all of the tunnels and rooms will have been constructed when the plant starts operating; the layout of the shafts and tunnels would allow mining and storage operations to proceed simultaneously. The first storage rooms would be ready when the plant begins operating and would be used to store waste while the next rooms are being mined. A typical storage room on the CH-waste level is about 45 feet wide, 16 feet high, and 1600 feet long. Rooms are separated by pillars of salt. Contact-handled waste would be stored in bulk except for a small quantity for experimentation. Records would be kept on all container storage locations.

The facility cask, holding RH-waste canisters, would be lowered in the hoist cage to the RH-waste transfer station at the lower end of the waste shaft. Here it would be removed from the hoist cage and put into a holding position or loaded onto a waste transporter for transfer to the RH-waste storage area. Decontamination and radiation-safety check stations would be located close to the shaft.

The RH waste would be stored in a 10-acre array of rooms. The demonstration of spent-fuel disposal will be in an adjacent 20-acre area, and the high-level-waste experiments in a third 20-acre area. Not all the tunnels will have been constrructed when the plant begins receiving RH wastes. The shaft-and-tunnel arrangement can allow underground development and storage operations to proceed simultaneously. A typical RH-waste storage room is about 14 feet wide, 24 feet high, and 500 feet long.

A diesel or electric powered waste transporter would move the facility cask from the shaft to a storage room, where the canister would be transferred directly from the cask to a storage hole below the storage-room floor. Special remote-handling procedures will be used for the emplacement of the canister in the salt. After emplacement, the storage holes will be plugged to floor level. Backfilling with salt will be part of the permanent-disposal procedure. The emplacement procedure for RH waste not intended for permanent disposal will depend on the type of waste or the type of experiment being conducted.

REFERENCES FOR SECTION III-D

1. Office of Waste Isolation, U.S. Department of Energy/Nuclear Division, Union Carbide Corporation, Technical Support for GEIS: Radioactive Waste Isolation in Geologic Formation (Vols. 8 and 10, Y/OWI/TM-36, April 1978).

2. R.A. Langley, Jr., "WIPP--The First U.S. Geological Nuclear Waste Disposal Plant," Transactions of the American Nuclear Society, European Nuclear Conference, Germany (Vol. 31, May 1979).

3. U.S. Atomic Energy Commission, Environmental Impact Statement--Radioactive Waste Repository: Lyons, Kansas (WASH-1503, 1971).

4. R. Bradshaw and W. McClain, Project Salt Vault: A Demonstration of the Disposal of High-Activity Solidified Wastes in Underground Salt Mines, Oak Ridge National Laboratory (ORNL-4555, April 1971).

5. F.M. Empson, et al, Results of the Operation of Project Salt Vault: A Demonstration of Disposal of High-Level Radioactive Solids in Salt, Third Symposium on Salt, Northern Ohio Geological Society (Cleveland, Ohio, 1969).

6. U.S. Department of Energy, Draft Environmental Impact Statement: Waste Isolation Pilot Plant (DOE/EIS-0026-D, April 1979).

7. D.W. Powers, et al, Geologic Characterization Report: WIPP SITE, Sandia National Laboratories (SAND 78-1596, August 1978).

8. Id.

9. J.A. Lieberman, The State of the Technology of Waste Management, Proceedings of the Symposium on Waste Management (Tucson, Arizona, 1979).

10. Op cit, Bradshaw (Ref. 4).

11. R.D. Cheverton and W.D. Turner, Thermal Analysis of the National Radioactive Waste Repository: Progress through March 1972, Oak Ridge National Laboratory (ORNL-4782, September 1972).

12. G.H. Llewellyn, Prediction of Temperature Increases in a Salt Repository Expected from the Storage of Spent Fuel or High-Level Waste, Oak Ridge National Laboratory (ORNL/ENG/TM-7, April 1978).

13. U.S. Nuclear Regulatory Commission, Information Base for Repository Design, Vol. 4: Mine Structural Features (NUREG/CR-0495, March 1979).

14. Id.

REFERENCES FOR SECTION III-D (CONT.)

15. Op cit, Powers (Ref. 7).

16. J.L. Ratigan and G.B. Callahan, Evaluation of the Predictive Capability of the Finite Element Method II: Project Salt Vault Thermal/Viscoelastic Simulation (Y/OWI/Sub-78/22303/II, July 1978).

17. K.K. Wahi, et al, Explicit Finite Difference Simulations of Project Salt Vault, Proceedings of the 19th U.S. Rock Mechanics Symposium (Lake Tahoe, Nevada, May 1978).

18. Op cit, NRC (Ref. 13).

19. U.S. Nuclear Regulatory Commission, Information Base for Repository Design, Vol. 6: Excavation Technology (NUREG/CR-495, March 1979).

20. Op cit, NRC (Ref. 19).

21. Op cit, ORNL (Ref. 4).

22. Kaiser Engineers Rpt. No. 78-56-R, Special Study No. 3: Waste Retrieval from Backfilled Regions (1978).

23. SAI, Thermal Analysis of a Ventilated High-Level Waste Repository (Y/OWI/Sub-76/16527, 1977).

24. SAI, Subsurface Waste-Handling Equipment Criteria and Specifications Development (ONWI/USB/79/E515-00900-3, 1979).

25. Kaiser Engineers, Retrieval Options Study Task 2 Topical Report, Office of Nuclear Waste Isolation, Battelle Memorial Institute (Draft ONWI/SUB/78/E512-02800-6, 1979).

26. P.D. Fairchild and G.H. Jenks, Avery Island Dome Salt In Situ Test (Y/OWI/TM-441, June 1978).

27. U.S. Department of Energy/U.S. Geological Survey, Earth Science Technical Plan for Disposal of Radioactive Waste in a Mined Repository (Draft, January 1980).

28. Op cit, Y/OWI/TM441 (Ref. 26).

29. Op cit, NRC (Ref. 19).

30. C. Salander, The Concept for the Disposal of Spent Fuel Elements from Nuclear Power Plants in the Federal Republic of Germany, International Conference on the Nuclear Fuel Cycle, London, England (September 1978).

31. A. MacLachlan, West Germany Lays Aside Plans for Single Nuclear Fuel Center, The Energy Daily (Vol. 7, No. 188, October 1, 1979).

REFERENCES FOR SECTION III-D (CONT.)

32. P.A. Witherspoon and O. Degerman, Swedish-American Program on Radioactive Waste Storage in Mine Caverns: Program Summary, Lawrence Berkeley National Laboratory (LBL-7019 and SAC-01, May 1978).

33. R.B. Lyon and E.L. Rosingel, Safety Assessment for Deep Underground Disposal-Pathway Analysis, Presented at the International Atomic Energy Agency Symposium on the Underground Disposal of Radioactive Wastes, Helsinki, Finland (IAEA-SM-243/169, July 1979).

34. H.W. Levi, Project 'Safety-Studies Entsorgung' in the Federal Republic of Germany, Presented at the International Atomic Energy Agency Symposium on the Underground Disposal of Radioactive Wastes, Helsinki, Finland (IAEA-SM-243/17, July 1979).

35. F. Firard, et al, A Risk Analysis Methodology for Deep Underground Radioactive Waste Repositories and Related Experimental Research, Presented at the International Atomic Energy Agency Symposium on the Underground Disposal of Radioactive Wastes, Helsinki, Finland (IAEA-SM-243/161, July 1979).

36. G.A.M Webb and M.D. Hill, Application of the Results of Radiological Assessment of High-Level Waste Disposal, Presented at the International Atomic Energy Agency Symposium on the Underground Disposal of Radioactive Wastes, Helsinki, Finland (IAEA-SM-243/25, July 1979).

37. L. Devell, et al, Disposal of High-Level Waste on Spent Fuel in Crystalline Rock: Factors Influencing on Calculated Radiation Doses in the Environment, Presented at the International Atomic Energy Agency Symposium on the Underground Disposal of Radioactive Wastes, Helsinki, Finland (IAEA-SM-243/55, July 1979).

38. On the Safety of Disposal of Radioactive Wastes in Salt Formations at Asse, Institute for Radiation and Environmental Research, Munich, Germany (January 1975).

39. Subgroup Report on Alternative Technology Strategies for the Isolation of Nuclear Waste (TID-28828, October 1978).

E WASTE EMPLACEMENT (ACTIVE OPERATION)

The emplacement of wastes in the repository in a controlled, safe manner is obviously an essential step in the disposal of these materials.

1) Status of Technology

a) Major Operational Activities A description of waste handling operations in an operating facility, as planned for the WIPP facility,[1] was given in Section III-D-1 of this report. The major activities that need to be considered during the period of active operation of the repository are:

- Receipt of spent fuel and interim surface storage--These operations are not different from those routinely conducted at reactor sites and other storage facilities and will not present any unique problems.
- Packaging in canisters and sealing. (See Section III-C)
- Transfer of canisters to underground facility--This is done via a shielded hoist system. The system used in Project Salt Vault[2] consisted of a concrete work platform, biological shield, headframe, hoist, protective enclosure, and a ventilation system to prevent contamination of the mine air in the event of accidental release of radioactivity from a canister. The hoist was also designed to provide emergency hoisting of personnel from the mine in the event of an accident. The system performed well during 19 months of operation. During this time 21 canisters were handled. Canisters were moved from one room to another underground and were brought back to the surface and loaded into shipping casks for return to Idaho.
- Underground transport and emplacement of canisters--A shielded, highly maneuverable underground transporter was designed and built for use in Project Salt Vault.[3] It was built to rigid specifications, including the ability to be easily disassembled into subassemblies small enough to pass through the mine shaft. The transporter was disassembled,

lowered into the mine in sections, and reassembled in an underground room. The transporter was backed into a room at the bottom of the shaft where specially designed equipment centered the canister and positioned it on the transporter. During transfers there were no personnel in the transfer room and all operations were carried out remotely. This system also performed very well during 19 months of operation.

- Closure--When a room has been filled with canisters, it may be backfilled immediately or backfilling may be delayed, possibly until near the end of the operational phase. This would depend on considerations of retrievability and the need for additional data. Instrumentation will be installed with the initial canisters to provide further data on response of the geological system to waste emplacement. These data will be compared with earlier in situ data and the predictions of computational models. Thus, by the time the facility is closed, there will be a very high degree of confidence in predictions of long term performances.

b) Engineered Safety Features Although the handling and disposal of radioactive wastes at a repository pose a lower-level of operational difficulty and potential for accidental release or sabotage than the other facets of the nuclear fuel cycle, a great deal of effort has been expended to incorporate state-of-the-art plant safety concepts into the preconceptual and conceptual repository designs. The extent of the engineered safety features included in current repository designs range from radiological safety to protection against natural phenomena. More specifically, engineered safety features include:

- Confinement systems to promote radiological safety for operational personnel and the public
- Monitoring networks for early warning and detection of potential radiation hazards
- Radwaste management facilities for safe treatment and disposal of on-site generated radioactive waste
- Physical security to guard against unauthorized intrusion and sabotage

* Structural integrity to withstand the forces of natural phenomena and missiles
* Industrial safety features such as fire protection and mine safety

Repository radiation confinement systems are designed to include two or more independent containment barriers for successive control against release of radioactivity to the operations personnel and the outside environment. Confinement systems offer shielding, physical separation, and successively lower atmospheric pressures in areas of greater potential for contamination. All exhaust air will be tested for particulate contamination and, if necessary, filtered through high-efficiency particulate air (HEPA) filters before being exhausted from the facility stack. A system of exclusion zones within waste-handling areas and throughout the entire surface facility will be established to protect on-site workers and the general public from exposure to radiation or radioactive materials.

Facility safety features will include radiological monitoring and control systems for personnel, plant area, stack discharge, and control zone perimeter. Radiation alarm systems would be provided to warn facility personnel of increases in radiation levels in normally accessible spaces and of above normal activity in plant effluents. The system would have redundancy and capability for self-testing its efficiency.

Radioactive wastes will result from the operations at the repository. These wastes, which will be in the form of solids, liquids, and gases, will result from waste-handling operations, decontaminations, filter changes, and suspended surface contamination. All wastes will be treated on-site, packaged (similar to other low-level waste), and probably disposed of in the repository for convenience.

A physical security plan for a repository has been established on the basis of safeguards against either deliberate intrusion and sabotage or accidental intrusion. This plan involves control areas of differing levels of security. One control area will be the surface facilities themselves with the control boundary being the site fence. In typical designs, this area offers round-the-clock security to the repository by employing a 24-hour security force, area surveillance, portal guards, double fences, electronic intrusion detection, and security locks. Outside this area some fencing, sign posting, and periodic guard patrols will maintain security for the repository. Within this area land use and drilling and mining activities will be controlled. The surface area in the control zone that is directly above the underground development will be subject to a more stringent patrol program than the remainder of the area in this zone.

Structural integrity of systems and components related to repository safety will be an important design consideration. The effects of natural phenomena and missiles will be accounted for in the design. Dynamic effects of missiles that might result from equipment failure or from other similar events will also be accounted for so as not to compromise the integrity of systems and components important to safety.

Finally, industrial safety features similar to those employed in all large industrial facilities and underground excavations will be employed at the repository. Federal mine safety codes such as 30CFR57, "Metal and Non-Metallic Underground Mines", and other applicable regulations will govern underground operational safety. A complete fire protection system will be available during all activities underground as well as on the surface.

The concept of engineered safety features at a nuclear waste repository is basically a combination of typical underground facility safety features, typical nuclear facility safety features, and safeguards against intrusion. The design of a "safe" repository from the standpoint of engineered safety features to maintain waste confinement under any circumstance is well within the state-of-the-art, and no serious gaps or uncertainties exist.

Extensive safety analyses have been performed as part of the Draft Environmental Impact Statement for the WIPP facility.[4] Three types of accidents were analyzed--surface fires, underground fires, and hoist failure resulting in canister rupture and dislodging of all the contents with partial crushing. The worst of these accidents was hoist failure. This resulted in dose commitments to lung, bone, and whole body to both the nearest resident and to the population within a 50-mile radius that were many orders of magnitude below natural background. The effects of natural forces such as earthquakes, tornadoes, and thunderstorms were also considered. Since all surface buildings essential for the safe handling of radioactive material are designed to be earthquake and tornado resistant, these phenomena should not result in release of radioactivity, although they may destroy auxiliary structures. Inflow of surface water due to thunderstorms or local flooding is not expected to occur. Siting and design criteria will largely eliminate this possibility, and appropriate protection against groundwater inflow during construction and operation can be provided.

2) Alleged "Gaps and Uncertainties"

Because of the advanced state-of-the-art in spent fuel handling, underground operations, and nuclear plant operations as well as actual experience in physical emplacement of wastes in a mine, no significant scientific or technical gaps or uncertainties exist in this area.

3) International Activities and Experience

The only relevant foreign experience in actual waste emplacement in a mine is the German pilot salt mine facility at Asse.[5] This facility was built in an abandoned mine in the Asse salt dome and has been receiving low- and intermediate-level waste since 1965. About 110,000 drums of low-level waste have been emplaced to date. A larger facility at Gorleben is currently in the planning stages.

4) Conclusions

Technology for handling and emplacement of waste canisters is well developed and presents no new problems. It was concluded in the Project Salt Vault report that "one of the major objectives of the demonstration experiment was to demonstrate the techniques and equipment for handling waste containers in an underground environment. Project Salt Vault experience toward achieving this objective can be considered an unqualified success."[6]

REFERENCES FOR SECTION III-E

1. U.S. Department of Energy, Draft Environmental Impact Statement: Waste Isolation Pilot Plant (DOE/EIS-0026-D, April 1979).

2. R. Bradshaw and W. McClain, Project Salt Vault: A Demonstration of the Disposal of High-Activity Solidified Wastes in Underground Salt Mines, Oak Ridge National Laboratory (ORNL-4555, April 1979).

3. Id.

4. Op cit, DOE (Ref. 1).

5. E. Albrecht, et al, "Disposal of Radioactive Wastes by Storage in a Salt Mine in the Federal Republic of Germany: Management of Low- and Intermediate-Level Radioactive Wastes," International Atomic Energy Agency, Proc. Symp. Aix-en-Provence, 1970, Vienna, Austria (STI/PUB/264, 1970), p. 753.

6. Op cit, Bradshaw (Ref. 2).

F REPOSITORY CLOSURE--BACKFILLING AND PENETRATION SEALING

A deep geologic repository for nuclear waste disposal will be penetrated by boreholes and shafts prior to, and during, construction. The boreholes will be relatively small-diameter holes drilled in the host rock for site exploration and/or geologic sampling. The shafts will be large holes providing access to the repository for mining operations, ventilation, equipment, workers, and nuclear wastes. Since the deep geologic repository concept relies in part on the surrounding rock for containment of radionuclides, any penetration through the surrounding rock represents a potential path for nuclide migration and must be sealed in a manner which will satisfy the overall repository performance objectives. The performance requirements and specific design of penetration seals are obviously dependent on the site-specific characteristics of the host rock and the specific geologic environment in which the seal is to be placed. For example, safety assessment analyses at the WIPP (Los Medanos) site indicate that even under highly conservative conditions of regional fluid flows through the repository via unplugged boreholes, no significant consequences to the public would result.[1] Nevertheless, it is important to assure necessary repository integrity, and accordingly adequate penetration seals are required to minimize or prevent:

- The movement of fluids or gas through the repository
- The movement of water to the repository (particularly in soluble host rock)
- The migration of radionuclides if flow does occur
- The evolution of hydraulic pathways or short-circuits away from the repository, ie, to maximize travel time of radionuclides that conceivably might migrate

1) Status of Technology

Since the first use of cement to shut off water in oil wells in the late 1800s, cement technology for borehole sealing has grown to include applications in natural gas and geothermal steam recovery. Essentially the same technology is also used in wells dug for water, nonnuclear waste disposal, in situ uranium leaching, and Frasch process sulfur recovery.[2] In this section "oil well cementing" refers to all of these.

Oil well cementing technology is well established in the United States and Europe and provides an excellent basis for developing borehole plugs to be used in nuclear waste repositories. Plugging is used in several basic applications:

- To grout a fractured stratum to protect the formation and seal off low-pressure zones
- To segregate strata, isolate zones, or block an aquifer, thus preventing infiltration of unwanted fluids and gases
- During directional drilling to divert drilling equipment (whipstocking)
- To support and protect downhole tubular equipment
- To abandon a hole and prevent interzonal migration of fluids and gases.

Oil, gas, and geothermal operators, service companies, and state regulatory personnel have expressed great confidence in current cementing technology. Short-term effectiveness has been achieved, but durability over the long time periods needed for radioactive waste isolation has yet to be proven.[3]

Shaft (large-diameter hole) sealing techniques have been developed for mining, chemical waste disposal, hydrocarbon storage, and nuclear explosive research. Types of seals developed have been governed by the function required of the seal in each case. Although certain similarities (eg, materials used) exist between plugging small boreholes and sealing shafts, shaft sealing is not a simple extension of borehole-plugging techniques. The most notable differences are that shaft seals normally (1) require more extensive structural design investigations, and (2) could be installed and inspected by personnel physically present at the working face, while borehole sealing is fully dependent on downhole tools and instruments.[4]

Shaft sealing for mining purposes can provide useful data on existing types of various sealing methods. Mine shaft sealing is usually done to deny physical access to abandoned workings or to limit water or air flows for personnel safety or pollution abatement. Valuable detailed studies of shaft sealing techniques have been conducted by the National Coal Board in Great Britain and by the Environmental Protection Agency in the United States.

However, some of the most useful information available on the strength and durability of underground shaft seals designed primarily to contain water has come from the gold mine workings in South Africa, where many seals have functioned adequately for more than 30 years.[5]

Some methods of mine shaft sealing have potential application to a nuclear waste repository. Examples of such methods are the double bulkhead seal and clay seals. Double bulkhead seals are constructed by placing two retaining bulkheads in a shaft opening with a seal in the space between the bulkheads. This seal consists of grout or concrete placed through pipes in the bulkheads or through vertical boreholes. Under some circumstances grouting of adjacent strata may be necessary to prevent leakage around the sides of the seal.[6]

Major emphasis was first placed on plugging technology for waste disposal purposes by the AEC in 1972. In 1973 and 1974 the use of cements for borehole sealing was evaluated and a test seal was emplaced in a borehole near Lyons, Kansas.[7,8] The results of that test were considered satisfactory, although it was suggested that the epoxy caps at the top and bottom of the seal could be replaced by clay or shale to improve long-term durability. Borehole plugging studies by the Massachusetts Institute of Technology using clay or shales showed that such plugs would have very low permeability and long durability. During 1975 and 1976 a series of feasibility studies and engineering analyses related to various plugging materials and methods (including compacted earthen materials, earth melting in salt, hydrothermal cements, and calcite plugs) were performed.[10,11,12,13,14] In 1977 and 1978 development of sensors for monitoring borehole plugs was evaluated,[15] a study of cement grouts for penetration seals was performed by the Corps of Engineers,[16] and specific hydrologic data from a borehole on the Hanford site was obtained.[17] In addition, ONWI has developed a plan for borehole plugging field testing.[18]

The NWTS repository sealing program has characterized the status of penetration sealing technology generally as follows:[19]

- The technology of how to seal boreholes and shafts exists. The capability for borehole sealing exists in the oil industry and the shaft sealing capability exists in the mining industry.
- We can adequately measure and evaluate in situ permeability of seals

- In general, we adequately understand the mechanical and chemical durability of sealing materials and can reasonably show long-term durability of some of these materials for up to several thousand years in some applications.
- The major areas requiring further development and testing relate to long-term compatability of sealing materials with geologic media and long-term interactions under the repository environment conditions.

2) Alleged Gaps and Uncertainties

The major gaps and uncertainties in this area relate to the long-term compatibility of the seal materials with the host rock and geologic media and the long-term performance of the seal system in the repository system environment. It is primarily these long-term considerations that result in the need for the NWTS program to validate the application of penetration sealing technology to the nuclear waste repository. This extensive on-going program has been described and documented.[20] Its major elements include:

- Characterization of seal-host environment
- Seal material considerations
- Engineering design of plugs and seals, including emplacement methods and equipment
- Performance modeling and consequence/risk analysis
- Field testing
- Instrumentation.

ONWI has initiated plans and programs to develop materials, emplacement techniques, and equipment for repository sealing which are compatible with conditions that may be present at the site selected for a geologic repository. Laboratory investigations of material-rock interactions are in progress, as are field investigations of geochemical conditions in candidate media. These activities will lead to field testing and demonstration of satisfactory plug designs.[21]

Plugging material studies will be conducted to establish the longevity of the seals in the media. Laboratory investigations of plugging material/rock interactions will be carried out with supporting investigations of material stability, field investigations of geochemical conditions at candidate locations, and appropriate laboratory tests of chemical interactions. Materials studies will lead to the selection of stable plugging materials (currently, cements are being emphasized) that will meet design requirements. Field testing will be conducted at candidate sites to ensure confidence in the design criteria and specifications for the plugs. Although emphasis at present is on borehole plugging, shaft sealing investigations are being planned as well as studies of tunnel/chamber seals and evaluations of backfilling.[22]

Ongoing analytical and laboratory programs for borehole plugging deal with generic evaluations of modeling methods, materials, and instrumentation, and site-specific applications for the WIPP project in New Mexico and the Columbia River Basalt Project in Washington State.

The major related hydrogeologic and geochemical modeling effort is by Battelle Northwest Laboratories where the Waste Isolation Safety Analysis Program (WISAP) is being developed. Laboratory testing on cementitious grouts is being continued at Sandia Laboratories, ORNL, the Corps of Engineers Waterways Experiment Station (WES), and at Penn State. These programs are aimed primarily toward optimization of cement grout mixtures for testing and include efforts to examine longevity through geochemical analysis and accelerated testing.

Two significant instrumentation development programs are also underway. The IRT Corporation[23] is completing development of an experimental wireless in situ monitoring device. Instrumentation techniques using wires are being developed by Sandia Laboratories in conjunction with field tests on borehole plugs. Sandia is conducting a 2-year program of major field testing using modifications of currently available plugging techniques. This program includes:

- Installation and testing of a plug in a 4000-feet-deep hole
- A variety of tests in a shallow hole with emphasis on operating in zones with aquifers and in recovering portions of the plug for laboratory testing
- Diagnostic testing in a borehole drilled so that it will be intercepted by extension of an existing underground potash mine.

A borehole-plugging test and demonstration program is also planned for the Basalt Waste Isolation Project (BWIP). The initial efforts for this project will be concerned with determination of materials suitable for the basalt environment.

Investigations related to geothermal wells, undertaken by DOE's Division of Geothermal Energy (DGE), are significant because:

- Applications will be for wells at 300C to 6000 feet
- Fluids encountered will primarily be brines with pH in the 4.0 to 5.0 range
- Grouts will be designed to function properly for at least 20 years when exposed to temperatures as high as 400 C.

Research programs to develop both grout materials and installation techniques to satisfy these conditions are being conducted by numerous universities, laboratories, and industrial companies.

3) <u>International Activities and Experience</u>

Borehole plugging is an issue being considered for future research and development efforts in Europe. Current and past efforts in Europe have largely been confined to industries such as mining, oil, and gas.

Borehole plugging is common practice in these European industries as well as in many civil engineering works. Techniques have often been developed by the same international contractors active in the United States. Therefore, many European practices are similar to those in the United States and throughout the world.

The development of shaft sealing techniques is more extensive in Europe because of requirements to stabilize many old mine shafts. Examples of general shaft sealing methods include:

- Combinations of grouting in collapsed zones and backfilling open portions of shafts with soil materials
- Filling with combinations of gravel and grout
- Providing concrete caps above shafts which are completely or partially backfilled or completely empty.

Grouts used in both borehole plugging and shaft sealing are predominantly cement-based with various additives similar to those used in the United States. Several recent developments in Europe have resulted in grouts which appear to have significant advantages for some applications. One particularly noteworthy development is colloidal concrete (COLGROUT^R), produced by rubbing or shearing actions during high-speed mixing. It has been used in nuclear power plant applications and has been found to provide a more complete seal, particularly underwater.

4) Conclusions

A substantial body of technical information and experience in the plugging and sealing of boreholes and shafts has resulted from operations over the years in the petroleum and mining industries. While the longevity requirements for penetration seals associated with a nuclear waste repository represent special needs, these needs have been recognized and identified and are appropriately reflected in ongoing and planned NWTS program objectives and plans. The materials and geochemical investigative efforts, modeling, and analytical work should enable reasonable prediction of seal performance and life based on sound scientific principles. These parts of the overall repository sealing program coupled with field testing and the current knowledge and experience base lead to the conclusion that such seals can be designed and emplaced in a manner consistent with the performance requirements for the overall repository system.

REFERENCES FOR SECTION III-F

1. C L C Christensen, Field Test Programs of Borehole Plugs in Southeastern New Mexico, Proceedings of the NWTS Program Information Meeting (ONWI-62, October 30, 1979).

2. D'Appolonia Consulting Engineers, Inc., The Status of Borehole Plugging and Shaft Sealing for Geologic Isolation of Radioactive Waste, Office of Nuclear Waste Isolation, Battelle Memorial Institute (ONWI-15, January 1979).

3. Id.

4. Id.

5. Id.

6. Id.

7. Dowell, Division of Dow Chemical Co., Borehole Sealing Final Report--March 1, 1973 to October 31, 1973, Oak Ridge National Laboratory (ORNL/SUB-73/15966, 1973).

8. Dowell, Division of Dow Chemical Co., Sealing AEC #1 Well--Lyons, Kansas--Final Report, Oak Ridge National Laboratory (ORNL/SUB-74/33542, February 1974).

9. Massachusetts Institute of Technology, Research Rpt, 75-28, Feasibility of Sealing Boreholes with Compacted Natural Earthen Material (June 1975).

10. Astronuclear Laboratory, Westinghouse Electric, A Study of Borehole Plugging in Bedded Salt Domes by Earth Melting Technology, Oak Ridge National Laboratory (ORNL/SUB-75/76230, June 1975).

11. Pennsylvania State University, Borehole Plugging by Hydrothermal Transport--A Feasibility Study, Oak Ridge National Laboratory (ORNL/SUB-4091/1, May 1975).

12. Pennsylvania State University, Borehole Plugging by Hydrothermal Transport--An Interim Report on Experimental Studies, Oak Ridge National Laboratory (ORNL/SUB- 4091/2, July 1975).

13. Pennsylvania State University, Borehole Plugging by Hydrothermal Transport--Final Report, Oak Ridge National Laboratory (ORNL/SUB-4091/3, February 1976).

REFERENCES FOR SECTION III-F (CONT.)

14. Charles Stark Draper Laboratory, Inc., Borehole Plugging by Compaction Process - Final Report, Office of Waste Isolation, U.S. Department of Energy (Y-OWI-Sub-7087-1, August 1976).

15. IRT Corporation, Development of a Sensor System for Monitoring Borehole Plugs, for Office of Nuclear Waste Isolation, Battelle Memorial Institute (IRT-0055-006, July 1978).

16. Concrete Laboratory - U.S. Army Engineer Waterways Experiment Station, Borehole Plugging Program (Waste Disposal): Initial Investigations and Preliminary Data for U.S. Department of Energy/Sandia Corporation (Misc. Paper C-78-1 (January 1978).

17. Science Applications, Inc., Hydrologic Testing in Borehole DC-2, Rockwell Hanford Operations (RHO-BWI-C-36, September 1978).

18. Office of Nuclear Waste Isolation, Battelle Memorial Institute, Development of Plan and Approach for Borehole Plugging Field Testing, Meeting Report (ONWI-3, August 1978).

19. Office of Nuclear Waste Isolation, Battelle Memorial Institute, Program Plan and Current Efforts in Repository Sealing for the National Waste Terminal Storage Program (ONWI-54, October 1979).

20. Id.

21. Office of Nuclear Waste Isolation, Battelle Memorial Institute, National Waste Terminal Storage Program Plan (ONWI-19, Rev. 1, December 1979).

22. Id.

23. Op cit, IRT (Ref. 15)

G POST-CLOSURE MONITORING AND PREDICTION OF LONG-TERM REPOSITORY PERFORMANCE

One of the basic purposes of geologic disposal is to separate wastes from the biosphere (Barrier #5, figure I-8 and figure III-H-I) to such a degree that extensive surveillance will not be needed once the repository is closed. For a limited time after the repository is closed, certain of the operational monitoring programs could be continued. This is particularly true of temperature and radiation measurements which could be continued with remote readout systems left in place when the repository is filled, closed, and sealed.

It should first be emphasized that, based on the high level of confidence with which we can select a site, design the repository, and package the waste, it is extraordinarily unlikely that monitoring will reveal any measureable movement of radioactive materials. Nevertheless, some additional monitoring measures will probably be required to assure people that no gross underestimate of risks has been made. It should be noted, however, that any monitoring for the escape of radioactivity from the waste packages, or even from the repository complex itself, will have to be done without significant compromise to the integrity of the repository. Accordingly, since the monitors will have to be located at some distance from the radioactivity, they will not provide any indication of movement of radioactivity in less than centuries. It is therefore apparent that post-closure monitoring realistically will not contribute to assessment of long-term repository performance. As we note below, however, monitoring of human activities may be able to provide some additional assurance that the repository system will not be compromised by human intrusion.

Assessment of the long-term performance of the repository must depend upon the availability of analytical models that can be used to predict long-term behavior, as discussed below.

1) <u>Status of Technology</u>

a) <u>Prediction of Long-term Behavior--Repository System Modeling</u> A number of methods are available for prediction of repository long term behavior after closure.[1] Models have been developed that analyze the phenomena that might result in the release of radionuclides from the waste and phenomena that might result in their transport to the biosphere and to people.[2-7] Both near-field and far-field performance have been analyzed using models which include the following factors of concern:

- Thermal loading
- Mechanical stress
- Hydrological transport and sorption
- Geochemical effects
- Radiation effects.

The more complicated of these models tend to be those which describe the transport mechanisms. Additional confidence in such models has been gained as a result of comparison with observed transport from natural nuclear reactor sites.[8]

Once site conditions and repository configurations have been determined, established analytical techniques are available to evaluate each of these factors and thereby evaluate performance of the repository with regard to retaining radioactive material A list of selected models appropriate for these analyses is provided in table III-G-1, and a description of each is given in Appendix III-G, to indicate the status of present technology. A more detailed description of some of these models is given in DOE/NE-007.[9] Processes of concern are discussed briefly in the following paragraphs.

<u>Thermal processes</u> Energy will flow from the heat-producing wastes in the repository and through the waste package, into the host rock, up through the overlying strata and, eventually, into the atmosphere. Although the quantities of heat are not large, the effects could modify ambient hydrologic and mechanical stress fields and these effects need to be considered. Many excellent numerical models exist to analyze the heat conduction and other heat transfer processes involved and have been successfully applied to repository analysis.

Table III-G-1

List of Analytical Models

System Analysis Models

NUTRAN
REPRISK
BARIER
TRENCH
CALMAC
AMRAW
MACRO1
WISAP

Fluid and Mass Transport

SWIFT
AT123D
FEMWATER
FEMWASTE
COOLEY
TERZAGI
PRICKETT AND LONNQUIST
KONIKOW-BREDEHOEFT
GETOUT
HART
GENAESIS
GROVE/GALERKIN
FRESURF 1 and 2
FREEZE
FLUMP
FEG
FARMER
DPRW
DICKMAN
DAVIS/FE3D
COLORADO-FD
COFAM
BRINE
BEWTA
AQUAMOD
MIGRAIN
GWTHERM
UNSATZ

Thermal Processes

ADINAT
BASFEH
CCC
COX
COYOTE
FAUST-MERCER
FLLSSM

Thermal Processes (cont)

GLM
HEATINGS
HYDRA1
REPRESS
SHAFT/79
SINDA/CINDA 3G
SPECTROM 41 (TRANCO)
TRUMP

Mechanical Processes

STEALTH
CAVS
JUDITH
ADINA
BUMINES
COUPLEFLO
DAMBIT
DAMSEL
MARC-CDC
PORFRC2
REPOS
SANGRE

Chemical Processes

EQ3/EQ6
ARDISC
BIOSSIM
FASTPATH
LEVINE
MINEQL
STM
WATEQ

Nuclide Depletion and Generation

CINDER
ORIGEN

These models take into account the complex geometry involved, the nonuniform rock properties, and the time dependence of the radioactive heat generation rate. Models exist to couple the heat transport to the mechanical stress fields or to the hydrologic flow fields.

Mechanical processes The mechanical stress field will be perturbed both by the induced thermal environment and by the mined construction of the repository. The analyses of these perturbations are supported by a vast body of knowledge and experience from mining engineering analyses. In particular a variety of finite element and finite differences models have been applied to the evaluation of the rock mechanics associated with the repository. These models are multidimensional, and can take into account nonlinear properties such as creep in the more plastic rocks, as well as fractures in jointed media.

Hydrologic fluid and contaminant transport The most important cause of nuclide transport from the repository is the groundwater flow system. Once nuclides have been released from the waste package and engineered containment, they can be transported subject to macroscopic fluid velocities and chemical retardation effects. Sophisticated models have been created to treat these processes; they take into account permeability, porosity, storage coefficients, and leakage. Analytical techniques to treat flow in saturated and unsaturated systems have been developed. Buoyancy effects due to the heat expected in the formation can be handled as well. Multiphase systems can also be treated. Models exist which explicitly take into account sorption and desorption effects as well as radionuclide depletion and daughter production. Flow in fractured media is currently treated in an approximate way based on secondary characteristics of the disturbed zones. However, development of improved theoretical treatments of fractured media flow are underway.

Geochemical processes The geochemical processes can cause the nuclides to move more slowly than the groundwater itself. Among these processes are ion exchange, precipitation, and surface adsorption of ions and colloids. Ion exchange, which is often the dominant process, is reasonably well understood and can be straightforwardly modeled as a retardation factor in the transport codes. Concentration of nuclides in the groundwater can be treated by a number of sophisticated aqueous chemistry models provided the equilibrium assumptions used are applicable. Excellent models exist for this purpose.

Radiation effects Nuclear radiation effects are limited to the near field and are not expected to be important with respect to the long-term performance of the repository system. Nevertheless a number of sophisticated transport codes exist which can be applied in this area. For example, the gamma and particle radiation to the rock surrounding the canister has been analyzed for a repository.

System models The containment of nuclides depends upon the coupled effects of many processes. For the analyst this means the simultaneous solution of a large set of coupled equations. Systems models which take into account the coupling and interdependency of the processes are being assembled and have been successfully applied to sensitivity studies for generic repository concepts.

In general, the development of the above analytical techniques has been more complete for processes described on a generic basis. For example, for the thermal and the hydrologic analyses this development has been possible because of the possibility of specification of the repository system on a generic basis. Data for specific sites is yet to come and, therefore, the final development and application of appropriate models has not yet been made in detail. For those cases where processes have already been specified on a local basis, there has been no difficulty in providing analytical tools to evaluate them. For example, the problem of brine migration in bedded salt formations was successfully taken into account in models once experiments were performed to measure induced migration rates, and the quantities of brine migrating in the vicinity of the repository had been reliably predicted and shown to be insignificant.

b) Long-term Safety Assessment Studies The subject of long-term safety assessments through use of predictive models is dealt with comprehensively in a companion document, "Long-term Safety of Nuclear Waste Disposal: A Basis for Confidence." That discussion will therefore not be repeated here. Essentially, on the basis of review of a number of models, the report shows that the most realistic and reliable analyses of long-term performance predict potential human exposures which are a small fraction of the variations in natural background even under conditions which assume partial failure of engineered and geologic barriers. For example, in the WIPP Draft EIS,[10] the worst scenario involving liquid breach and transport, initiated 1000 years after repository closure, resulted in a dose received by a maximally exposed person of 0.4 millirem (whole body), compared to background of 100 millirem, even though the assumptions included a leach rate equal to the rate of salt dissolution.

c) Long-term Monitoring Four kinds of post-closure monitoring do appear to be possible: geologic, hydrologic, radiologic, and of human activities.

Geologic monitoring is primarily concerned with detecting variations in geologic parameters that may reveal a possibility for the release of radioactivity, whether the variations are caused by natural geologic events or by the presence of the repository. The fundamental measurement would be periodic resurveys of the surface to observe the depth and areal extent of subsidence associated with closure of the subsurface cavities. In addition, a periodic surface geologic reconnaissance would be conducted for fractures and other phenomena indicative of subsurface movement.

Hydrologic monitoring would continue following the operational phase inasmuch as the more serious long-term concerns for the repository would require transport of radionuclides by groundwater. The basic hydrologic monitoring would consist of periodic sampling and radiobiological analysis of water from open boreholes downgradient from the disposal area.

The postoperational radiologic monitoring program could include measurements of activity levels in biological indicator species. The sampling program would give direct assurance that some unanticipated event has not bypassed the natural and man-made barriers against release of radioactivity and that radionuclides have not been missed in the radiobiological monitoring of downgradient groundwater.

In reality the post-closure monitoring program will have as its major goal the monitoring of human activity in the vicinity. Although EPA has suggested[11] that institutional control not be relied on beyond 100 years, monitoring and other measures such as wide archiving of records concerning the repository, societal memory, and durable markers may reasonably extend the period of institutional control for some centuries. If such control is maintained for as little as three centuries the hazard due to human intrusion into the waste will have been reduced by a factor of 1000. In the companion document, a number of examples are cited of human activities for which records have survived for much longer periods of time that this (Section 2.3.4). The establishment and wide dissemination of information at and about the repository will result in a very high probability that knowledge of the existence of the repository will survive for many hundreads of years.

2) Alleged "Gaps and Uncertainties"

The major potential uncertainty is the question of how well existing models can predict far-field performance over the time period of concern. However, a detailed comparison was made in the companion document of a number of long-term safety studies, using different models and assumptions, involving a number of release scenarios. The results in virtually every case showed human exposure levels significantly below background. This strongly indicates that a well designed repository will effectively limit human exposure to safe levels, and any uncertainty in the models will result only in differing estimates of how small a fraction of background any human exposure would represent.

In addition, an extensive program for improvement of models and expansion of their capabilities is underway as part of the Earth Sciences Technical Plan.[12] This program, as well as the Waste Rock Interaction Program,[13] will contribute to the data base needed to refine long-term predictions. Further model development is also continuing under the Waste Isolation Performance Assessment Program.[14] Finally, the NWTS Field Testing Program[15] is providing additional verification of radionuclide transport models.

3) Conclusions

Some post operational monitoring will be necessary, but monitoring devices cannot be placed in or close to a repository without some potential threat to the integrity of Barriers #4, 5, and probably 6. Therefore monitoring must be done at some distance, and it is extremely unlikely that it will detect any measureable movement of radioactivity over a period of many centuries. By that time the hazard represented by the waste will be little greater than that of the ore body from which it came. The main goal of post-closure monitoring will be to assuage public concern and to prevent human intrusion.

The assessment of long-term performance must be done using predictive models. A number of models are available, and extensive analyses of long-term repository safety have been conducted. Comparison of a variety of models, applied in a conservative way, shows that they all predict acceptable doses to the public as a result of any credible exposure pathway scenario. This agreement supports confidence that models can be used to predict long-term repository performance and that the performance in terms of potential radiation exposure of the public will be well within safe limits.

Appendix III-G
Description of Analytical Models

System Analysis Models

NUTRAN — Monte Carlo for Sensitivity Analysis; 1-D
(Koplik, TASC)

NUTRAN calculates dose to man resulting from radioactivity carried out of waste repositories by groundwater and evaluates post-emplacement risks from the repository; attractive feature is its intrinsic systems concept although some of the individual processes are treated rather simplistically leading to physically unrealistic results.

REPRISK
(Egan, USEPA)

This model calculates unintended releases from a repository subject to probabilities for events inserted as failure rates rather than from a probability density function.

BARIER
(Lester, SAI)

BARIER evaluates performance of the engineered barriers around a canister emplaced in a repository, taking into account material properties and geometry as well as ionic strength and oxygen content of appropriate regions.

TRENCH
(Oztunali, Dames & Moore)

TRENCH relates amount of radionuclides leaving bottom of burial trench per unit time to the current of radionuclides crossing a hypothetical boundary per unit time.

CALMAC
(Rogers, Ford Bacon & Davis)

CALMAC calculates maximum average concentration of nuclides due to transport along various pathways and allows limits for waste inventories to be set in a system modeling approach.

AMRAW
(Logan, Los Alamos)

AMRAW provides calculational methods for risk assessment and economic analysis in radioactive waste management systems.

MACRO1
(Kauffman, LLL)

MACRO1 performs sensitivity analysis for general coupled systems of components or process models.

WISAP Scenario Analysis Model
(PNL)

This pro[illegible]m generates results of selective scenarios and is useful for studying [illegible]e consequences of selected combinations of events.

FLUID AND MASS TRANSPORT

SWIFT — Finite Difference; 3-D
(Campbell, Sandia)

This is a comprehensive nuclide transport code to describe migration through the groundwater system, taking into account the process of nuclide decay and operation of daughter products. Particularly suited to far-field simulation mechanisms.

AT123D — Analytical; 3-D
(Yeh, ORNL)

AT123D provides a generalized computer code for estimating the transport of wastes in aquifer. Desired modifications would be the inclusion of appropriate coupling among the flow, thermal and solute transport mechanisms in the code for a more realistic simulation of these processes.

FEMWATER — Finite Element; 2-D
(Yeh, ORNL)

FEMWATER simulates groundwater dynamics in saturated, unsaturated surface systems; the unsaturated flow mechanism may be particularly useful for certain geomedia.

FEMWASTE — Finite Element; 2-D
(Yeh, ORNL)

FEMWASTE computes waste transport through porous media under dynamic groundwater conditions; simulation through the unsaturated media is its attractive feature.

COOLEY (USGS) — Finite Element; 2-D

The model predicts transient or steady-state hydraulic heat distribution in confined, semiconfined or unconfined aquifer under a wide variety of boundary conditions.

TERZAGI (Narasimhan, LBL) — Finite Difference; 3-D

TERZAGI solves for three-dimensional fluid flow with 1-D consolidation in saturated systems; may be relevant for soils that exhibit compressibility under overburden pressure.

PRICKETT AND LONNQUIST (Prickett, Lonnquist, ISWS) — Finite Difference; 2-D

This is a generalized code that can simulate the flow of groundwater in heterogeneous aquifer under nonleaky and/or leaky artesian conditions; can also handle water exchange between surface and groundwaters.

KONIKOW-BREDEHOEFT (Konikow, Bredehoeft, USGS) — Finite Difference; 2-D

This is a generalized computer code that simulates solute transport in flowing groundwater; its flexibility and coupling of flow equation with the solute transport equation are two of the attractive features of this computer program.

GETOUT (Lester, SAI) — Semianalytic; 1-D

This model predicts the long-term migration of radionuclides through the geosphere from nuclear waste disposal sites.

HART (Hart, SAI) — Finite Difference; 2-D

HART calculates groundwater flow in the presence of changing mechanical rock stress--recent addition to STEALTH.

GARD (Rosinger, AECL) — Semianalytic; 1-D

GARD can be used to calculate the rate of movement of radionuclides from a proposed deep underground vault to the surface or near surface environment.

GENAESIS Finite Element; 2-D
(Huyakorn, Dames and Moore)

The three-dimensional version can handle coupled single phase flow, thermal and contaminant transport. The 2-D version can handle multiple phase flow in heated porous medium.

GROVE/GALERKIN Finite Element; 2-D
(Grove, USGS)

This model solves mass-transport equations, including radioactive decay; application to a field problem is demonstrated.

FRESURF 1 and 2 Finite Element; 2-D
(Neuman, Univ. of Arizona)

This model solves both 2-D and axisymmetric flow problems.

FREEZE Finite Difference; 3-D
(Freeze, IBM)

This model calculates water flow in a groundwater basin under saturated-unsaturated conditions with a transient.

FLUMP Finite Element; 2-D
(Neuman, Univ. of Arizona)

This model simulates 2-D groundwater flow.

FEG Finite Element; 2-D
(Mercer, Princeton Univ.)

This model can be used to simulate flow and heat transport in a groundwater system.

FARMER Finite Difference; 3-D
(Farmer, LSU)

FARMER calculates the saline plume associated with groundwater flow in the vicinity of a salt dome.

DPRW Monte Carlo
(Ahlstrom, BPNL)

This is a general simulation model for a variety of environmental transport processes.

DICKMAN Numerical
(Dickman, EG&G)

This model predicts transport of nuclides through soil.

DAVIS/FE3D Finite Element; 3-D
(Gupta, Univ. of California)

This model predicts transient piezometric heads and salt transport in large natural multiaquifer basins.

COLORADO-FD Finite Difference; 3-D
(Brutsaert, CSU)

COLORADO-FD is a three dimensional, partially saturated flow model.

COFAM Semianalytical
(Lu, N.Y. State Dept. of Health)

This is a simplified mathematical model for analyzing the migration of leachate and radioactive material contained in radioactive waste burial trenches.

BRINE Numerical
(Fuller, LLL)

BRINE predicts migration of brine inclusions in groundwater flow near heat source.

BEWTA Implicit F.D.; 2-D
(Lin, M.Y. Dept. of the Environment)

The model simulates the Boussinesq Equation for a Two-Dimensional Water Table Aquifer.

AQUAMOD Numerical
(Booth, ORNL)

This code is used in analyzing radionuclide transport between receiving waters and bottom sediments and the resulting doses to man.

MIGRAIN Finite Difference; 3-D
(SAI)

MIGRAIN calculates the interstial flow of a compressible fluid and is applicable to the migration of inclusions of water in rock.

GWTHERM Finite Difference; 2-D
(Runchal, Dames & Moore)

GWTHERM computes the flow of water taking into account the buoyancy effects induced by a heat source.

UNSAT2 Finite Difference, 2-D
(Neuman, U. of Arizona)

UNSAT2 is a code to evaluate unsaturated fluid flow in porous media.

ADINAT Finite Element; 3-D
(Bathe, MIT)

ADINAT is a general purpose heat conduction code compatible with the stress code ADINA.

BASFEH Finite Element; 2-D
(BWIP)

BASFEH is a heat conduction code developed for the basalt waste isolation project.

CCC Finite Difference; 3-D
(Tsang, LBL)

Analysis of response of geothermal reservoirs under injection and production procedures.

COX
(Cox, ORNL)

COX calculates radiation heat transfer in spent fuel canisters.

COYOTE — Finite Element; 2-D
(Gartling, Sandia)

COYOTE solves the two-dimensional nonlinear heat conduction problem.

FAUST-MERCER — Finite Difference; 2-D
(Mercer, GeoTran)

This coupled model can be used to simulate two-phase flow and heat transport in a groundwater system; needs modification to adapt it to respository-related phenomena.

FLLSSM — Semianalytic; 3-D
(Julien, Kaiser Engineers)

FLLSSM computes temperatures in rock due to a finite array of line sources representing waste canisters.

GLM — Semianalytical; 3-D
(Rickertsen, SAI)

GLM predicts approximate temperature, thermolastic stresses and groundwater flow near a waste repository; the model should be useful in scoping the effect of heat generating wastes in geologic media.

HEATING5 — Finite Difference; 3-D
(Turner, ORNL)

HEATING5 can be used in a variety of heat transfer problems associated with the emplacement of nuclear wastes in a repository; advantage is the simplicity of input.

HYDRA1 — Finite Difference; 3-D
(McCann, PNL)

HYDRA1 computes the temperatures in spent fuel canisters due to radiations, conduction and convection heat transfer processes.

REPRESS
(SAI)

REPRESS is a code which calculates pressures in a two-phase system knowing the total amount of fluid and total volume present.

SHAFT79 Finite Difference; 3-D
(Pruess, LBL)

The simulator SHAFT79 can be used to study the depletion of different types of geothermal reservoirs for a variety of idealized model reservoirs; this code developed for geothermal applications will require modifications in order to adapt it to repository-related processes.

SINDA/CINDA 3G Finite Difference; 3-D
(Sandia)

SINDA computes temperatures for general heat conduction problems.

SPECTROM 41 (TRANCO) Finite Element; 2-D
(Callahan, RESPEC)

SPECTROM 41 is a general heat conduction code. A 3D version, SPECTROM 341, has also been developed.

TRUMP Finite Difference; 3-D
(Edwards, LLL)

TRUMP is a flexible 3D heat conduction code which has been applied to waste isolation problems.

MECHANICAL PROCESSES

STEALTH Finite Difference; 3-D
(Hofmann, SAI)

General purpose static and dynamic thermal and stress analyses suitable for structural analysis and rock mechanics for repository studies.

CAVS Finite Difference; 3-D
(Maxwell, SAI)

CAVS is a constitutive model to describe rock fractures due to the thermomechanical loadings in waste repositories; can also be interfaced with STEALTH.

JUDITH Semianalytical; 2-D
(St. John, Univ. of Minnesota)

JUDITH predicts thermoelastic effects due to time varying heat sources at finite depth; may be useful for scoping the effects of themal gradients on stress field.

ANSR-1 Finite Element; 3-D
(Mondkar, Univ. of California)

ANSR performs analysis of nonlinear structural response.

ADINA Finite Element; 3-D
(Bathe, MIT)

ADINA is a general purpose finite element code which is quite versatile and widely used throughout the stress analysis community.

BUMINES Finite Element; 3-D
(Agbabian Associates)

This model performs geochemical stress analysis, taking into account linear as well as nonlinear material properties.

COUPLEFLO Finite Element; 2-D
(Dawson, Cornel U.)

COUPLEFLO is a program to compute coupled creep and conductive-convective heat transfer.

DAMBIT Boundary Element; 3-D
(Hocking, Dames & Moore)

DAMBIT performs two- and three-dimensional analyses for stress near rooms in repositories.

DAMSEL Finite Difference; 2-D
(Hocking, Dames & Moore)

Modeling of stresses in rocks including elastic and plastic effects.

MARC-CDC Finite Element; 3-D
(Singhal, ORNL)

MARC-CDC can be used to calculate stresses in continuous media.

PORFRC2 Finite Element; 2-D
(Chan, LBL)

To model fully coupled water flow and stress in a porous medium.

REPOS Boundary Element; 3-D
(Sinha, Terra Tek)

REPOS performs stress analysis on repository-scale situations.

SANGRE Finite Element; 3-D
(Anderson, Los Alamos)

SANGRE predicts long term transient creep of geomechanical structures which obey an arbitrary creep law.

CHEMICAL PROCESSES

EQ3/EQ6 Finite Difference
(Wolery, LLL)

This comprehensive computer code package performs distribution of species calculations for natural water systems and equilibrium models of aqueous geochemical systems--particularly suited for geochemical simulations.

ARDISC Semianalytical; 1-D
(Strickert et al, ANL)

ARDISC (Argonne Dispersion Code) simulates the migration of nuclides in porous media.

BIOSSIM Numerical
(Garfinkel, Univ. of Pennsylvania)

This code calculates the time course of chemical reactions in chemical (or biochemical) systems.

FASTPATH Numerical
(Apps, LBL)

FASTPATH models the chemical evolution of a complex chemical system with water present as a function of reaction progress.

LEVINE Numerical
(Levine, LLL)

LEVINE computes the thermodynamic equilibrium for large chemical and geochemical systems.

MINEQL Numerical
(Westall et al, MIT)

MINEQL calculates the chemical equilibrium composition of aqueous systems.

STM Numerical; 2-D
(Grove, USGS)

This Solute Transport Model deals with the transport of waste including radionuclides in groundwater and the reaction between the waste and its environment. The applicability of STM is in its capacity to predict radionuclide migration in a given subsurface environment.

WATEQ Numerical
(Truesdell & Jones, USGS)

WATEQ calculates the equilibrium distribution of inorganic aqueous species of major and important minor elements in natural waters using chemical analysis and other methods.

NUCLIDE DEPLETION AND GENERATION

CINDER
(England, LASL)

This model computes fission products, actinide inventories and depletion.

ORIGEN
(Kee, ORNL)

This is a general isotope generation and depletion code.

REFERENCES FOR SECTION III-G

1. H.C. Burkholder, Waste Isolation Performance Assessment--A Status Report, Office of Nuclear Waste Isolation, Battelle Memorial Institute, Columbus, OH, (ONWI-60, 1979).

2. S.E. Logan and M.C. Berbano, Development and Application of a Risk Assessment Method for Radioactive Waste Management, U.S. Environmental Protection Agency (EPA 520/6-78-005, (1978).

3. J.E. Campbell et al., Risk Methodology for Geologic Disposal of Radioactive Waste: Interim Report, Sandia National Laboratories (SAND 78-0029, 1978)

4. Arthur D. Little, Inc., Technical Support of Standards for High-Level Radioactive Waste Management, U.S. Environmental Protection Agency (EPA-520/4-79-007, 1979).

5. J.A. Stottlemyre, et al, A Conceptual Simulation Model for Release Scenario Analysis of a Hypothetical Site in Columbia Plateau Basalts, Battelle Pacific Northwest Laboratory (Draft PNL-2892, 1979).

6. J.A. Stottlemyre, et al, "Computer Enhanced Release Scenario Analysis for a Nuclear Waste Repository," Scientific Basis for Nuclear Waste Management, Vol. 1, Plenum Press, New York, N.Y. (1979), p. 521.

7. Long-Term Safety of Nuclear Waste Disposal: A Basis for Confidence, The Analytical Sciences Corporation (TR-3195-1, July 9, 1980).

8. G.A. Cowen, Migration Paths for OKLO Reactor Products and Applications to the Problem of Geological Storage of Nuclear Wastes, Los Alamos National Scientific Laboratory, Los Alamos, NM, (LA-UR-77-2787, 1977).

9. U.S. Nuclear Regulatory Commission, "In the Matter of Proposed Rulemaking on the Storage and Disposal of Nuclear Waste," Statement of Position of the U.S. Department of Energy (DOE/NE-007, April 15, 1980).

10. U.S. Department of Energy, Draft Environmental Impact Statement, Waste Isolation Pilot Plant, (DOE/EIS-0026-D, April 1979).

11. U.S. Environmental Protection Agency Criteria, 43 Fed. Reg. 53264, (November 15, 1978).

12. Office of Nuclear Waste Management, Battelle Memorial Institute/U.S. Geological Survey, Earth Science Technical Plan for Disposal of Radioactive Waste in a Mined Repository, (Draft, DOE/TIC-11033, April 1980.

REFERENCES FOR SECTION III-G (CONT.)

13. Proceedings of the WISAP Task 4 Second Contractor Information Meeting, Battelle Pacific Northwest Laboratory (PNL-SA-7352, October 1978).

14. H.C. Burkholder, Waste Isolation Performance Assessment Program, Transactions of the American Nuclear Society, (Vol. 33, 1979), p. 380.

15. Office of Nuclear Waste Isolation, Battelle Memorial Institute, Geologic Disposal of Radioactive Waste: Program Plan for Field Tests in Geologic Formations, (Draft ONWI-80 (1), (1979).

H SUMMARY

In the previous portions of Section III the status of technology of the various components of an overall waste repository system has been reviewed. These components, taken together as a total system, provide a high degree of confidence that the waste can be effectively contained by the system, to the extent and for the times required to protect the health and safety of the public. The relationship of each component to overall containment is illustrated in figure III-H-1.* Our conclusions concerning each of the system components are given below.

<u>Site Identification and Characterization (Section III-A)</u>

(Barrier #1)

The role of site selection and characterization is to choose a repository site with characteristics which provide, with a high degree of confidence, assurance that the waste will be adequately contained over the time periods of interest. In our review we have shown that the geologic processes which could influence containment:

1) tectonic movements,
2) igneous activity,
3) rock deformation,
4) erosion/dissolution/deposition,
5) groundwater movement, and
6) climate and related changes

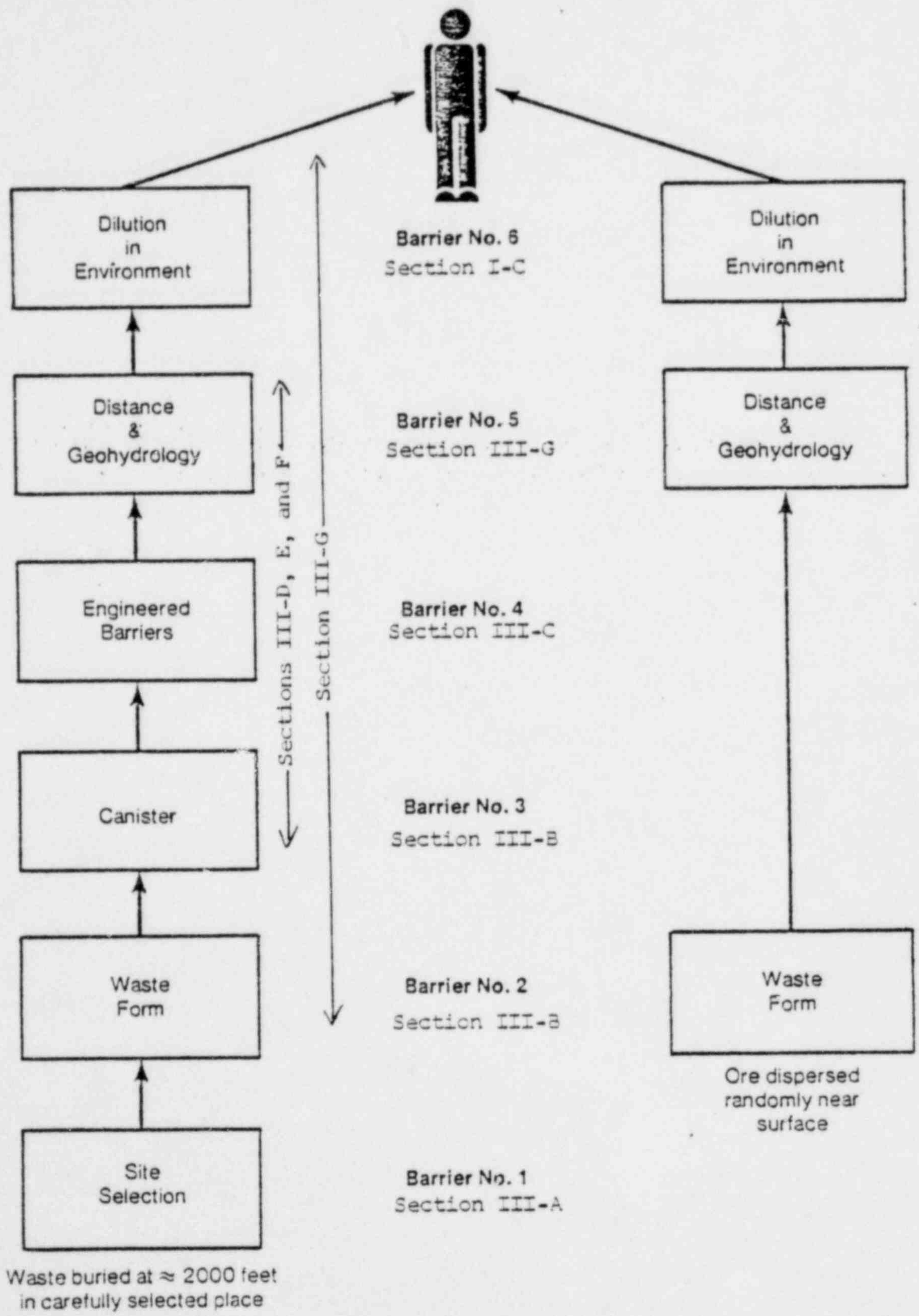
are slow (see table III-A-1), and the rate of change even slower. Thus, in a properly selected site, containment should not be significantly reduced over periods of several hundred thousand to a million years or more. It was also shown that exploratory techniques are available to permit selection of such stable sites. Beyond that it was shown that the period of time over which the

---

* Figure III-H-1 is basically the same as figure I-8, with the addition of section references to pertinent text discussions.

Figure III-H-1

Barriers between Buried Radioactivity and Man



requirements for a repository exceed those of an ore body is geologically short (about 500 years) so that the very-long-term demands upon a repository have been overemphasized. We conclude that a high level of confidence exists that sites for repositories can be identified and characterized in a timely manner and with adequate assurances of safe construction, operation, and long-term containment.

Waste Form and Package (Section III-B)
(Barriers #2 and #3)

The form of the waste itself, spent fuel,* represents a significant containment barrier in the form of a very low leach rate (Barrier #2) should water ever reach the waste. The high degree of containment of fuel elements themselves has been conclusively and rigorously demonstrated in their use in the reactor, albeit in an essentially pure water environment rather than groundwater. The leach resistance of the fuel elements could, if deemed necessary, be enhanced by the use of a metal matrix or other stabilizers. In any event the waste form will provide a significant barrier which is expected to last over very long time periods.

The canister (Barrier #3) can be expected to last for a sufficient period of time to provide virtually complete containment during emplacement and any retrieval period deemed desirable. It can also provide containment in the important early years when fission products are controlling--the only period when the demands on the repository system significantly exceed those on the ore body.

We conclude that the combination of these two barriers, canister and waste form, plus any additional overpack, clearly provides containment in excess of that which nature provides for the ore body (due to its waste form) in the early years when the repository requirements significantly exceed those for the ore. And in the long run the degree of containment provided by the waste form itself is likely to be comparable to that of the ore body.

---

* It is expected that solidified HLW would be put into a form, g, glass, with a leach rate as low as, or lower than, that of spent fuel.

Additional Engineered Barriers (Section III-C)

(Barrier #4)

The space between the emplacement hole and the waste package and the entire mined-out volume is available for use in providing additional engineered barriers, if deemed necessary. The most likely use of this capability may well be to select materials aimed at the potential migration of specific isotopes which modeling or RQ* studies indicate should be further restricted by additional containment barriers.

We conclude that this barrier, which may not be needed at all and which is not available to the ore body, represents an additional tool that may be brought to bear as additional conservatism or as modifiers for particular isotopes.

Repository Design and Construction (Section III-D)

(Impinges on Barriers #3, 4, and 5)

In the strictest sense repository design and construction do not represent containment barriers in themselves. However, proper design and careful construction are required to assure that all of the barriers (most particularly Barriers #3, 4, and 5) function well. We have presented an extensive review of the status of design studies here and abroad and of the capability for constructing such a facility. We conclude that a repository can be designed on as conservative a basis as needed.

Waste Emplacement (Section III-E)

(Impinges on Barriers #3, 4, and 5)

Clearly the waste emplacement operations must be carried out in such a manner as not to cause deterioration of any of the barriers, particularly #3, 4, and 5. The technology available for handling spent fuel and for underground operations is developed to the point that there is no doubt this can be

---

* See Section I-C.

accomplished. We conclude that the wastes can be emplaced safely and without detrimental effects upon the containment barriers.

Repository Closure--Backfilling and Sealing (Section III-F)

Impinges on Barriers #3, 4, and 5)

Here again the operations of backfilling and sealing are not barriers to containment in the strictest sense, but the care and efficacy with which they are done may well dictate the degree of confidence, particularly for Barriers #4 and 5. Most, if not all, of the added containment supplied in Barrier #4 (engineered barriers) will come from materials and operations used during backfilling. The care used in selection and control of materials and physical properties (eg, packed density) of the backfill will largely determine how effective the engineered barriers will be.

Similarly, it is clear that shafts and boreholes represent initial penetrations though the overlying geologic structure, and thus are potential planes of weakness in Barrier #5 (Distance and Geohydrology). How well shafts and boreholes are sealed represents an important aspect of the degree of confidence in the integrity of the surrounding geologic structures. We have reviewed the extensive experience and literature which exists in the petroleum and mining fields and the impressive amount of work being conducted in the DOE program. We conclude with a high degree of confidence that backfilling, sealing, and closure operations can successfully enable the engineered barriers and the surrounding geologic structures to provide the desired degree of containment.

Assessment of Long-term Repository Performance (Section III-G)

In this section, we discussed the manner in which the overall system performance may be assessed, including the role of post-closure monitoring. Monitoring presents a fundamental dilemma. If monitoring devices, particularly those aimed at direct measurement of radioactivity in water, are placed very close to or in the repository, the monitoring devices themselves would represent a potential by-pass of Barriers #4, 5, and probably 6. Consequently any direct monitoring should be done at some distance; this means, however, that no positive indication can be expected, even under exaggerated hypothetical conditions, for a period of many centuries. Thus, while monitoring will certainly be put in place--to assuage even unreasonable concerns, if for no other reason--it has to be recognized that it will not realistically contribute to assessment of long-term repository performance.

Instead, such assessment must depend upon predictive modeling. Our companion document entitled "Long-term Safety of Nuclear Waste Disposal; A Basis for Confidence" reviews in detail the work which has demonstrated the basis for confidence that predictive models can be conservatively applied. It also reviews in detail the results which have been obtained by many authors using some of these models and shows that all reasonable exposure pathway scenarios, even when exercised using conservative assumptions, predict doses to individuals which should be acceptable. These same models and assumptions, when applied to ore bodies, produce predicted doses comparable to those predicted from a repository. This gives us confidence that our analogy to the ore body, which we have carried as a theme throughout this report, is a legitimate and proper one.

Since man has lived successfully with ore bodies throughout his entire existence, and a step-by-step analysis of the repository system shows it to be superior in containment capability to the ore body, we conclude that there can be confidence that long-term performance of the repository can be predicted and will conform to any reasonable requirements that may be imposed.

IV SCHEDULE OF AVAILABILITY OF DISPOSAL SYSTEM

In previous sections of this document we have provided technical and scientific support for the position that the capability and tools exist to select a repository site and to design, construct, and operate an overall disposal system in a geologic repository in a manner that will be safe and environmentally acceptable to current and future generations. A remaining question concerns the prospective schedule for such a program and the relationship of this schedule to the deliberations of the Commission concerning its degree of confidence that these wastes can be either stored or disposed of safely. To answer these questions it is necessary to review the program and schedules projected by DOE and the institutions and institutional processes which might affect the program schedules. We do so in the following subsections.

The review of institutions and institutional processes includes the extent of the commitment of the Executive Branch to implementing the national waste program under the leadership of DOE, the extensive interest shown by Congress in providing not only the funding for the ongoing program but also specific consideration of key policy issues, and the more specific features of interactions with State and local authorities. The discussion below focuses on the institutional considerations which are pertinent to the schedule for availability of a geologic repository.

A DOE PROGRAM AND SCHEDULES

The Fiscal Year 1982 Budget Document[1] of DOE's Office of Nuclear Waste Isolation contains a "summary logic network" showing the steps required to achieve an operational geologic repository by 1997. DOE's Statement of Position in this proceeding states that implementation of its waste disposal strategy will result in the establishment of an operating geologic repository some time between 1997 and 2006 (p I-4). DOE explains that the exact date of operation will depend upon a number of variables. From a technical standpoint the major considerations related to the projected DOE schedule are:

a) the requirement to examine multiple geologic media, including specifically hard rock systems (granite) before the initial development site is selected;

b) the requirement for exploratory shafts before submission of a license application; and

c) the longer construction time necessary in hard rock.

If more extensive site evaluation is required, ie, a fully developed exploratory shaft, the licensing application milestone would be delayed some 3 1/2 years. If examination of the hard rock medium is a prerequisite for site selection, about another year would be added. If the hard rock medium were selected for initial development an additional 2 1/2 to 3 years would be required for repository construction. These are the types of considerations which, together with uncertainties in review and licensing schedules, could delay initiation of repository operation until 2006.

We have carefully reviewed all elements of DOE's programs, the most important of which we have discussed in previous sections of this report. We believe that DOE's present programs are focusing upon all the matters that need to be addressed and that DOE's forecast of potential operational dates includes more than adequate allowance for uncertainties, including the institutional considerations we discuss in Section IV-B.

Even under the constraints discussed by DOE, we believe that the operational date would be closer to 1997 than 2006, since technology exists to proceed in a salt formation for the first repository. In fact, it is our view that the scope and extent of review of alternative media and sites prior to establishment of the first repository need not be as extensive as presently contemplated by DOE and that therefore a repository could be operational before 1997. Thus, for example, a repository could be operational well before 1997 if alternative sites were evaluated without unnecessary subsurface investigations and if the licensing process for the first repository were begun without completion of hard-rock alternative evaluations.

We are convinced that the national interest would best be served by an accelerated program. As we have previously noted, since spent fuel can be safely stored indefinitely, as a practical matter there is no safety reason for expediting the site selection program. However, delay in implementing such a program can only erode the necessary public and political support. The Hearing Board convened to receive nationwide testimony on DOE's DEIS reached essentially the same conclusion.[2] Since commercial and defense-related wastes now exist, and since there is no doubt regarding the Federal responsibility for and commitment to their proper disposal (see below, IV-B), we are convinced that the waste management decisions made during the next several years will recognize the national benefits and imperative in avoiding delay and achieving prompt implementation.

Thus, it is our view that a persuasive case can be made that the first repository can be operational before 1997 if DOE chooses such a course. For purposes of this proceeding, however, the precise schedule if not of critical importance; the fact that technical capability exists to proceed at whatever pace is selected by Federal decisionmakers is sufficient for the Commission to reach affirmative conclusions.

B INSTITUTIONAL CONSIDERATIONS

Equal in importance to the technical issues involved in a program for the management of nuclear waste is the institutional coordination necessary to implement a national waste management program. Successful and timely accomplishment of the goals of the nuclear waste management program requires both Federal/State coordination and coordination among the Federal agencies involved in the program. The major policy considerations are being considered by the Executive Branch and the Legislative Branch of the Federal Government, in consultation with State officials.

The ability of these institutions to implement an acceptable nuclear waste management program in a timely manner depends on an acknowledged and focused commitment to proceed with the Federal program, taking into account the possibly competing interests and concerns of these institutions and of the public. The following discussion deals with the institutional issues and the basis for our confidence that policies and processes in place and under active consideration will ensure both timely and acceptable decisions in the Federal program. A dominant theme underlying the commitment of the Federal agencies and institutions involved is the national interest in assuring implementation of the program. This national interest perspective gives added confidence that institutional commitments will be sustained, thus avoiding possible impediments to timely execution of the program.

1) National Interest Perspective

The backdrop for examination of the capability and commitment of our institutions to deal effectively with nuclear waste management issues is, in the first instance, the very real implications for national security and economic viability of any impediment to nuclear power generation that could result if such issues are not resolved in a timely fashion.

At a time when the uncertainties of foreign oil supplies continue to escalate, the vital importance to this nation of non-oil sources of domestic energy is drastically clear. This national interest perspective, particularly as it highlights the need for continuing nuclear power production, is reflected, for example, in President Carter's December 1979 statement,[3] as follows:

> "We cannot shut the door on nuclear energy. The recent events in Iran have shown us the clear stark dangers that excessive dependence on imported oil holds for our nation. We must make every effort to lead this country to energy security. Every domestic source, including nuclear power, is critical if we are to free our country from its overdependence on unstable sources of high-priced foreign oil. <u>We do not have the luxury of abandoning nuclear power or imposing a lengthy moratorium on its future use</u>. A nuclear plant can displace up to 35,000 barrels per day." (emphasis added)

The current value of the energy produced by nuclear power plants, acknowledged by the President in the above statement, is best put into perspective with the following statistics recently reported by the Department of Energy:[4]

1979 US ENERGY PRODUCTION/CONSUMPTION

Source	Oil Equivalent (million barrels/day)
Crude oil production/Alaska	1.36
Nuclear powerplants (50.6 GWe)	1.31
Oil imports/Saudi Arabia	1.35
Oil imports/Libya	0.65
Oil imports/Iran	0.30

Thus, even if no additional nuclear power plants were to be licensed we can see that the national energy supply contribution from nuclear is roughly equivalent to the current production in the extensive Alaskan area. This production is also roughly equivalent to our largest single import source, ie, Saudi Arabia, two times the Libyan import level, and four times the 1979 Iranian import level. Moreover, nuclear plants in advanced stages of construction will provide roughly the equivalent of an additional 1,500,000 barrels per day, for a total oil equivalent of the current import level from the Arab members of OPEC

That Congress also appreciates the importance to the nation of nuclear energy is reflected, for example, in the consideration by both the Senate and the House of Representatives, during the First Session of the 96th Congress, of proposals for limited nuclear construction moratoria. In both Houses of Congress the debate was dominated by whether such an action was in the national interest, and in both instances, the proposals were defeated overwhelmingly. These solid indications of Congressional opposition to proposals which could hamper future domestic nuclear energy production have been buttressed by numerous legislative initiatives, some of which are discussed below, to deal specifically with the nuclear waste management program.

The recent report (Energy in Transition, 1985-2010) of the Committee on Nuclear and Alternative Energy Systems (CONAES) of the National Research Council/National Academy of Sciences forcefully argues the continued need for nuclear energy utilization and development in the United States. This report is the result of 4 years of labor by over 350 distinguished individuals from highly diverse backgrounds. The report concludes that coal and nuclear power are the only large-scale alternatives to oil and gas in this country and that "a balanced combination of coal- and nuclear-generated electricity is preferable, on environmental and economic grounds, to the predominance of either." In order to meet the projected need for electricity in the coming decades--and this projection presumes a strong conservation effort--the CONAES Committee found by the year 2010 the need for a six-fold increase in nuclear output over actual 1978 nuclear generation (based on a moderate growth scenario) to as much as a tenfold increase (high-growth scenario).

Thus our national interest--whether discussed in terms of national security, economic viability, or environmental considerations--requires continued and increasing utilization of commercial nuclear energy. There is additionally the need for continued use of nuclear energy for US naval reactors. Use of nuclear materials for military programs will continue for the foreseeable future. Each of these programs has generated, and will continue to generate, nuclear wastes.

Because of existing inventories of nuclear wastes from both military and commercial nuclear activities and because of the national interest in continuing such activities, implementation of a nuclear waste management program is not a matter of choice, it is a matter of necessity. In light of this national interest perspective, it is obvious that both Federal and State institutions have every incentive to resolve nuclear waste management issues and to do so on a timely basis.

2) Commitment of the Executive Branch

President Carter's February 1980 statement regarding nuclear waste management policy,[5] based in large part on the extensive consensus building process of the Interagency Review Group (IRG) report, restated the longstanding responsibility of the Federal Government to deal with the existing as well as future waste management needs:

> "Today, I am establishing this nation's first comprehensive radioactive waste management program . . . I share this responsibility with elected officials at all levels of our government . . . For more than 30 years, radioactive wastes

have been generated by programs for national defense, by the commercial nuclear power program, and by a variety of medical, industrial and research activities . . . My objective is to establish a comprehensive program for the management of all types of radioactive waste . . . Our primary objective is to isolate existing and future radioactive waste from military and civilian activities from the biosphere and pose no significant threat to public health and safety."

The President's policy statement addressed not only the very specific roles of various Executive Branch agencies which were active participants in the IRG process, but also the activities of the States and the public in general. This policy statement also clearly acknowledged the existing inventory of Federal (military and research), medical, and commercial nuclear waste which must be dealt with.

a) The DOE Role as Lead Agency As a central element in the refocusing of the nation's waste management programs, the President made it clear that DOE is to play a major role both as a lead agency and as coordinator for the many facets of the program. When DOE was first established, the enabling act[6] explicitly provided for a dominant DOE role in the development and implementation of the nation's nuclear waste program. The President's policy statement[7] reiterated this important lead role of the DOE as follows:

> "In addition, the President's policy calls for full involvement of State, Regional, and local governments and organizations in program planning and execution. The President has designated the Secretary of Energy to be responsible for overall program integration and to establish necessary coordination mechanisms. The Secretary of Energy will assume the lead role for: 1) coordinating all Federal, non-regulatory aspects of waste management; 2) working out effective relationships with the regulatory bodies such as the Environmental Protection Agency and the Nuclear Regulatory Commission; and 3) developing strong and effective ties between the Federal Government and the States on all aspects of radioactive waste, storage, and disposal."

To this end DOE is taking necessary organizational steps. The Assistant Secretary for Nuclear Energy has the overall responsibility provided in the DOE organic statute. Reporting to him is a Deputy Assistant Secretary for Nuclear Waste Management, responsible for managing all aspects, both military and civilian, of the DOE storage and disposal programs. The Deputy Assistant

Secretary also chairs an Interagency Working Committee composed of representatives of DOI (USGS), EPA, NRC, and DOT and charged with assuring proper coordination of the Federal activities for implementation of the national waste plan. Through DOE's Office of Nuclear Waste Management the objectives of the program are implemented. Thus there is not only a formalized mechanism for coordinating various programs and interagency efforts but there is also a well-identified accountability for various facets of the program--an important element in assuring opportunity for systematic reviews of progress and reorientation of program resources.

The President's confirmation of the lead role of the DOE within the Executive Branch in coordinating Federal agency activities and developing relationships with the states is an important step in assuring an effective mechanism for the appropriate level of review and timely decisionmaking required for Federal program implementation.

To complement this organizational commitment to implementing an effective and acceptable program, the President also announced that additional documentation of the National Plan for Nuclear Waste Management will be made available by 1981 and will be updated every 2 years thereafter. As a primary planning document, supplemented by ongoing NEPA and regulatory policy development, this Plan will provide yet another opportunity for public and Congressional scrutiny of the pace and objectives of the program. Since it will be submitted for public review in draft form and will be updated periodically, interested parties will be provided a continuing opportunity to participate in the process of implementing the national program for storage and disposal.

b) The Regulatory Agencies In addition to DOE, EPA and NRC also have important institutional roles in the implementation of the national waste management program.

EPA is responsible for adopting environmental radiation protection standards applicable to the management of all radioactive wastes and has published for public comments a proposed draft of these general standards. It is expected that EPA standards for disposal of HLW will be issued for public comment in the near future.

The pertinent regulatory responsibilities and implementing actions of the NRC are described in the "Summary of NRC Licensing Program for High-level

Radioactive Waste Disposal" filed by the NRC staff in this proceeding on April 16, 1980. As indicated in this NRC submittal the portions of the proposed NRC regulations which would govern the procedural aspects of the licensing review of a geologic repository (Subparts A through D of 10 CFR Part 60) were published for public comment in late 1979, and it is expected that a final regulation will be published in December 1980. An advanced notice of rulemaking pertaining to the technical criteria for regulating geologic disposal of HLW (Subpart E of 10 CFR Part 60) was published for public comment on May 13, 1980 (45 FR 31393), and according to the NRC staff, the current schedule contemplates that a final rule will be adopted in December 1981. The formal public comment process with respect to the substantive content of these criteria, however, has just begun.

Neither EPA nor NRC has proceeded with the development of applicable standards and regulations as rapidly as would have been desired. As discussed in Section IV-A, however, it is evident that the schedules described by DOE in its Statement of Position for achieving operation of the first geologic repository amply provide for the uncertainties in schedule that might arise from the development and implementation of the regulatory program.

3) Institutional Support--Congress

The Congress has an important role in the development and implementation of the national waste management program both in its adoption of national policy in substantive legislation and in its shaping of programs of Federal agencies through the budgeting and oversight processes.

As we discuss below, Congress has consistently funded programs relating to waste management research and development, regulation, and project implementation in a manner that reflects the importance it places on such programs. Moreover, while the Administration was reviewing its waste management priorities beginning in 1977, the Congress undertook a number of important initiatives to direct and support the development of a comprehensive program.

Finally, Congress has begun consideration of important questions relating to Federal/State relations. All of these activities demonstrate without a doubt that Congress will be a constructive force in fostering and assuring the timely resolution of any institutional problems.

a) Funding of Programs; Legislative Initiatives The Congress has consistently appropriated funds as requested by successive Administrations for a variety of research, development, and demonstration activities relating to both the defense waste programs and the commercial spent fuel and waste programs. The levels of funding provided in the programs and the continuing support of Congress reflects the priority placed on an effective resolution of waste management issues. The table below provides the year-by-year funding of key programs since the DOE came into being:

FEDERAL FUNDING FOR KEY DOE PROGRAMS

(millions)

	Commercial/ Spent Fuel	Commercial Waste	Defense Waste
Fiscal Year 1978	$5	$181	$257
Fiscal Year 1979	$11	$212	$256
Fiscal Year 1980	$19	$220	$282
Fiscal Year 1981 (pending)	$21	$299	$376

Total funding authorized by Congress for supporting NRC programs in this same time frame amounts to $84 million.

As part of the funding process, several important initiatives were undertaken by Congress to deal with the specifics of spent fuel and nuclear waste management. One of the earliest such initiatives was that taken by the Congress in 1978 when it approved the funding for an AFR project, scheduled for initial operation by 1983, during its consideration of the Public Works Appropriations for Fiscal Year 1979.[8] Additional funding for this project was included during the deliberations on the Fiscal Year 1980 Energy and Water Development Appropriation bill as an indication of the continuing interest of Congress in providing important momentum for this program.

During its deliberations on the Fiscal Year 1980 DOE authorization bill, the House of Representatives adopted a provision directing DOE to proceed with a program for siting and demonstration of nuclear waste repositories using currently available technology.[9] More recently, in reporting the Fiscal Year 1981 DOE authorization bill for civilian programs,[10] the House Science and Technology Committee approved a comparable provision calling for expedited design, constructi)n, and operation for four technology-demonstration repositories. It is apparent that initiatives such as these will be scrutinized at least annually during the Congressional review of the DOE authorization and appropriation bills to determine that progress is being made to reach program objectives in a timely manner.

Congressional interest in waste policy is also reflected in legislative initiatives outside of the funding process. For example, late in the first session of the 96th Congress, the Senate Energy and Natural Resources Committee reported favorably the "Nuclear Waste Policy Act".[11] This bill includes a provision requiring the establishment of a Federal system for the disposal of all high-level radioactive waste resulting from civilian nuclear activities, including completion of construction and initial operation of at least one demonstration mined-geologic repository by January 1, 1988. This action by the Senate Committee with principal jurisdiction over nuclear waste management policy represents an important statement of the Congressional support for an aggressive and comprehensive program.

There can be no doubt as to the intense interest of Congress in assuring that nuclear waste management issues are resolved promptly. Thus, in addition to the steps being t k n by the Administration, there can be confidence that Congress will play a constructive role in achieving timely resolution of such issues.

b) <u>Congressional Consideracion of Federal/State Relationships</u> As discussed in IV-B-4, the active and effective involvement of State and local government in the siting, licensing, and development of a geologic repository for commercial wastes will be an important element of implementation of Federal waste managment policy.

There have been a number of bills introduced in Congress dealing with Federal/State relationships in the management of commercial radioactive wastes, but none has yet been approved in either house. Congress has, however, undertaken initiatives with respect to similar or comparable projects or programs which, as discussed below, may be useful precedents. In any event, the obvious Congressional interest and the presence of the precedents make it evident that Congress will be prepared to act should a legislative solution appear to be the most effective mechanism for establishing appropriate Federal/State relationships.

The most pertinent precedent appears in the conference report on the DOE National Security and Military Applications of Nuclear Energy Authorization Act of 1980,[12] which directed DOE, in connection with the WIPP project, to "consult and cooperate with the appropriate officials of the State of New Mexico with respect to the public health and safety concerns of such State in regard to such a project and shall give consideration to such concerns and cooperate with such officials in resolving such concerns." This model for involvement of a State in the development of a key national interest project and in the resolution of related public health and safety issues, was signed into law (PL 96-164) by President Carter on December 29, 1979. Although this model related to a proposed unlicensed facility for defense wastes, it sets forth an approach which could be useful in the siting and licensing of facilities for commercial wastes.

The Congress has also addressed, in other contexts, the important issue of the extent to which Federal activities affecting local communities should be provided special treatment, eg, support for the local infrastructure transition to the modest level of development that may be required for a waste disposal or storage site. One mechanism adopted by Congress has been payment in lieu of taxes to deal with developments at sites of national laboratories (eg, Hanford, WA; Oak Ridge, TN; Los Alamos, NM). Recently, Congress has considered expanding this concept to provide for assistance to communities potentially impacted by the establishment of necessary large energy facilities.[13] Thus, there are precedents that may be looked to in addressing potential local impact issues as sites for geologic repositories are considered.

Although Federal/State relationships in radioactive waste management are slowly evolving through administrative action, there can be additional confidence that if problems are not resolved promptly Congressional remedies will be available.

4) <u>State and Local Participation</u>

The active and effective involvement of State and local governments has been and will continue to be an important element of implementation of the Federal waste management policy. The President reaffirmed this keystone of Federal policy in his February 1980[14] statement:

> "My Administration is committed to providing an effective role for State and local governments for the development and implementation of our nuclear waste management program."

A concrete step taken by the President to formalize the concept of Federal/State cooperation was the creation by Executive Order of the State Planning Council designed to "strengthen our intergovernmental relationships and help fulfill our joint responsibility to protect public health and safety in radioactive waste management." Also, a companion commitment was made to provide financial and technical assistance to State and other jurisdictions to facilitate their full participation in review and licensing proceedings. The membership of the State Planning Council represents an important cross-section of the State and local governmental authorities involved in the activities of the current plan as proposed by the President. Although the State Planning Council cannot be expected to supplant the direct participation of affected States on site-specific issues, it assures additional opportunity for collective State input in the development and implementation of the overall waste management plan.

DOE has made substantial progress in cooperating with State and local authorities and encouraging them to participate directly in the process of planning and preliminary site selection activities. The consultation process already has been initiated in New Mexico, Utah, Louisiana, Mississippi, Nevada, Texas, Michigan, and Washington. The DOE's Statement of Position describes actions taken and planned to achieve State and local cooperation in the national screening phase, regional studies, area studies, and site characterizations.

These DOE activities involving specific State and local authorities are, of course, in addition to DOE's efforts to foster broad participation by State and local officials in the development of the Federal waste management programs and policies. Thus, for example, in addition to the usual solicitation of public comments on the DEIS, the DOE convened a Hearing Board, which held hearings in five major cities throughout the United States in order to facilitate and encourage the presentation of views by the public and State and local officials. As another example, in late 1979 DOE convened a workshop[15] of Federal, State, and local officials and members of the public to examine a wide spectrum of questions relating to Federal/State relationships.

It is clear that the Administration has recognized the importance of early and comprehensive State and local involvement in the planning and implementation of Federal waste management programs. The attention being given to these matters by the Administration will provide a basis for resolving remaining questions either through specific cooperative actions with affected States and municipalities or through legislation which takes into account Federal, State, and local interests.

C CONCLUSIONS

Even if the nation never generated another kilowatt-hour of electricity with nuclear power, there would still be the same compelling need to implement a nuclear waste disposal program. Nuclear wastes are already in existence from the nuclear weapons program, the military nuclear reactor programs, and Federal research and development programs. The wastes in spent nuclear fuel have been in existence since the first domestic application of nuclear technology for commercial energy production in 1957. Naval nuclear reactors, research reactors, and military weapons facilities will continue to generate nuclear wastes in programs which contribute in large measure to our national security. Implementation of a nuclear waste management program is not a matter of choice, it is a matter of necessity.

Moreover, since continued utilization of commercial nuclear energy contributes to our national security, our economic well-being, and our protection of the environment, there is every reason to expect that institutional issues relating to waste management will be resolved on a timely basis to avoid adverse impacts on these national interests. While the institutional aspects of the nuclear waste management program are complex, considerable progress has been made by involved institutions to resolve potential obstacles to the timely and successful siting, licensing, and development of geologic repositories. The Administration is committed to achieving coordination of activities of Federal agencies and to providing an effective role for State and local government in the development and implementation of the national program. While the pace has been slow, progress is being made in resolving institutional questions. DOE has been given the lead role in management and coordination of the waste management program; EPA and NRC have made progress in developing standards and establishing a regulatory program. The President has established a State Planning Council for formal input from the States; and DOE has taken numerous actions to foster and encourage cooperation with State and local officials.

Congress has fully funded all aspects of the national program for the management of defense and commercial nuclear wastes, has undertaken a number of important initiatives to direct and support the development of a comprehensive national waste management program, and has taken actions with respect to similar or comparable projects or programs which may be useful precedents in resolving institutional problems. All of these activities demonstrate that Congress will be a constructive force in fostering and assuring the timely resolution of any institutional problem, including the development of a legislative solution if that should prove to be the most effective mechanism for establishing appropriate Federal/State relationships.

We believe that DOE's present programs are focusing upon all matters that need to be addressed, and the DOE's forecast of potential operational dates for the first geologic repositories includes more than adequate allowance for uncertainties, including institutional considerations. We are convinced that the national interest would be best served by an accelerated program, and that waste management decisions made during the next several years will recognize the national benefits and imperatives in avoiding delay and achieving prompt implementation. Thus DOE's projections of operation commencing in the 1997-2006 period are conservative, and an operating date prior to 1997 is fully achievable. For purposes of this proceeding, however, the precise schedule is not of critical importance; the fact that technical capability exists to proceed at whatever pace is selected by Federal decisionmakers is sufficient for the Commission to reach affirmative conclusions.

The overriding national interest requires that any institutional problems associated with nuclear waste management be resolved effectively and promptly. Confidence that this can be done is supported by over 200 years of experience, during which time Federal and State officials have dealt with problems which were orders of magnitude more significant. There is simply no reason to believe that our institutions and their processes will fail to deal successfully with nuclear wastes.

REFERENCES FOR SECTION IV

1. Office of Nuclear Waste Isolation, Battelle Memorial Institute, Fiscal Year 1982 Budget (ONWI, March 1980).

2. U.S. Department of Energy, Hearing Board Report on the Draft Environmental Impact Statement on the Management of Commercially Generated Radioactive Waste, DOE/EIS-0046-D (February 1980).

3. Statement by the President Announcing Actions in Response to the Report of the President's Commission on the Accident at Three Mile Island, 15 Weekly Comp. of Pres. Doc. 2202 (December 7, 1979).

4. Energy Information Administration, U.S. Department of Energy, Report to Congress (DOE/EIA-0008/4(79), April 1980).

5. President's Message to Congress on Comprehensive Radioactive Waste Management Program, 16 Weekly Comp. of Pres. Doc. 303 (February 12, 1980).

6. Department of Energy Organization Act of 1977, 203(a) (8), 42 U.S.C. 7133 (1977).

7. Fact Sheet: The President's Program on Radioactive Waste Management, 16 Weekly Comp. of Pres. Doc. 296 (February 12, 1980).

8. Public Works for Water and Power Development and Energy Research Appropriation Bill of 1979, H.R. 12928, 95th Cong., 2nd Sess. (1978).

9. Department of Energy Civilian Programs 1980 Authorization Act, H.R. 3000, 96th Cong., 1st Sess. 107.B.4 (1979).

10. House Comm. on Science and Technology, Report to Accompany H.R. 6627, Department of energy Civilian Programs 1981 Authorization Act (107.B), H.R. Rep. 96-967 Part III, 96th Cong., 2nd Sess. (1980).

11. Senate Comm. on Energy and Natural Resources, Report to accompany S. 2189, Nuclear Waste Policy Act, S. Rep. 96-548, 96th Cong., 2nd Sess. (1980).

12. Comm. of Conference, Report to accompany S. 673, Department of Energy National Security and Military Applications of Nuclear Energy Authorization Act of 1980, Conf. Rep. 96-702, 96th Cong., 1st Sess. (1979).

13. Energy Impact Assistance Act of 1979, S. 1699, 96th Cong., 1st Sess. (1979).

REFERENCES FOR SECTION IV (CONT.)

14. Op cit, President (Ref. 3).

15. Office of Nuclear Waste Isolation, Battelle Memorial Institute, Consultation and Concurrence Workshop Proceedings, September 23-26, 1979 (ONWI-87, January 1980).

V COST OF DISPOSAL

In addition to establishing confidence in the technological base for the disposal of nuclear wastes, it is also important to determine that the costs of such disposal are not so great as to make implementation of the system prohibitively expensive.

An independent review of the estimated costs of proposed waste disposal technology was presented in two recent comprehensive studies on the subject, one by DOE and one by ONWI.[1,2] No attempt was made to do an in-depth review of these cost estimates, but sufficient review and testing of the estimates was done to give us confidence that both the costs themselves and the conceptual technology upon which they are based are adequate and comprehensive.

Both referenced studies presented their results in terms of "levelized unit costs" expressed as 1978 $/MTHM. For our purposes, we have rearranged and recomputed these costs, updating them to January 1979 dollars and expressing them in terms of mills per kWh.*

Table V-1 gives the estimated costs (from the DOE study) for disposal of spent fuel in various geologic media using accelerated mining.** The associated capital cost estimates are given in table V-2.

Table V-3 gives the results from the ONWI study, again in terms of $/kg HM and mill/kWh. In this case the reference values were already in January 1979 dollars. It can be seen that this estimate results in somewhat lower costs than those cited previously.

It should be noted that costs shown refer only to the repositories; they do not include the costs of the other elements of the total waste handling

---

* The conversion factor used was $1/kg HM = 0.0042 mill/kWh.

** Accelerated mining is defined as mining out the entire volume at the beginning as opposed to mining only a required volume each year.

Table V-1

Estimate of Levelized Unit Cost for

Spent Fuel Repositories, Accelerated Mining, $/kg HM (1979)[a]

Geologic Media	Waste Type	Total Levelized Unit Cost[a]	Levelized Cost Mill/kWh
Salt	PWR	51.10	0.213
	BWR	57.50	0.240
	average	53.60	0.224
Granite	PWR	77.40	0.354
	BWR	84.80	0.336
	average	80.50	0.336
Shale	PWR	52.50	0.219
	BWR	69.00	0.288
	average	58.80	0.245
Basalt	PWR	87.20	0.364
	BWR	94.10	0.393
	average	89.90	0.375

a "Technology for Commercial Radioactive Waste Management," DOE/ET-0028, US Department of Energy, May 1979.

b
1) Assumes a cost of money of 7 percent to the Federal Government.
2) Overall uncertainties for all total levelized unit costs are estimated to be ± 50 percent.
3) Includes construction and mining cost, operating cost, and backfilling and decommissioning costs.

Table V-2

Capital Cost Estimates[a]

Facility	Total Facility Cost (January 1979 Dollars)
Spent Fuel in Salt	1,068,000,000
Spent Fuel in Granite	2,714,000,000
Spent Fuel in Shale	1,370,000,000
Spent Fuel in Basalt	3,187,000,000
HLW in Salt	1,221,000,000
HLW in Granite	2,111,000,000
HLW in Shale	1,306,000,000
HLW in Basalt	2,381,000,000

[a] "Technology for Commercial Radioactive Waste Management," DOE/ET-0028, US Department of Energy, May 1979.

Table V-3

Capital and Operating Cost Estimates[a]

	Capital Cost $ millions	Total Levelized Unit Cost $/kg HM	Levelized Cost mill/kWh
Spent Fuel	950	19.4	0.081
HLW	940	32.2	0.134

a "Assessment of LWR Spent Fuel Disposal Options," ONWI-39, Bechtel National, July 1979.

cycle such as transportation, potential AFR storage, and processing and encapsulation (P/E). Table V-4 (based on the ONWI study) includes these additional segments and thus provides an estimate for the total cost of waste disposal.

These estimates suggest that the total cost of waste disposal of spent fuel will be about 1/2 mill/kWh, and the total cost of disposal of HLW will be less than 1 mill/kWh (even without credit for recycled materials). Clearly these costs represent a minor contribution to power production costs and accordingly do not present any major obstacle to implementation of a disposal system.

Table V-4

Total Cost of Waste Disposal[a]

	Capital Cost $ millions	Total Levelized Unit Cost, $kgHM	Levelized Cost mill/kWh
Spent Fuel			
Transportation	Leases	26.9	0.112
AFR	2,160	53.3	0.222
P/E facility	1,017	16.7	0.070
Repositories (3)	2,850	19.4	0.081
Total	6,027	116.3	0.485
HLW[b]			
Transportation	Leases	29.7	0.124
P/E facility	4,230	124.0	0.517
Repository	940	32.0	0.134
Total	5,170	185.9	0.775

a "Assessment of LWR Spent Fuel Disposal Options," ONWI-39, Bechtel National, July 1979.

b Does not reflect any credit for recycled materials.

REFERENCES FOR SECTION V

1. U.S. Department of Energy, Technology for Commercial Radioactive Waste Management (DOE/ET-0028, May 1979).

2. Office of Nuclear Waste Isolation, Battelle Memorial Institute, Assessment of LWR Spent Fuel Disposal Options (ONWI-39, July 1979).

VI CONCLUSIONS ON STATUS AND AVAILABILITY OF DISPOSAL SYSTEM TECHNOLOGY

The previous sections of this report and the accompanying companion document ("Long-term Safety of Nuclear Waste Disposal: A Basis for Confidence") have described and discussed in considerable detail several basic aspects of the fundamental questions being addressed by this proceeding, namely

- What is it that must be accomplished by management of nuclear wastes?
- What alternative approaches or systems are available for nuclear waste management?
- What are the technological components and issues associated with nuclear waste management?

Our examination of the status of technology has been based on the requirement that an acceptable nuclear waste disposal system must protect the public now and in the future from undue risk of excessive exposures to radiation. Exposure of populations to radiation equal to some fraction of the variations in natural background radiation would appear to be a reasonable, rational general criterion for the degree of radiation protection provided by a suitable waste disposal system.

The time period of major interest and concern for containment of highly radioactive wastes is about 500 years. Beyond these times the potential for exposure of populations from these wastes via the water-borne (most likely) pathway is comparable to or less than that from the naturally occurring uranium ore from which the waste was derived. This is due to the ability to select a location for a repository from particularly suitable sites deep underground, while uranium ore is found at random surface or near-surface locations. In addition, the repository system can include such engineered barriers as may be desired to enhance overall system performance.

In this section we present our conclusions derived from examination of the foregoing questions and recommend that they be incorporated in the overall of the Commission in this proceeding. These salient points are sum , fashion as follows:

sed on conservative, detailed examination and evaluation of all the gnificant system components, we conclude that there is a high degree f confidence that the desired protection can be provided by a system or the disposal of spent fuel or solidified HLW in appropriately located, designed, constructed, and operated repositories in suitable deep geologic formations. Other alternatives may also provide such protection but are not as fully developed. Disposal in subseabed may be a "second generation" system variation of the deep geologic repository. The possibility that something "better" may become available in the future should not delay going forward with the acceptable repository systems now available.

- Technology exists, and in some instances has been demonstrated by operational experience, for practicable, conservative implementation and application of all components of an overall geologic repository system.

- While site-specific data and information are essential in order to provide appropriate bases for specific repository engineering design, construction, and operation, the current incompleteness of such data and information does not constitute "gaps" in our scientific or technologic knowledge. No scientific or technologic "breakthrough" is required for the proper use and application of the data and information as it is acquired.

- Where so-called "uncertainties" are perceived to exist (mostly in the areas of waste/host rock interaction and radionuclide migration), they are readily amenable to conservative bounding and can be compensated for by a wide range of operational approaches and engineered barriers. While various repository safety analyses indicate that geologic barriers can provide the necessary containment of wastes, appropriate combinations of natural and engineered barriers can provide further assurance of desired performance by the overall waste management system, particularly during the period of higher relative risk.

- Even though additional development may be required for some of these engineered barriers, they represent straightforward extensions of existing technologies and are based on passive application of known physical and/or chemical principles.
- Reliance on the performance of the overall repository system is based on the collective interactive performance of its components (waste form, repository site and design, geologic environment, etc). It is emphasized that engineering flexibiity and redundancy can be applied as further risk-mitigating options, if necessary; however, such applications should be based on rational cost/benefit considerations.
- The DOE schedule for program implementation, based on a conservative multigeologic media investigation strategy and an extensive licensing process, realistically can be met and is amenable to acceleration. There is no reason why a repository cannot be located, designed, and constructed on schedule and at practicable cost.
- There are no institutional barriers to implementation of the repository program that cannot be resolved in timely fashion. The Administration and the Congress have shown a commitment to resolution of waste issues and implementation of necessary programs to accomplish such resolution, and mechanisms have been established to provide for necessary involvement of the States.

The foregoing conclusions are supported by scientific and technical knowledge and information summarized below.

a) Site selection criteria are well developed, and methodology exists for determining the geologic, seismologic, and hydrologic characteristics of sites and their surroundings. Areas have been found where most natural geological processes operate at rates such that containment in a mined repository would not be significantly reduced over periods of at least several hundred thousand years, and where rates of groundwater movement and lengths of flowpaths to the biosphere are suitable to contain radionuclides for periods of thousands to millions of years. Continuing development and improvement of predictive models and continuing in situ monitoring during the operational phase of the repository will provide an even higher level of confidence in the ability of the repository to contain radionuclides for the necessary length of time.

b) Spent fuel represents a significant containment barrier in the form of a very low leach rate (Barrier #2) should water ever reach the waste. The high degree of containment of fuel elements themselves has been conclusively demonstrated by their use in the reactor, albeit in an essentially pure water environment rather than groundwater. The leach resistence of the fuel elements could, if deemed necessary, be enhanced by the use of a metal matrix or other stabilizers. In any event, the waste form will provide a significant barrier which is expected to last over very long time periods.

c) Additional engineered barriers to the migration of radionuclides can be provided, if necessary, by a number of mechanisms such as the addition of selected materials which can absorb or otherwise retard the movement of radionuclides from the package and the repository.

d) Currently available technology is adequate to proceed with the design and construction of a repository, using appropriate engineering conservatism. Rock properites can be adequately characterized by a combination of laboratory and in situ testing. Induced stresses due to excavation can be predicted and controlled by excavation methods and the design of rooms and corridors. Thermomechanical models are available, and comparison with operating experience in Project Salt Vault indicates that they are adequate. Furthermore, the thermal source term can be controlled by specifying the age of the fuel and canister spacing. Retrievability has been demonstrated in a field test over limited time periods and should not be a problem.

e) Technology for handling and emplacement of waste canisters in a repository is well developed and has been demonstrated in Project Salt Vault.

f) Techniques and materials for backfilling, sealing, and closure operations are under development. Based on experience in the petroleum, petrochemical, and mining industries, there is little doubt that effective materials for shaft and borehole sealing can be developed.

g) Models are available to predict the long-term behavior of repositories. Analyses of accident scenarios using these models in a very conservative way predict human exposure levels below natural variations in background, except for drill crew workers in the highly unlikely event of direct penetration of a canister.

In summary, a conclusion by the Commission is fully justified that the technology and technical capability and tools exist to identify a suitable site for a deep geologic repository and to design, construct, and operate an overall repository system for d sposal of wastes in the form of spent fuel in a manner that will be safe to current and future generations. A conservative, step-wise approach to system implementation and the availability of engineered barriers provide a practical basis for adequately coping with perceived "gaps" or "uncertainties". No scientific or technological "breakthroughs" are needed for implementation of the system.