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DETERMINATION OF A RADIOACTIVE WASTE CLASSIFICATION SYSTEM

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FOREWORD

The U.S. Nuclear Regulatory Commission (NRC) has retained the University of California Lawrence Livermore Laboratory (LLL) to provide technical support for its Nuclear Waste Management Program. Development of a classification system for radioactive waste is part of this program.

This LLL project:

- Determined objectives for the required waste classification (WC) system.
- Identified and evaluated relevant system parameters.
- Developed a suitable format (classes of waste) for the system.
- Determined the interfaces between classes.

Work was done in two phases, largely with the assistance of subcontractors. Phase 1--through development of a system format--was done with help from Science Applications, Inc. (SAI), McLean, Virginia, from June through December 1976. Ford, Bacon & Davis Utah, Inc., of Salt Lake City, Utah, helped with phase 2 from March through October 1977.

Technical advisory panels (TAP's) of experts from industry, government, and research institutions provided guidance during each phase. Members of the first TAP (Appendix A) met in Reston, Virginia, in August 1976, after which a revised working document was prepared. TAP members critiqued the document at the second meeting in La Jolla, Calif., in October 1976. LLL published a interim report[1] based on the critiqued document.

This report summarizes the project's work and presents our rationale and conclusions.

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ABBREVIATIONS

AEC	(U.S.) Atomic Energy Commission
AIChE	American Institute of Chemical Engineers
ALAP	As low as possible
ALARA	As low as reasonably achievable
AMAD	Activity median aerodynamic diameter
ANSI	American National Standards Institute
CFR	Code of Federal Regulations
DOE	(U.S.) Department of Energy
EPA	Environmental Protection Agency
FP	Fission products
GI	Gastrointestinal
HLW	High-level waste
IAEA	International Atomic Energy Agency
ICRP	International Commission on Radiological Protection
INEL	Idaho National Engineering Laboratory
LET	Linear energy transfer
LLL	Lawrence Livermore Laboratory
LLW	Low-level waste
LWR	Light water reactor
MPC_a	Maximum permissible concentration in air
MPC_w	Maximum permissible concentration in water
MPE	Maximum permissible exposure
MPQI	Maximum permissible quarterly intake
NCRP	National Commission on Radiation Protection
NEPA	National Environmental Protection Act
NRC	(U.S.) Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
RCF	Reference containment facility
RHI	Relative hazard index

TAP	Technical advisory panel
TRU	Transuranic nuclides
WC	Waste classification
WM	Waste management

ABSTRACT

Several classification systems for radioactive wastes are reviewed and a system is developed that provides guidance on disposition of the waste. The system has three classes: high-level waste (HLW), which requires complete isolation from the biosphere for extended time periods; low-level waste (LLW), which requires containment for shorter periods; and innocuous waste (essentially nonradioactive), which may be disposed of by conventional means. The LLW/innocuous waste interface was not defined in this study. Reasonably conservative analytical scenarios were used to calculate that HLW/LLW interface level which would ensure compliance with the radiological exposure guidelines of 0.5 rem/y maximum exposure for a few isolated individuals and 0.005 rem/y for large population groups. The recommended HLW/LLW interface level for ^{239}Pu or mixed transuranic waste is 1.0 $\mu Ci/cm^3$ of waste. Levels for other radionuclides are based upon a risk equivalent to this level. A cost-benefit analysis in accordance with as low as reasonably achievable (ALARA) and National Environmental Protection Act (NEPA) guidance indicates that further reduction of this HLW/LLL interface level would entail marginal costs greater than $\$10^8$ per man-rem of dose avoided. The environmental effects considered were limited to those involving human exposure to radioactivity.

SUMMARY

This is the final report for phase 2 of the Lawrence Livermore Laboratory (LLL) project to develop a radioactive waste classification (WC) system for the Nuclear Regulatory Commission. We reviewed existing WC systems and developed a three-class system for solid waste based on ultimate disposition:

- Innocuous waste that may be handled as normal trash because of its very low radioactive levels.
- Low-level waste (LLW) requiring active confinement (confinement or holdup with controlled or predictably low release rates). This waste class would include materials that, due to their low specific hazard levels and/or short decay times, may be adequately controlled in a suitably designed and operated containment facility.
- High-level waste (HLW) requiring isolation (complete containment with no expected release to the biosphere for extended periods of time). This class of waste will contain radioactive materials of very high hazard potential and/or long decay times.

Figure S1 is a schematic of the proposed WC system. No value was set for the innocuous waste/LLW interface because further study is needed to provide a precise and defensible value.

The key to setting the HLW/LLW interface value was to define LLW concentration limits such that guidelines for radiation exposure to the public are not exceeded. To this end, we:

- Defined reasonably conservative exposure guidelines (0.5 rem/y max exposure for a few individuals and 0.005 rem/y for a large group).
- Reviewed Department of Energy (DOE) LLW disposal facilities.
- Established a model reference containment facility (RCF) for LLW.
- Identified conservative exposure scenarios.
- Determined source terms and release fractions from the RCF.

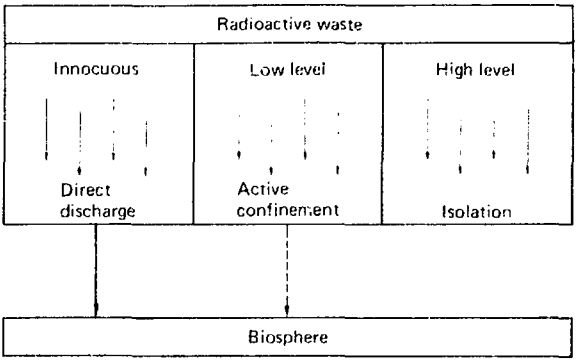

FIG. S1. Proposed waste classification system.

- Described the transport of the radioactivity through the environment to man.
- Calculated the maximum individual dose and the population dose.
- Related these calculated doses to the radiation exposure guidelines.

After these steps were defined, the dose to the exposed population could be related to the concentration of radioactivity in the waste at the time of burial. Figure S2 shows our results for the analytical scenarios. We recommend a HLW/LLW interface level for ^{239}Pu or mixed transuranic waste of 1.0 $\mu Ci/cm^3$. Levels for other radionuclides are calculated for a risk equivalent to this level. A cost-benefit analysis in accordance with ALARA and NEPA guidance indicates that the cost-effectiveness of reducing the HLW/LLW interface level further would be greater than $\$10^8$ per man-rem averted. Only those environmental effects related to human exposure to radioactivity were considered.

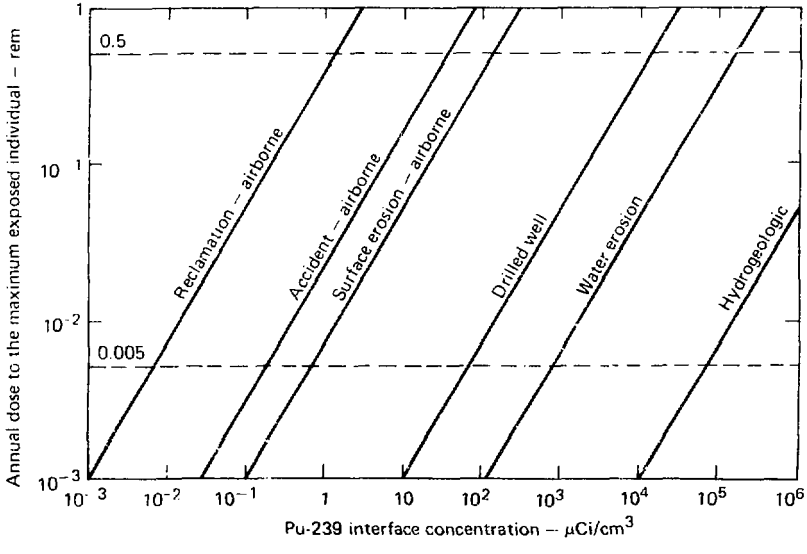

FIG. S2. Annual dose vs ^{239}Pu HLW/LLW interface level for six analytical scenarios.

INTRODUCTION

The U.S. Nuclear Regulatory Commission (NRC) has retained the University of California Lawrence Livermore Laboratory (LLL) to provide technical support for its Nuclear Waste Management Program. A part of this program involves the development of a classification system for radioactive waste. Such a system should be applicable to all sources of radioactive waste, especially the nuclear fuel cycle, which is the predominant source. Projections for light water reactor (LWR) waste production (Fig. 1 and Tables 1 and 2)[2,3] provide an idea of the extent of the waste management problem.

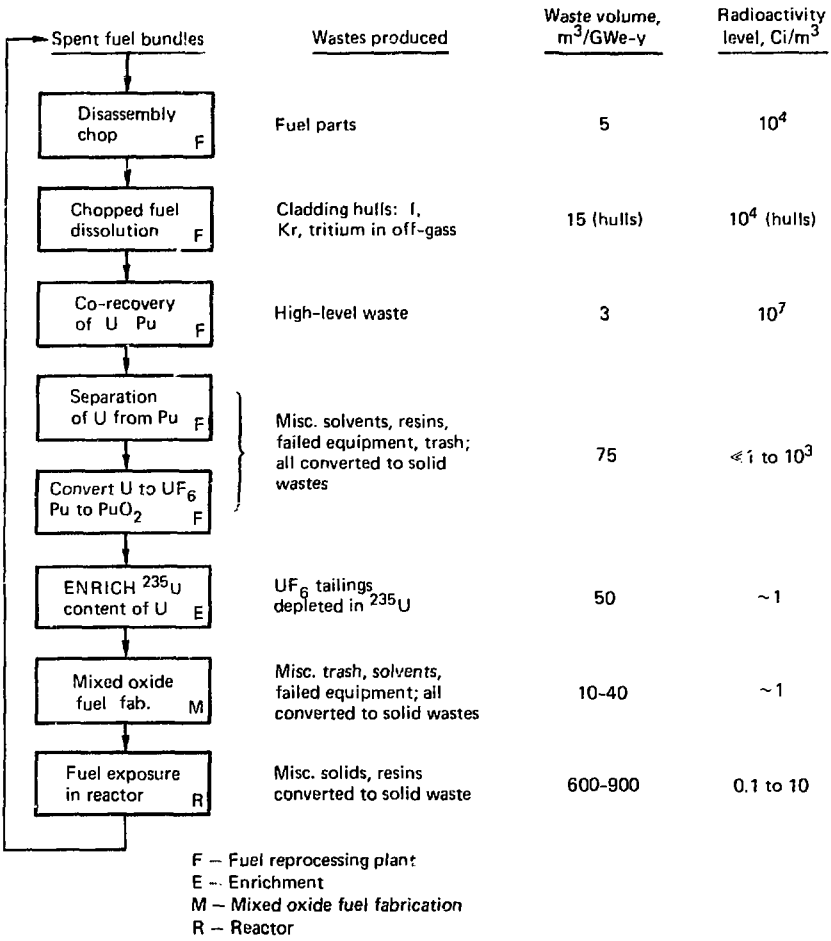

FIG. 1. Process operations and wastes in the LWR fuel cycle.[2]

TABLE 1. Fuel cycle wastes from generation of 1000 MWe-y by LWR's using mixed-oxide (U-Pu) fuels.

Waste type	Volume, m^3	Activity, MCi	Mass of actinides, tons	Thermal power, kW
High-level solidified	3.0	81.5	0.25	450
Cladding hulls	2.6	0.8	0.017	3.3
Noble gases	1.0	0.24	-	0.36
Iodine	0.049	1.25×10^{-6}	-	7.5×10^{-7}
LWR tritium (water)	140	7.4×10^{-4}	-	2.3×10^{-5}
FP tritium (solidified)	0.34	0.018	-	0.00062
Carbon-14	-	1.66×10^{-5}	-	-
Low-level TRU	46	0.047	0.004	0.04
Intermediate-level TRU	13.4	0.11	0.0012	0.054
Nontransuranic	600	0.002	-	0.0065

Source: Ref. 3

Note: Age of waste is assumed to be approximately 1 y.

TABLE 2. Postfission wastes per GWe-y expected from the LWR fuel cycle.

Waste form	Volume[a], m^3 (except as noted)	Radioactivity, Ci
	Reactor wastes	
Slurries	2E2[b]	2E1[c]
Sludges	3E1	3E1[c]
Resins	6E1	4E3[c]
HEPA filters	5E0	1E1[c]
Charcoal	2E0	1E0[c]
Trash	2E2	5E0[c]
Failed equipment	6E0	2E5[c]
Water filters	4E0	3E2
	Mixed-oxide fuel fabrication wastes	
Combustible solids	6E1	9E2
Noncombustible solids	1E1	2E2
Liquids, slurries	2E0	3E2
HEPA filters	4E0	3E2

Source: V. Trevorrow, Argonne National Laboratory, Argonne, IL, private communication (1976).

[a]Volumes (except for high-level fuel reprocessing wastes) represent those of primary waste forms, before volume-reducing treatments.

[b]1E1 to be read as 1×10^{1}; 1E-1 to be read as 1×10^{-1}, etc.

[c]Radioactivities of reactor wastes are assumed to be based on surveys at shipping time, about 6 mo after removal from reactor.

TABLE 2. (cont'd.)

Fuel Reprocessing Wastes				
Waste form	Volume, m^3 (except as noted)	Radioactivity, Ci	Radioactivity, neutrons/sec	Thermal power, kW
Hulls	1E1	3E5[d]	6E7[d]	2E0[d]
High-level solid	4E0	1E7[e]	1E11[e]	6E1[e]
Low-level liquid	1E6	2E4[f]		
Gases				
Krypton	3E0[g]	3E5		
Xenon	3E1[g]	6E-4		
Iodine	9E3 gr	1E0		
Carbon-14	4E0 gr	2E1		
Tritium	2E0 gr	2E4		
Non-high-level solids				
Combustible TRU	5E1			
Combustible nonTRU	3E1			
Noncombustible TRU	5E1			
Noncombustible nonTRU	1E2			
Slurries, sludges, resins TRU	5E-1			
Slurries, sludges, resins nonTRU	5E0			
Filters TRU	7E0			
Filters nonTRU	5E-1			

[d]Based on 5-y cooling after discharge.

[e]Based on 10-y cooling after discharge.

[f]Based on assumption that 70% of tritium in spent fuel will appear in low-level aqueous waste.

[g]Gaseous volume at STP.

The LLL project was conducted in two phases. The first phase:

- Established the purpose and objectives for the WC system
- Identified and evaluated relevant system parameters
- Developed a suitable format (classes of waste) for the system.

The second phase mostly determined interface values between the classes of waste.

Phase 1 ended with publication of a working document[1] covering:

- Sources of wastes containing long-lived radioactive materials resulting from fuel-cycle activities.
- Data on waste volumes, characteristics, and current status of waste management operations.
- Suggested alternative approaches and methods for waste classification.
- Evaluation of the identified approaches and discussion of their advantages and disadvantages.
- Selection of an approach and format.

Technical advisory panels (TAP's) of experts from industry, government organizations, and research laboratories assisted the LLL project. Early in its tenure, the first TAP helped to establish four basic objectives and requirements for an acceptable WC system. In order of priority they are:

1. The protection of public health and safety for present and future generations. It is generally recognized that complete and absolute protection (i.e., zero risk) for every individual living now or in the future is impossible. Society will accept some risks if they are low enough and if the benefits to be derived are seen as outweighing the risks. A risk vs benefit evaluation should therefore be part of the foundation for the WC system. This objective is consistent with ALARA guidance and with the intent of the NEPA.
2. The system should have a sound technical basis and should allow for societal, environmental, or technical issues to be resolved within its framework. The system should, however, be based primarily on technical considerations.

3. The system should be consistent with good economic practice. A system that would impose undue economic penalties is unacceptable. Evaluating the operating costs of the system in conjunction with a benefit/risk analysis should provide a sound economic basis for the proposed classification system.
4. The system should be practical for all industries producing radioactive wastes. Generally, the simpler the system, the more practical it becomes to the user.

Since protection of public health and safety is the paramount consideration in the formulation of a WC system, it is important to specify the areas of public health and safety that are of major significance. In this regard, avoidance of undue exposure to radioactivity becomes the primary goal, with other potentially adverse health and environmental effects being of secondary consideration.

As a guide to acceptable radiation exposure, current radiation protection criteria established by the International Commission on Radiological Protection (ICRP),[4] the National Council on Radiation Protection and Measurements (NCRP),[5] and the Code of Federal Regulations (10CFR20)[6] can specify limits for potential individual exposures. Population exposure (man-rem) limits can be made to be consistent with ALARA guidance.[7]

In either case, one must make certain assumptions and develop suitable analytical models to see that the criteria are met. It is important, also, that all assumptions be clearly stated so that the conclusions may be rationally evaluated.

This report is the final document of the project. It describes the work done, recommends a WC system, and discusses the technical basis for the recommendations.

DEVELOPMENT OF A CLASSIFICATION FORMAT

Radioactive waste classification (WC) systems generally fall into one of three categories based on the source of the waste, the physical characteristics of the waste, or the waste disposal methods. Factors that could be considered for waste classification include:

- The toxicity or hazard of the waste.
- The total quantity of waste (in curies, mass, or volume).
- The longevity of the waste (half-life).
- Potential biological concentration mechanisms for various types of radioactivity.
- Consideration of whether the waste was natural or manmade.
- Heat production.
- Potential economic value.
- Physical state (solid, liquid, gas).
- Consideration of whether the entire material is considered as waste or whether the waste is a form of contamination of some other valuable material.
- Type of radioactivity (α, β, γ, neutrons, etc.).
- Requirement for shielding.

In phase 1 of this study, several previously proposed WC systems were reviewed and evaluated in detail as discussed in Appendix C. In addition, certain concepts for WC systems that were prepared by TAP members during phase 1 are reviewed and discussed in Appendix D.

The advantages and disadvantages of the various WC systems were discussed in the TAP meetings in the context of NRC needs as well as needs of industry, government, and waste management operators. The insights from these reviews and discussions have led to certain conclusions and suggest a format for a WC system that might best serve the specified objectives.

ASSESSMENT OF WASTE CLASSIFICATION SYSTEMS

Classification by Source

Classified according to its source, radioactive waste could be reactor waste, spent nuclear fuel, reprocessing waste, or fuel fabrication waste. Reprocessing waste can be further identified as spent fuel cladding, liquid process waste, or general process trash. Because such a classification system reveals little about the nature of the waste or the means to be used for handling, processing, or disposal, it is not a useful system on which to base waste management regulations.

Classification by Physical Characteristics

Most reported classification systems are based on the physical characteristics of the radioactive waste. For example, the IAEA recommended in 1970 that radioactive wastes first be classified as solids, liquids, or gases. Liquid and gaseous wastes were then subdivided by activity levels in units of curies per milliliter or curies per cubic meter. Solid wastes were subdivided by surface radiation dose rate in units of rads per hour. The TAP members generally believed that a WC system should consider physical characteristics of the waste primarily as they affect its hazard potential.

Classification by Disposal Methods

Different ways to dispose of radioactive wastes have certain common features that lead to broad categories of waste disposal. These features include the degree of containment achievable and the degree of isolation or extent of social commitment required for each disposal option. The TAP members agreed that for regulatory purposes a WC system should be based on disposal methods.

The final disposal of radioactive wastes can be considered the most important operation in any radioactive waste management program. It is also the operation that requires the most definition. A WC system based on disposal options should provide information and guidance allowing maximum flexibility for implementing waste management programs by the waste generating facilities.

There are many possible waste disposal methods and a WC system based on method of disposal must be broad enough to include them all. Some suggested methods include:

- Direct discharge or dispersal to either the atmosphere or surface water.
- Storage until the radioactive isotopes have decayed to an innocuous level.
- Shallow land burial.
- Ocean dumping.
- Deep geological emplacement.
- Ice cap emplacement.
- Extraterrestrial disposal.
- Ocean disposal (seabed burial).
- Deep well injection.
- Shale fracturing.

These methods vary according to the degree of containment, isolation, and social commitment. For example, direct dispersal to the environment provides no confinement while deep geological emplacement should provide essentially complete containment and isolation for geological periods of time. Social commitment refers to such ongoing functions as record keeping, systems and security maintenance, and system and environmental monitoring. Management systems such as surface storage or shallow land burial require significant social and resource commitment for an extended period of time--long after the disposal facility is actually receiving radioactive material. By contrast, deep geological emplacement--after the operational phase is discontinued--would require minimal social commitment to the extent of preventing mining or drilling into the region surrounding the waste materials.

It is possible to formulate a WC system based on disposal methods without being limited to the means of disposal. All disposal systems fall into one of three categories:

1. Dispersal to the environment (nonradioactive waste)
    a. Stack discharges
    b. Liquid effluent discharges
    c. Disposal of solids to sanitary land fills
2. Disposal methods requiring social commitment (low-level waste)
    a. Shallow land burial
    b. Storage near surface
    c. Ocean dumping
    d. Deep well injection
3. Disposal methods providing long-term isolation (high-level waste)
    a. Deep geological emplacement
    b. Ice cap emplacement
    c. Extraterrestrial disposal
    d. Oceanbed disposal

The above disposal methods have been proposed or used in the past. They are not necessarily feasible or acceptable.

BASES FOR DEFINING WASTE CLASSES

Categorizing the various waste disposal methods helps in classifying radioactive waste materials because we can establish appropriate interface values between disposal categories. Figure 2 lists a number of considerations that may be important in deciding what type and amount of radioactive waste can be disposed of by the various methods. The list is meant to allow for the broadest possible approach by the regulatory agencies that must prepare waste management criteria and regulations.

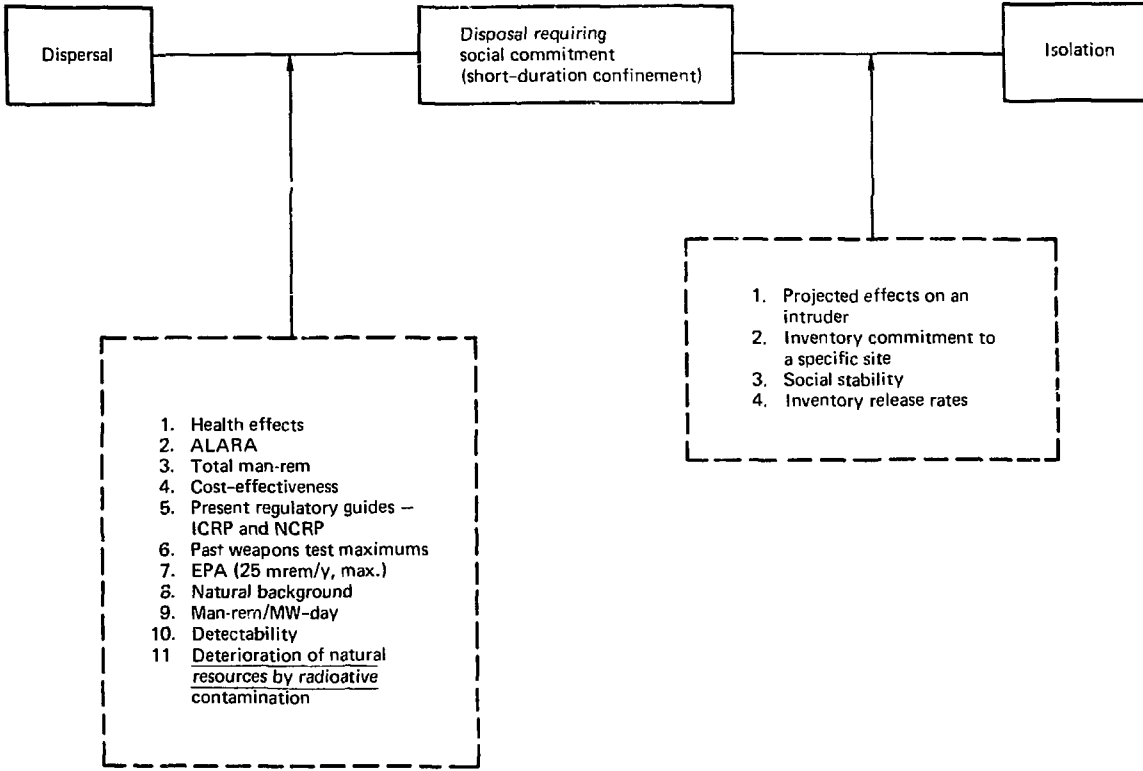

FIG. 2. Considerations for applying waste disposal methods.

Items 1 through 11 in Fig. 2 relate to the waste management criteria that would dictate whether a material could be dispersed or whether it would have to be contained. The TAP judged that the primary goal of any waste management program should be to limit the potential radiation dose to man. Accordingly, nine of these items deal with closely related aspects of the dose-to-man question.

Several different types of criteria could be established to limit the type and amount of radioactive material released to the environment by dispersion. For example, absolute values could be set for one or more of the following: total health effects; total man-rem; man-rem per unit of electric power generated; cost-effectiveness guidelines (for radiation protection); and maximum credible dose to any individudal or critical group. Any dispersal to the environment that exceeds a predetermined value would have to be captured or treated. The radioactive material resulting from such capture or treatment would than have to be disposed of by a method in one of the two waste containment categories.

A second means of formulating waste management criteria might be to use existing regulations and regulation guides. For example, present ICRP and NCRP standards could be used to establish dispersal limits. Also, the EPA's proposed individual dose limit of 25 mrem/y from nuclear fuel cycle operations, including waste disposal, could also serve as a guide to limit the quantity of radioactive material that could be dispersed to the environment.

Another approach is to establish an acceptable man-rem cost-effectiveness value. For example, 10CFR50, Appendix I,[7] sets a value of $1000/man-rem. Thus, any method of reducing population exposure that has a marginal cost of $1000/man-rem or less should be used. A different approach would lie in a regulatory philosophy that says the dose to man shall be ALARA. This philosophy is already implied by paragraph 20.1(c) of 10CFR20.[6] Such guidance, however, would have to be translated into practical criteria. The previously cited examples, in terms of dollars per man-rem or man-rem per MWe-y, may suffice. Some other approach might also be in order.

Of the first group of considerations listed in Fig. 2, two do not deal directly with dose to man: detectability and deterioration of natural resources. Some people believe strongly that no level of dispersal of radioactive material to the environment is acceptable. If zero release were to be adopted as a waste management criterion, the amount of material that could be released by dispersion would essentially be limited by our ability to detect radioactivity. A strict, literal interpretation of zero release would thus preclude any operations that involve radioactivity.

Deterioration of natural resources addresses itself mainly to ^{85}Kr releases, which contaminate natural krypton in the atmosphere. A waste management criterion might be based on acceptable levels of resource deterioration by treating certain substances or environments as potential resource bases.

A combination of two or more of the approaches discussed could provide for the development of acceptable regulations and waste management criteria. For example, an ALARA regulatory philosophy together with a cost-effective calculation and a maximum acceptable health effect or individual dose limit might serve to define what could be dispersed, what would require containment, and what must be isolated for long time periods.

CRITERIA FOR DEFINING WASTE CLASSES

Figure 2 also lists waste management criteria that might be used to determine the type and amount of radioactive material that can be disposed of by confinement as opposed to isolation.

Shallow land burial can be used as an example to discuss this second set of criteria. With regard to shallow land burial, the TAP generally reached the following conclusions:

1. Shallow land burial is a disposal method that requires ongoing social commitment.
2. During operation and for the first 100 y after decommissioning of a shallow land burial site, some low-level discharge to the environment can be expected.
3. The magnitude of the releases is a function of such factors as site location, area rainfall, geology, hydrology, engineering features, and operation procedures. Release rates are site specific and depend on these factors.
4. Surface geology can change significantly in time periods as short as 5,000 to 25,000 y. Social and institutional patterns can also be expected to change during these time periods. Consequently, near-surface storage or disposal systems should not be used for long-term isolation of radioactive material.
5. Some release of radioactive material to the environment from radioactive waste management programs is acceptable.
6. For a shallow land burial site, and with specific information on meteorology, geology, hydrology, engineering design, and other relevant factors, it is possible to estimate the amount of radioactive material that would be released from that site over a period of perhaps a few hundred years.
7. A conservative estimate of the duration of social commitment required for a disposal site is no more than a few hundred years.

Similar statements can be made about storage facilities, ocean dumping, or deep well injection. They include:

- Degree of containment is procedure and site specific.
- Long-term containment (>1000 y) cannot be guaranteed.
- Releases can be estimated for various time frames.

The TAP members reviewed the possible fate of radioactive material disposed of by the shallow land burial method and recommended that two time frames be considered--the operational phase of the burial ground and the first few hundred years after decommissioning, and the subsequent period.

During the first time frame, release of radioactive material to the environment depends on specific site characteristics, engineering features, and operational procedures. The TAP members generally thought that enough data could be obtained to predict the release of radioactivity from a given site, or at least specify a range of values that would characterize this release. They believed that the release rate (fractional release) can be assumed to be constant for each site, whereas the total quantity released is a function of the site inventory. Accidental intrusion during the first few hundred years was not considered a problem because of the availability of site records and security systems.

For the first few hundred years, therefore, total site inventory is the factor that should be controlled and regulated. Our previous discussion of waste management criteria for dispersal could also apply to determining acceptable release from a shallow land burial site.

It was assumed that no records of the site would exist and no security system would be operating after a few hundred years. The site could become an agricultural area or part of a potable water supply system. Therefore, the inventory of radioactive material remaining at the site is important. A detailed analysis of the possible pathways to man from such a site must be done, and a set of waste management criteria must be formulated to define what an acceptable dose to man might be at some time in the future. The criteria

for an acceptable dose to man in the future may or may not be the same as those applied to the present population.

Without site records and security, such intrusion as drilling, mining, or surface excavation may occur. To prevent injury to anyone disturbing the site after a few hundred years, the concentration of radioactive material in any one waste package may have to be limited.

As indicated earlier, numerical values for any limit on waste inventory or concentration would preferably be method and site specific. It is important to note that total curies of activity would probably not be the units chosen to define either the inventory or concentration limits. A hazard index should be used instead. Inventory limits could be based on the volume of water required to dilute the total site inventory to maximum permissible concentration in water (MPC_w) limits. The reason for using MPC_w as a basis is that the inventory limits are intended to restrict the quantity of radioactive material slowly released from a given site. Since the dose to man from the material released might largely result from water in the food-chain pathway, MPC_w is only an indication and not a direct measure of the potential risk.

Another consideration in setting absolute inventory limits is that radioactive materials decay. Hence, the total hazard index for a given site is always changing. During the first time frame, corrective steps can be taken if higher-than-expected releases occur, while during the second time frame, release is assumed to be undetected. Consequently, the inventory limits for the first time period may be several times greater than those for the second. Large quantities of short-lived radioactive isotopes could be disposed of by methods requiring long-term social commitment, but the amounts of longer-lived radioactive isotopes might be limited. Detailed analysis of the specific disposal method and disposal site is required to determine which of the three limits above would be controlling.

TRANSURANIC WASTES

The question of whether wastes containing transuranic (TRU) nuclides should be given special consideration, as has been done in the past, was discussed as a separate issue. The TRU properties used to justify placing them in a separate category are:

- Extremely long half-lives.
- High toxicity.
- Emission of alpha rays having a high linear energy transfer.
- Manmade origin.
- Potential for criticality accidents in sufficiently high quantities and concentrations.
- Detectability of extremely small quantities.

Assessment indicates that, except for the criticality potential, none of the identified TRU properties is unique. In the context of radioactive waste, where the TRU would be mixed with large quantities of diluent material, the possibilities of criticality are insignificant. Where the TRU concentration would be high enough to pose a criticality problem, the material could be viewed as a valuable resource rather than a waste. Although the TRU isotopes are highly toxic because of their high specific radiotoxicity, they are not the most toxic material known to man and should be viewed in proper perspective. Previous studies indicate no technological basis for distinctive treatment of TRU as opposed to other radioactive materials.[8,9]

PROPOSED FORMAT AND GUIDELINES FOR A WC SYSTEM

The waste classification system should be based on the ultimate disposition of the waste material. Although a classification system is definitely needed for guidance in disposal operations and in the handling of radioactive wastes, the need is primarily to give direction on the disposition of the wastes. Waste generating facilities could be designed to generate the waste in the optimum form for safe and economical disposal.

For present purposes, it may be assumed that all waste handling operations will comply with existing standards, regulations, and rules of good practice (i.e., shielding, heat dissipation, etc., will be provided where required). The main object of the present work is to provide guidance to the waste producer regarding the final disposition of the waste.

This does not preclude the need for an operational WC system to provide guidance on waste handling operations. It was noted, however, that optimally designed operational WC systems might best be designed individually to meet the specific needs of various waste handling facilities.

An adequate and acceptable WC system can best be defined by three waste classes. (For present needs and purposes the proposed waste classification system will apply only to solid radioactive waste forms.):

- Non-radioactive waste that may be handled as routine trash because of its innocuously low radioactive levels.
- Low-level waste requiring active confinement (Confinement or holdup with controlled or predictably low release rates). This waste class would include materials that, due to their low specific hazard levels or short decay times, may be adequately controlled in a suitably designed and operated containment facility.
- High-level waste requiring isolation (complete containment with no expected release to the biosphere for geological periods of time). This class of waste will contain radioactive materials of very high hazard potential or long decay times.

Figure 3 is a schematic of the proposed WC system and shows that it meets the objective for simplicity. Provision of a more complicated system is not necessary and might even be counterproductive.

Alpha emitting material including transuranics are not classified separately but should follow essentially the same disposal criteria as other radioactive waste material.

In the classification system, the method governing the disposition of waste should be based primarily on the hazard potential (hazard index, longevity, and migration) and expressed in terms of radioactivity per unit volume or mass at the time of disposal.

Because small release rates are possible during active confinement of low-level wastes (dotted line in Fig. 3), total inventory limits may be required for all such facilities.

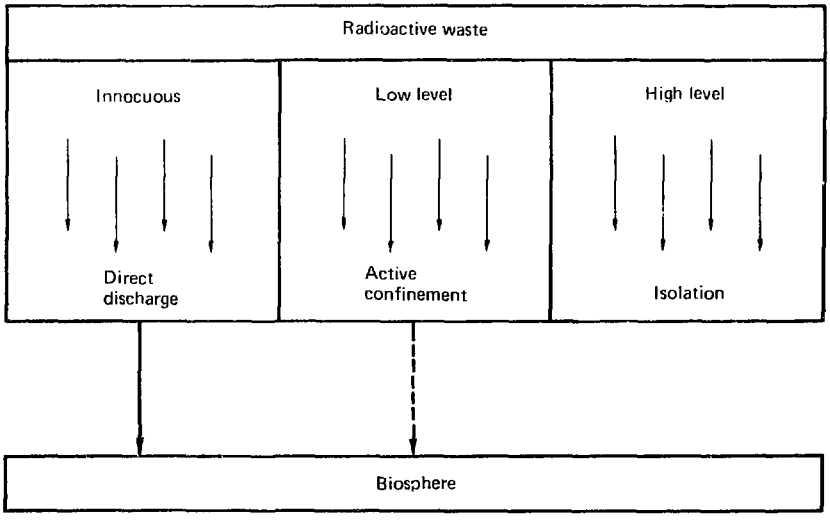

FIG. 3. Schematic of proposed radioactive waste classification system.

ESTABLISHING INTERFACE LEVELS

The key to establishing quantitative interface levels between the three proposed waste classes lies in defining the concentration limits for radioactive waste in the LLW class so that guidelines for radiation exposure to the public are not exceeded.

Radiation guidelines are defined on the basis of the annual dose equivalent that the public might receive from exposure to the radioactive material in the waste. As a guide in this study, a maximum annual dose equivalent of 0.5 rem/y was used when the exposure involved a few individuals. When the potential size of the exposed population was large, 0.005 rem/y was used.

Current practice in the United States is to dispose of LLW by shallow land burial. There are now six commercial LLW burial facilities and the U.S. Department of Energy (DOE) operates five other major facilities.

We reviewed the characteristics of these facilities to establish conservative yet realistic parameters as input to a computational model based on a reference containment facility (RCF). Table 3 summarizes data on these existing facilities. Appendix F covers the review in more detail.

Releases of radioactivity from the RCF can be estimated from analyses of present LLW. Projections of the activities and volumes of wastes to be generated in the future are also important in determining the relative costs and potential risks associated with LLW containment facilities. Table 4 gives the characteristics of radionuclides generated in light water reactors (LWR) and destined to be treated as LLW.[10] Table 5 gives the estimated volume and radioactive concentration of four waste classes generated per GWe-y by LWR's.[2]

TABLE 3. Survey of existing low-level waste disposal facilities.

Location	Commercial capacity, m^3	Climate	Nearby rivers	Cover depth, m	Observed radionuclide migration
Hanford, Washington	---	Semiarid	10 km to Columbia	2.5	Through uptake by deep rooted plants
Richland, Washington[a]	9×10^5	Semiarid	10 km to Columbia	2	Not observed
Beatty, Nevada	7×10^5	Arid	3 km to Amargosa	2	Not observed
INEL, Idaho[a]	---	Semiarid	3 km to Big Lost	1	Possibly by on-site groundwater
Los Alamos, New Mexico[a]	---	Semiarid	8 km to Rio Grande	1.5	On-site vadose water zone
Sheffield, Illinois	2×10^5	Humid	Site boundary	1	Not observed
Morehead, Kentucky	3×10^6	Humid	500 m	1	On and off-site ground and surface water
Oak Ridge, Tennessee[a]	---	Humid	On-site	1	On-site groundwater, off-site surface water
Savannah River, South Carolina[a]	---	Humid	On-site Savannah	1.2	On-site groundwater
Barnwell, South Carolina	2×10^6	Humid	Site boundary	3	Not observed
West Valley, New York	2×10^5	Humid	On-site	3	On-site groundwater, off-site surface water

[a]DOE sites

TABLE 4. Calculated concentrations of radionuclides in LWR solid wastes shipped to commercial burial grounds.

Nuclide	Half-life, y	Concentration, Ci/m^3	Total activity, Ci/Gwe-y
^{3}H	12.3	7.06	4449
^{60}Co	5 3	2.47	1557
^{137}Cs	30.1	1.76	1112
^{124}Cs	2.06	1.06	667
^{51}Cr	0.075	1.06	667
^{55}Fe	2.7	0.71	445
^{54}Mn	0.85	0.35	222
^{90}Sr	29	0.0035	2
^{238}Pu	87.8	0.0035	2
^{239}Pu	24,400	0.0035	2
^{241}Pu	15	0.035	2
^{237}Np	2,100,000	0.0035	2
^{241}Am	433	0.0035	2
^{242}Cm	0.45	0.0035	2
^{244}Cm	17.9	0.0035	2
^{59}Ni	8,000	0.0014	0.9
^{99}Tc	213,00	0.0003	0.02
^{129}I	15,900,000	0.00001	0.004
	Total	14.53	9136.

Source: Ref. 10, Appendix H.

TABLE 5. Projected relative waste concentrations and volumes per GWe-y.

Waste categories as generated	Gross radioactivity concentration, Ci/m^3		Volume, m^3/GWe-y
	Upper limit	Average	
Routine low-level	100	20	660
Intermediate level 1	1,000	350	75
Intermediate level 2	10,000	3,500	20
High-level	10,000,000	3,500,00	3

Source: Ref. 2.

The RCF used as our model is described in the appendixes. It is a shallow land burial facility assumed to be in a semi-arid region and located 1 km from a large river. An aquifer lies 10 m below the bottom of the burial trenches; the water in this aquifer flows toward the river at a velocity of 111 m/y (1 ft/day). The total capacity of the RCF is 6×10^5 m^3, large enough to contain the volume of LLW generated by 1000 GWe-y of nuclear power production. Figure 4 is a schematic drawing of the RCF. Table 6 lists the parameters used in describing the behavior of the buried wastes in the RCF.

The methodology for determining the waste class interfaces involves five basic steps:

- Identifying a set of conservative exposure scenarios.
- Determining the transport of the radioactivity through the environment to man.
- Calculating both the maximum individual dose and the total population dose.
- Relating these calculated doses to the radiation exposure guidelines.

Once these steps are defined, the dose to the exposed population can be related to the concentration of radioactivity in the waste at the time of burial.

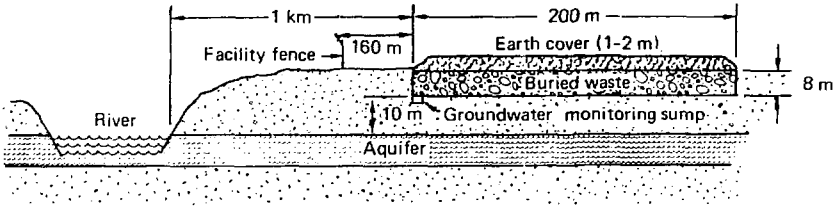

FIG. 4. Schematic of reference containment facility.

TABLE 6. Reference containment facility parameters and values.

Parameter	Value
Site plan area, m^2	2×10^6
Size of trenches--length, width, depth, m	200 × 10 × 8
Number of pits	100
Distance to site boundary, m	160
Distance between pit and aquifer, m	10
Water velocity from pit to aquifer, m/y	10
Annual precipitation, m/y	0.1 - 2
Aquifer flow area, m^2	1000
Distance from site to surface water, km	1
Aquifer water velocity, m/y	111
Dispersion coefficient, m^2/y	0.42
Meterology - Pasquill stability conditions	E and F
Average windspeed, m/sec	1.6
Bulk soil and waste density, g/cm^3	2
Reference dust loading, mg/m^3	2

The interface level can be based on volume concentration, which simplifies the process of determining the classification of any given waste container.

The major exposure to the general public from radioactive waste released to the environment results from deposition of the radionuclides in the body. Deposition occurs after the radioactive material enters the body by any of four different mechanisms: ingestion, inhalation, injection (as from a wound), or infusion (where the radionuclide passes through the skin barrier into the bloodstream). The two entry pathways of most concern are ingestion and inhalation. Therefore, this analysis is limited to these two entry pathways.

The scenarios selected for study were as follows:

1. Inhalation Exposure
   a. Reclamation and reuse of the burial site after institutional controls were removed.
   b. Continuous airborne releases due to wind erosion of soil from the disposal site.
   c. Airborne release caused by an accident at the site during the operation phase.
2. Ingestion Exposure
   a. Leaching of the radionuclides into an underground aquifer that ultimately discharges into a surface stream and hence to human food and water.
   b. Erosion of the overfill cover with transport of the radionuclides by surface runoff to a nearby river and hence to human food and water.
   c. Well drilling and consumption of the water directly below the disposal site.

Figure 5 diagrams the exposure pathways.

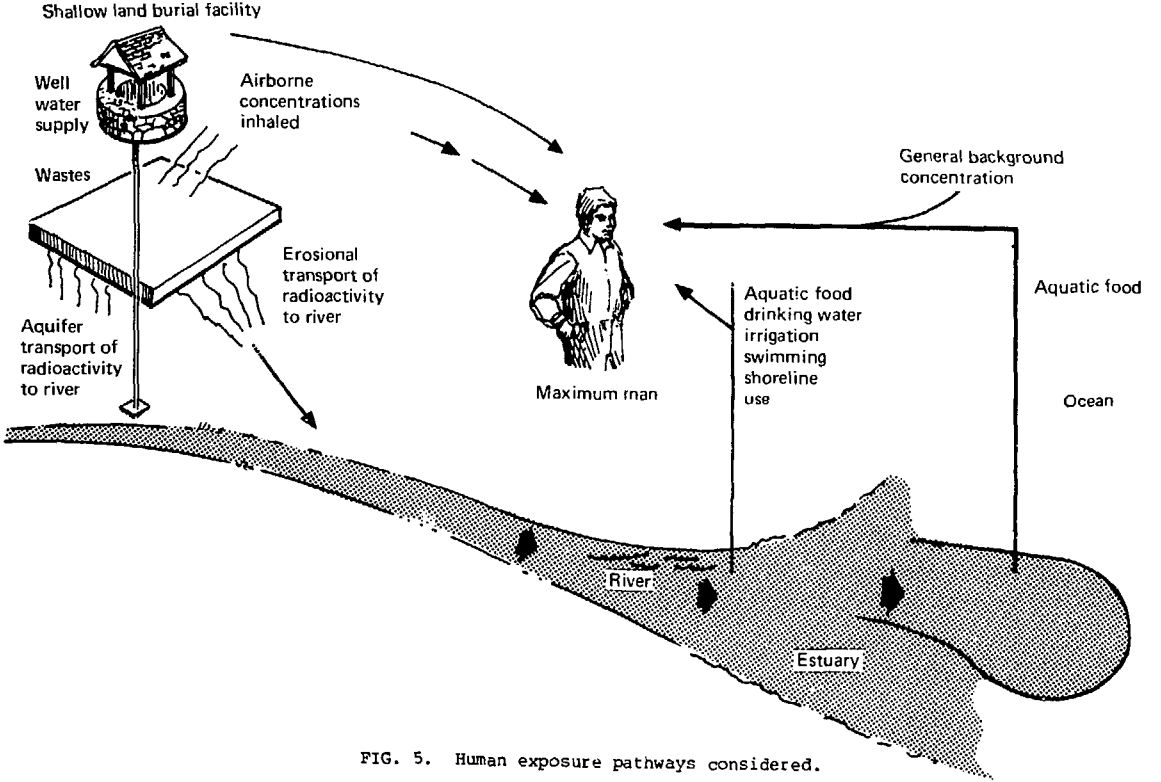

FIG. 5. Human exposure pathways considered.

The source terms and release fractions from the RCF were based partly on assumptions and partly on empirical data. The assumptions are identified in the detailed calculations. We believe them to be conservative. Tables 6 and 7 show the data used in determining the source terms. The volumetric concentration in the RCF after burial is assumed to be 30 times less than the maximum permitted concentration in waste containers at time of burial. We arrived at this factor by assuming a dilution factor of 3 due to the surrounding soil mixing with the waste during burial, and a factor of 10 from the ratio of peak concentration to the average waste concentration.

The transport of the radioactivity through the environment to man was defined using empirical data. In the case of airborne transport, we used the Pasquill diffusion equations to determine the concentration of activity downwind.

TABLE 7. Nuclide-specific parameters and values.

Nuclide	Reference inventory, Ci	Sorption coefficient	Reference leach rate, y^{-1}
^{90}Sr	24,000	100	1×10^{-2}
^{129}I	0.73	1	1×10^{-2}
^{137}Cs	35,000	1,000	1×10^{-2}
^{237}Np	12,000	100	6×10^{-2}
^{239}Pu	65,000	10,000	6×10^{-4}
^{241}Am	65,000	1,000	6×10^{-4}

In the case of water transport, we determined the leached nuclide migration in the aquifer from a second-order differential mass-balance equation that considers longitudinal dispersion, convection, sorption, and radioactive decay.

The calculations determining source terms and resulting dose to the public from airborne releases are detailed in Appendix H. The calculations for water transport are given in Appendix I.

DOSE CALCULATIONS

The toxicity or harm to man from a radionuclide present in the body depends on the specific activity of the nuclide, its chemical form, and the mode of entry into the body. The chemical form of the radionuclide affects the solubility in the body, which in turn determines the rate of translocation to and from the organs of the body. Three categories of solubility are defined by the ICRP.[11] These are the "D" class (readily soluble), the "W" class (moderately soluble), and the "Y" class (relatively insoluble). Ingested insoluble compounds containing radioactive elements tend to pass through the gastrointestinal tract with little uptake. When inhaled, however, a significant fraction of insoluble material can remain in the lungs for as long as several years.

Doses for airborne exposures using recent ICRP data[12] on the reference man were calculated using the AERIN code developed at LLL.[13] Examples of computed organ doses due to inhalation of ^{239}Pu are given in Figs. 6 and 7 for short-term and prolonged exposures respectively. Use of ICRP data on reference man results in lower maximum permissible concentrations in air and water than the current guides given in 10CFR20.[6] Care should therefore be exercised in comparing these values with 10CFR20 and appropriate correction factors should be applied for valid comparisons.

We determined that for mixed TRU waste, it would be fairly conservative to assume all the alpha radiation emanated from ^{239}Pu decay. Table 8 gives organ doses for TRU mixtures aged 6 mo and 100 y relative to those for ^{239}Pu.

A number of isotopes were investigated in each of the exposure pathway scenarios. These isotopes were selected to represent the worst case in any given scenario. Other isotopes were then scaled to the appropriate reference isotope by the ratio of their radiologic effects from air and water pathways to obtain interface values for each isotope of concern.

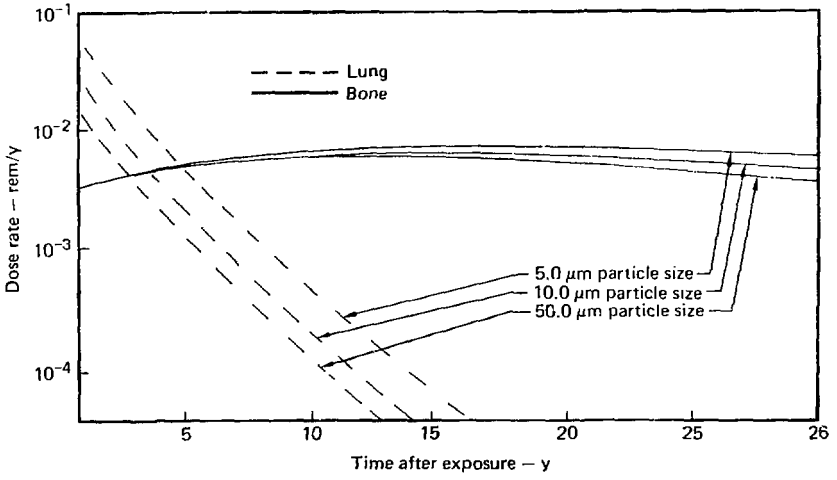

FIG. 6. Effects of short-term inhalation (500 pCi/d for 30 d) of ^{239}Pu oxide.

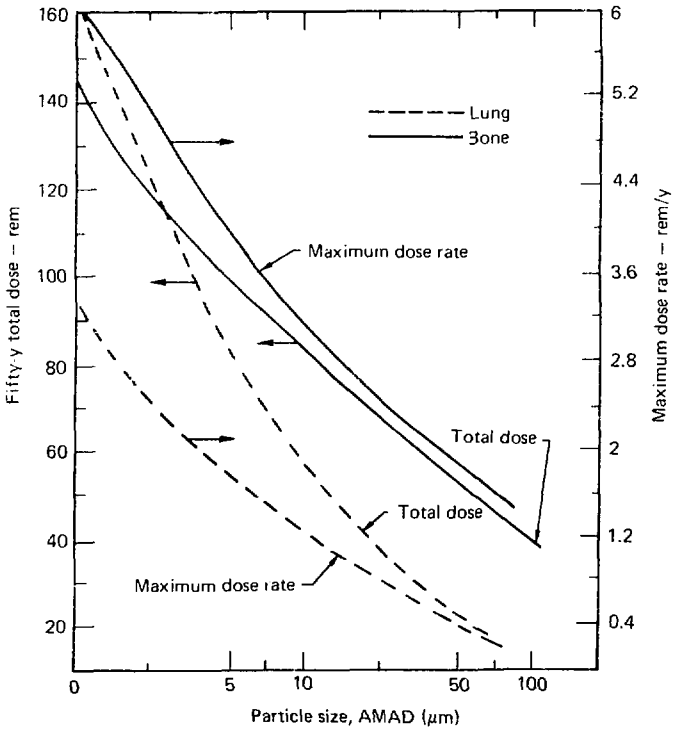

FIG. 7. Dose resulting from continuous 50-y inhalation exposure to 1.0 pCi/m^3 air concentration of ^{239}Pu.

TABLE 8. Organ doses for mixed TRU nuclides aged 6 mo and 100 y relative to ^{239}Pu doses.

Nuclide	Assumed TRU mixtures: Percentage of total curies	
	TRU mix cooled 6 mo	TRU mix cooled 100 y
^{238}Pu	1.7	17.8
^{239}Pu	0.2	3.4
^{240}Pu	0.3	6.6
^{241}Pu	77.0	16.9
^{241}Am	0.1	54.3
^{242}Cm	19.1	0.0
^{244}Cm	1.6	0.8

Accumulated lung dose, rem/nCi inhaled (30-day exposure, 10-μm particles)

Year	^{239}Pu	TRU mix cooled 6 mo	TRU mix cooled 100 y
1	0.06	0.01	0.05
5	0.14	0.03	0.16
10	0.15	0.03	0.17
15	0.16	0.03	0.17
20	0.16	0.03	0.17
30	0.16	0.03	0.17
40	0.16	0.03	0.17
50	0.16	0.03	0.17

TABLE 8. (cont'd.)

Accumulated bone dose, rem/nCi inhaled (30-day exposure, 10-um particles)			
Year	^{239}Pu	TRU mix cooled 6 mo	TRU mix cooled 100 y
1	4.3×10^{-4}	8×10^{-4}	4×10^{-3}
5	0.03	8×10^{-3}	0.03
10	0.07	5×10^{-3}	0.07
20	0.17	7×10^{-3}	0.11
30	0.26	8×10^{-3}	0.14
40	0.34	0.012	0.21
50	0.42	0.015	0.33

RELATIVE HAZARD INDEX DEFINED

Calculations indicated that the reclamation scenario dictated the most restrictive interface concentrations between LLW and HLW. We also found that ^{239}Pu could reasonably be used as the reference isotope in the reclamation scenario.

The interface concentration for each isotope was then normalized to a relative hazard index (RHI) to present the results in a concise format. The RHI is defined by the equation:

$$RHI = \frac{DC}{\rho (MPC)_a} ,$$

where C is the concentration of the nuclide in the soil (i.e., 1/30 of the permitted peak concentration in the waste container), D is the average dust loading in the respirable air (mg/m^3) and ρ is the bulk density of the soil.

The use of the RHI allows the relative risk of a radioactive species to be described by a single number.

The reclamation scenario assumes that exposure occurs only after institutional control of the facilities has ended. Since institutional control would be maintained for 100 to 300 y after closure of the burial facilities, credit can be taken for the radioactive decay of the short-lived nuclides during that period. Calculations based on these assumptions indicate that the accident scenario can give the most restrictive interface concentrations for those isotopes with short half-lives.

EVALUATION AND DISCUSSION OF RESULTS

Predicted doses resulting from the scenarios discussed in the previous section were calculated and the results are summarized in Fig. 8. Appendixes H and I present detailed discussion of the calculations. A general discussion follows.

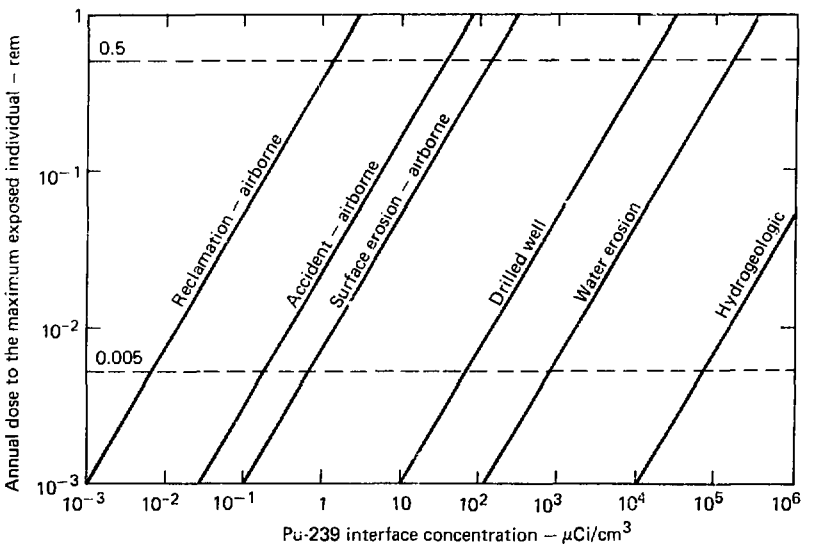

FIG. 8. Annual individual dose vs HLW/LLW interface concentrations for ^{239}Pu calculated for six exposure scenarios.

INHALATION SCENARIOS

The eventual reclamation and reuse of the disposal site after institutional controls are removed dictates the most restrictive LLW/HLW interface concentrations. ^{239}Pu is one of the most restrictive isotopes in the inhalation scenarios studies. Interface concentrations for other isotopes except ^{129}I were determined by normalizing the ^{239}Pu interface concentration by the MPC_a for ^{239}Pu and multiplying by the MPC_a for the other isotopes. ^{129}I is discussed under "Ingestion Scenarios" below.

Since the reclamation scenario applies only to the period following loss of administrative control of the burial site (100 to 300 y after closure), radioactive decay is taken into account only for isotopes with short half-lives. There is a limit, however, in considering the decay because the near-term accident scenario becomes the major restriction at higher concentrations of the short-lived isotopes.

The HLW/LLW interface concentrations for individual radionuclides are shown in Table 9. A concentration of 1 $\mu Ci/cm^3$ of ^{239}Pu in the waste container at the time of burial results in about 0.5 rem/y exposure to an individual worker in the reclamation scenario. In the accidental release scenario, a concentration of 1 $\mu Ci/cm^3$ in the waste container results in a maximum individual dose of 0.01 rem/y to an off-site resident.

INGESTION SCENARIOS

For the waterborne scenarios, the HLW/LLW interface concentrations were less restrictive than for the airborne scenarios. However, this does not hold for ^{129}I.

TABLE 9. Single radionuclide HLW/LLW interface concentration values (adjusted for 100 y decay).

Nuclide	Half-life, y	MPC*, μCi/ml	Interface concentration, μCi/cm^3	
^{3}H	12.33	2×10^{-7} (S,I)**	2×10^{7}	(Note b)
^{51}Cr	0.075	8×10^{-8}	6×10^{6}	(Note b)
^{54}Mn	0.86	1×10^{-9} (S,I)	2×10^{6}	(Note b)
^{55}Fe	2.7	3×10^{-8} (I)	2×10^{6}	(Note b)
^{60}Co	5.27	3×10^{-10} (I)	2×10^{4}	(Note b)
^{59}Ni	8×10^{4}	2×10^{-8} (S)	3.4×10^{5}	
^{63}Ni	100	2×10^{-9} (S)	6.7×10^{4}	
^{90}Sr	29	$3\ 10^{-11}$ (S)	2×10^{3}	(Note b)
^{90}Y	0.007	3×10^{-9} (I)	2×10^{5}	(Note b)
^{99}Tc	213	2×10^{-9}	4.6×10^{4}	
^{106}Ru	1.010	2×10^{-10} (I)	2×10^{4}	(Note b)
^{129}I	1.59×10^{7}	2×10^{-11}	1.8	(Note d)
^{134}Cs	2.06	4×10^{-10}	2×10^{4}	(Note b)
^{137}Cs	30.1	5×10^{-10} (I)	4×10^{4}	(Note b)
^{232}Th	1.4×10^{10}	1×10^{-12} (S,I)	17	
^{235}U	7.04×10^{8}	4×10^{-12} (I)	68	
^{238}U	4.5×10^{9}	3×10^{-12} (S)	50	
^{237}Np	2.14×10^{6}	1×10^{-13} (S)	1.7	

TABLE 9 (cont'd.)

Nuclide	Half-life, y	MPC*, μCi/ml	Interface concentration, μCi/cm^3	
^{238}Pu	87.8	7×10^{-14}(S)	2.6	
^{239}Pu[a]	2.439×10^4	6×10^{-14}(S)	1.0	
^{241}Pu	15	3×10^{-12}	130	(Note c)
^{241}Am	433	2×10^{-13}(S)	4.0	
^{242}Cm	0.45	4×10^{-12}	3×10^2	(Note b)
^{244}Cm	17.9	3×10^{-13}	24	(Note b)
(Note e)	-	1×10^{-10}	1.7×10^3	
(Note f)	-	2×10^{-14}	0.33	

Notes

*From 10CFR20.[6] **S (soluble), I (insoluble).

a. Bases from which other numbers were derived.

b. Interface concentration limited by accident scenario.

c. Based on ingrowth of daughters.

d. Based on the well-water scenario.

e. Any other single nuclide, not listed above, with decay mode other than alpha emission or spontaneous fission.

f. Any other single nuclide, not listed above, that decays by alpha emission or spontaneous fission.

Because ^{129}I has a long half-life and essentially no sorption in the soil, the well-water scenario limits the interface concentration of this isotope to 1.8 μCi/cm^3 in the buried waste.

RELATIVE HAZARD INDEX

To indicate the risk for a given nuclide and its interface value (Table 9), we must define the RHI in terms of the corresponding MPC_a, since the potential effects of airborne releases were shown to be generally more serious than those from waterborne releases. For ^{239}Pu the RHI is equal to 1.7×10^4 if we assume a density of $2 g/cm^3$

We based our interface values on the releases of ^{239}Pu. Interface levels for other nuclides have been normalized to that of ^{239}Pu by use of the appropriate MPC values. Use of the MPCs assumes that they were uniformly determined, and that they can adequately serve as relative indicators of risk for the nuclides involved.

Figure 9 is a representation of the waste classification system in the RHI format. This figure also shows how the radioactive half-life is considered.

The HLW/LLW interface concentrations are suitably increased to account for radioactive decay, depending on the delay between the end of burial operations and reclamation of the land for construction. Delays of 100, 200, and 300 y were assumed. For half-lives greater than a few hundred years, radioactive decay is not considered in determining the longevity of the hazard, since the conservative assumption is made that these nuclides will exist forever.

The most restrictive scenario (eventual reclamation of the land) provides the basic HLW/LLW interface. However, the accident scenario becomes the most restrictive for short half-life materials because the accident is assumed to occur before there is any significant radioactive decay.

For mixtures of nuclides in the wastes, the relative hazard indices can be weighted as indicated in 10CFR20.[6] That is, the cumulative hazard index can be found by adding up the concentration-to-MPC ratios of the individual nuclides multiplied by their dust-loading corrections. Appendix J gives details and examples of the use of this system.

For the interface between innocuous and low level waste, further study is required to provide a precise and defensible value.

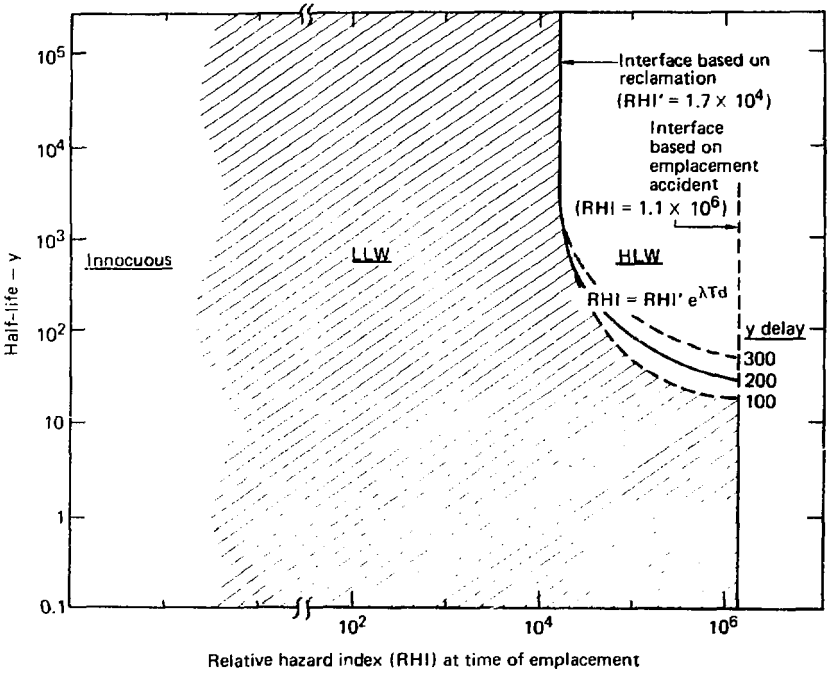

FIG. 9. Waste classification system in relative hazard index format.

COST-BENEFIT ANALYSIS

Figure 10 presents the results of cost-benefit calculations performed to evaluate the cost effectiveness of shallow land burial vs repository isolation as a function of HLW/LLW interface level.

Given the projected waste quantities and costs in Appendix G, we can determine the cost effectiveness vs interface level. Increasing the interface level makes additional quantities of waste acceptable for disposal as LLW at a lower cost than that of isolation. However, the risk in terms of population dose would increase. For example, the first projected category of waste higher in concentration than routine LLW would contain 1000 Ci/m^3 of gross activity. The generation of 1.0 GWe-y of nuclear power would produce 75 m^3 of 1000 Ci/m^3 waste. For a marginal cost of \$3400/m^3, \$255,000/GWe-y could be saved by handling this waste as LLW rather than sending it to the repository. Alternatively, it would cost \$3400/m^3 more to send it to a repository, but there would be a concomitant decrease in potential risk from this volume of waste.

If the HLW/LLW interface level were to be established solely on the basis of cost-benefit analysis, rather than at a cost effectiveness of \$1000/man-rem, the level would be set at about 60,000 Ci/m^3. Such concentrations are higher than we would get if HLW were simply treated as LLW and placed in a shallow land burial facility. Thus, the cost-benefit approach does not yield a limiting upper concentration interface. Although not as restrictive an indicator of the relative hazards as are the maximum individual dose approaches, cost-benefit analysis provides useful insight and perspective on the waste management question. Assuming the Appendix I of 10CFR50 value of \$1000/man-rem as an indicator of acceptability, the interface levels suggested in this study would fall well within ALARA guidelines. The incremental cost effectiveness of reducing the interface beyond 1.0 $\mu Ci/cm^3$ would be about 10^8 \$/man-rem.

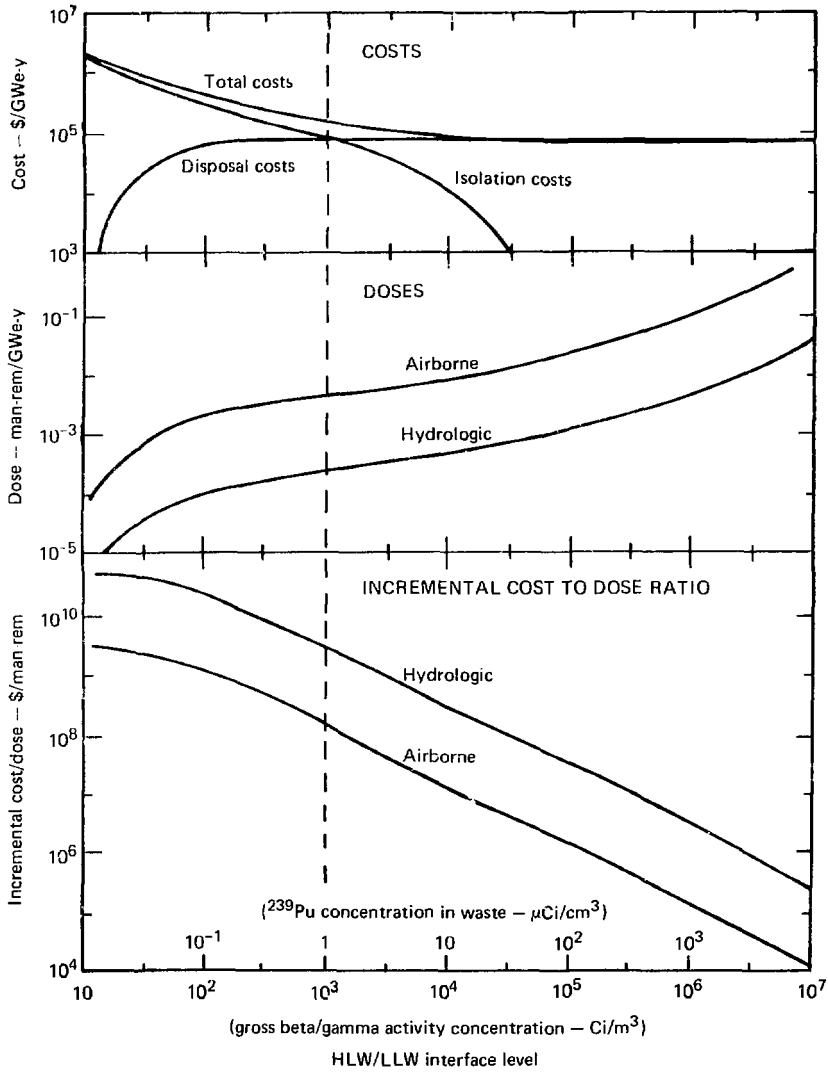

FIG. 10. Cost-benefit analysis results.

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APPENDIX A

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APPENDIX B

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APPENDIX C

PREVIOUSLY PROPOSED RADIOACTIVE-WASTE CLASSIFICATION SYSTEMS

Several waste classification (WC) systems have been proposed during the past few years. A review of these systems follows.

IAEA RADIOACTIVE-WASTE CATEGORIES

The IAEA (1967) tabulated information from 11 countries on their WC systems.[C1] It was found that no two countries have the same classification system and that some countries use more than one system.

The IAEA system categorizes the waste as liquid, solid, or gaseous, with each category broken down as described below.

Liquid Wastes

The subclassification under liquid wastes is based on activity concentration levels as follows:

Category	Activity level, A, μCi/ml	Remarks
1	$A < 10^{-6}$	Not normally treated
2	$10^{-6} < A \leq 10^{-3}$	Without shielding[a]
3	$10^{-3} < A \leq 10^{-1}$	Shielding possible[a]
4	$10^{-1} < A \leq 10^{-4}$	Shielding necessary[a]
5	$10^{4} < A$	Cooling necessary

[a]Treatment is by usual methods (i.e., evaporation, ion-exchange, or chemical treatment).

Solid Wastes

In the solid-waste category, there are three considerations, as follows:

- For strong beta and gamma emitters, the surface dose rate is the controlling factor.
- For strong alpha emitters, the activity per volume (Ci/m^3) controls.
- For large amounts of fissile materials, additional packing requirements, such as IAEA "Regulations for the Safe Transport of Radioactive Materials," are controlling.

For the three considerations, the following classification is suggested:

Category	Radiation dose, D, on surface of wastes, R/hr	Remarks
1	$D \leq 0.2$	β-γ-emitters dominant;
2	$0.2 < D \leq 2$	α-emitters are insignificant
3	$2 < D$	α-emitters dominant;
4	α activity expressed in Ci/m^3	β-γ-emitters insignificant; criticality is no problem[a]

[a]If criticality is a matter of concern with the waste, it is assumed that the solid wastes are treated or packaged or both to prevent criticality; it is also suggested that Pu and U wastes are potentially retrievable and may be separated for economic reasons.

Gaseous Wastes

The basic consideration for classifying gaseous wastes is activity level expressed in Ci/m^3:

Category	Activity level, A, Ci/m^3	Remarks
1	$A \leq 10^{-10}$	Effluents not usually treated.
2	$10^{-10} < A \leq 10^{-6}$	Effluents usually treated by filtration.
3	$10^{-6} \leq A$	Effluents usually treated by methods other than filtration.

Discussion

Figure C1 is a diagram of the IAEA waste classification system. The following waste characteristics must be known for this system to be used to categorize any given waste:

- Physical state (solid, liquid, gaseous).
- Activity level for α-dominant solids, for liquids, and for gases.
- Radiation dose at the surface for β-γ-dominant solids.
- Radiation type, indirectly for solids.

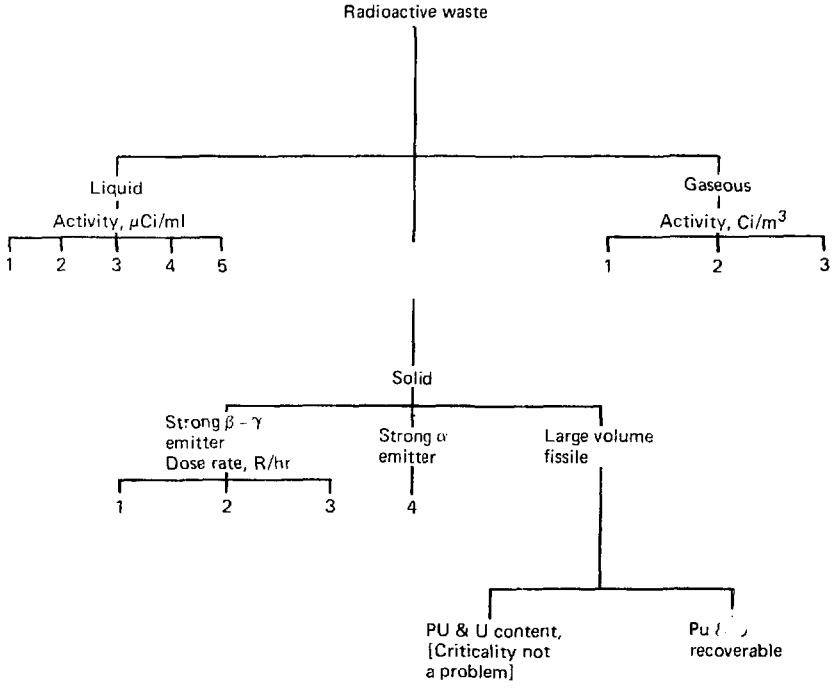

FIG. C1. IAEA waste classification system.

In addition, the following comments apply:

- Compared with other systems, the IAEA system is uncomplicated.
- The expression of radioactivity in curies gives little indication of the toxicity or hazard potential.
- The categories give no guidance for disposal methods.
- The system does not consider the half-life or longevity of potential hazards.

AIChE RADIOACTIVE WASTE CATEGORIES

The waste classification system formulated under the sponsorship of the AIChE[C2] is based on the following premises:

- The major parameter for classification is the "ratio of radionuclide concentration in the waste (at the time of consideration) to the MPC of the same waste constituents in water, air, or solids" in terms of MPC, MPE, or MPQI, as recommended by the International Commission on Radiological Protection (ICRP).
- Wastes should be classified while they are in containment or before release. Decay during storage may result in reclassification of wastes.
- Individual radionuclide concentrations must be determined for exact categories.
- Liquids and gases are treated as fluids; solids require further considerations.
- The smaller ICRP-MPC value should be used when MPCs are specified for different solubilities.
- Primary classification parameters do not include chemical content, physical properties, origin, total amounts of waste, half-life, or degree of natural reconcentration once released.
- Definitions proposed are intended for use to describe the characteristics of wastes in judgments involving treatment, disposal, regulatory action, and definitions of safety.

Table C1 summarizes this waste classification system. Using the table requires the following information:

- Where wastes contain mixtures of radionuclides, the ratio of nuclide concentration to MPC for example, is determined by using the MPC for known mixtures or unidentified mixtures of radionuclides as recommended by the ICRP.
- Units of MPQI/kg are used for solids because this measure is comparable to units for liquids and gases even though solids are not considered ingestible.

Waste Categories

As Table C1 shows, this classification system divides the waste into five major categories, Classes A, B, C, D, and E, in ascending order of radionuclide concentrations. The following statements apply.

Class A (Population Level). Class A includes waste with radionuclide concentrations not exceeding (uncontrolled) population level MPC values. This statement should not be construed to mean that there are no restrictions on the release of these wastes to the environment other than constraints such as ALAP and ALARA.

Class B (Occupational Level). One or more of the nuclides or the mixture of radionuclides exceeds Class A limits, but no radionuclide or mixture in liquid or gaseous wastes exceeds the specified MPC (40 hr/wk) for normal occupational exposure and no nuclide or mixture in solid wastes exceeds specified MPQI/kg for normal occupational exposure.

Class C (Low Level). Limits on radionuclide concentrations pertain to one or more radionuclides or to a mixture of radionuclides. Treatment of some wastes in this category may convert the major fraction to a Class A or Class B waste, leaving a minor fraction of concentrated wastes. Subclasses C-1 and C-2 are based on ICRP maximum permissible radiation exposure to the whole body. This breakdown, in many cases, would separate alpha from beta-gamma wastes. (Category C is based on radionuclide content; the subcategories are based on surface radiation.)

TABLE C1. AIChE radioactive waste categories.

Class A (Population Level)	
Liquid and gas	Less than or equal to the maximum permissible concentration (MPC)[a] for members of the population at large (including persons living in the neighborhood of controlled areas)
Solid	Less than or equal to the maximum permissible quarterly intake (MPQI) for members of the population at large (including persons living in the neighborhood of controlled area) kg solid
Surface radiation[c]	Less than or equal to the maximum permissible whole body exposure (MPE)[a] for members of the population at large[b] (including persons living in the neighborhood of controlled areas)
Class B (Occupational Level)	
Liquid and gas	Greater than Class A but less than or equal to (MPC)[a] occupational exposure (40-hr wk)
Solid	Greater than Class A but less than or equal to (MPQI)[d] for occupational exposure/kg solid
Surface radiation[c]	Greater than Class A but less than or equal to (MPE)[c] for occupational exposure[b] (40-hr wk)
Class C (Low Level)	
Liquid and gas	Greater than Class B but less than or equal to $10^4 \times$ Class B
Solid	Greater than Class B but less than or equal to $10^4 \times$ Class B
Surface radiation	
Class C-1	Less than or equal to (MPE)[e] (occupational) at surface[b]
Class C-2	Greater than (MPE)[e] (occupational) at surface[b]

TABLE C1. (cont'd.)

Class D (Intermediate Level)	
Liquid, gas, or solid	Greater than Class C but less than or equal to 10^4 × Class C
Solid	Greater than Class C but less than or equal to 10^4 × Class C
Surface radiation	
Class D-1	Less than or equal to (MPE)[e] (occupational) at surface[b]
Class D-2	Greater than (MPE)[e] (occupational) at surface[b]
Class E (High Level)	
Liquid, gas or solid	Class D
Surface radiation	
Class E-1	Less than or equal to (MPE)[e] (occupational) at surface[b]
Class E-2	Greater than (MPE)[e] (occupational) at surface[b]

[a]ICRP Publication 6, Recommendations of the International Commision on Radiological Protection, as Amended 1959 and Revised 1962, (1964), Pergamon Press.

[b]In the 1958 report of ICRP Committee II, the recommended MPE to the gonads or the whole body is 5 rem/y for occupational exposure. For the population at large, it is assumed to be 1/100 of 5, or 0.05, for whole body genetic effect.

[c]Applies to surface of container, object, or pool of liquid.

[d]MPQI = (MPC_w for 168-hr occupational exposure) (2200 ml/day, 91 days/quarter) = (MPC_w) (2×10^5); MPC applies to water.

[e]Section C of Radiation Protection-Recommendations of the International Commission on Radiological Protection (adopted September 9, 1958) (Pergamon Press, London, 1959) (ICRP, Pub. 1).

Class D (Intermediate Level). In general, Class D wastes will be stored for safety and decay purposes, or converted to or incorporated in solids for ultimate disposal. (Incorporation into organic matrix or cement before disposal is considered for Class D wastes.) Subcategories are defined similarly to Class C subcategories.

Class E (High Level). These wastes usually will be stored for safety and decay purposes, or converted or incorporated into radiation-stable solids for disposal. Subcategories are similar to Class D subcategories.

Discussion

Figure C2 is a diagram of the AIChE waste classification system. Three waste characteristics must be known for this system to be used to categorize any given waste:

- Physical state (solid, liquid, gaseous).
- Radionuclide content as MPC for liquids and gases or MPQI for solids.
- Surface radiation as MPE for solids.

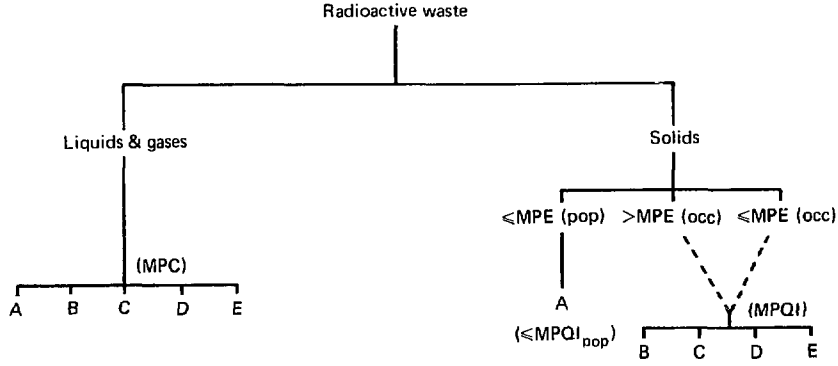

FIG. C2. AIChE waste classification system.

The following comments also apply to this system:

- Its categories indicate potential hazards.
- It is relatively simple.
- It offers guidance for a suitable disposal method.
- It considers physical states.
- It may be operationally complicated.
- It does not consider half-lives to ascertain duration of hazard.

GERA'S PROPOSED WASTE-CLASSIFICATION SYSTEMS

Gera suggests that radioactive wastes should be classified according to the duration of required containment.[C3] Clear segregation of wastes requiring containment for relatively short times from those presenting a significant environmental hazard for geologic time periods would probably be sufficient. Two points that Gera makes are that (1) the destiny of the waste is a relevant classification factor, and that (2) half-lives of the radionuclides contained in the waste are a controlling element in future waste management steps. Figure C3 is a diagram of this proposed system.

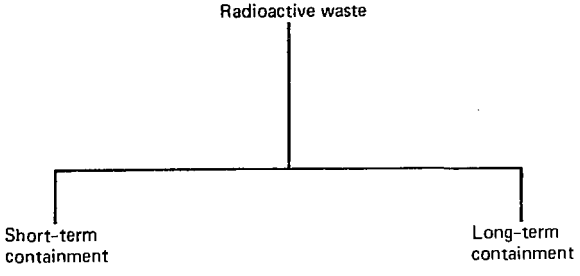

FIG. C3. Gera's classification system based on duration of containment.

Tentative limits of concentration for long-lived alpha emitters to be used for the segregation of solid wastes have been proposed in the United States and in France. These limits are 10μCi/kg and 10,000 time MPC_w, respectively. Concentration limits might not suffice as criteria. It may be necessary to introduce an additional limit on the long-lived activity that can be disposed of in particular burial grounds. This inventory limit should be based on the possible transfer of the radioactivity to man after abandonment of the burial ground.

1968 CLASSIFICATION SYSTEM

Gera[C4] previously developed a rigorous waste classification system based on MPCs, activity concentrations, half-lives and exposure rates.

System

This system is described in Tables C2, C3, and C4, and diagrammed in Fig. C4. Detailed knowledge of the radionuclides and their concentrations is required for use of this system. Subsequent work on the development of a functional waste classification system led Gera to conclude that his proposed 1968 system and those comparable to it were "rather impractical and not relevant to the actual needs of waste management."

The 1968 classification system requires that the following waste characteristics be well known: (1) physical state, (2) MPC, (3) half-life, (4) activity concentration, and (5) exposure rate. In addition, it should be noted that this system: (1) considers physical states, (2) provides guidance for disposal of waste, and (3) tends to be complex in that it requires detailed knowledge of waste composition. These attributes imply that a detailed knowledge of the radionuclide content is also available for each waste.

TABLE C2. Categories of liquid and gaseous wastes proposed by Gera (1968).

Categories according to MPC multiplication factors	Examples of possible waste categories	Categories according to treatment
A. < MPC	A-1 (Discharge) A-2 (Treatment; DF < 10^3)	1. No treatment; discharge
B. From MPC to 10^3 MPC	B-1 (Discharge) B-2 (Treatment; DF < 10^3) B-3 (Treatment; 10^3 < DF < 10^6)	2. Treatment with DF up to 10^3
C. From 10^3 MPC to 10^6 MPC	C-1 (Discharge) C-2 (Treatment; DF < 10^3) C-3 (Treatment; 10^3 < DF < 10^6) C-4 (Treatment; DF > 10^6)	3. Treatment with DF between 10^3 and 10^6
D. From 10^6 MPC to 10^9 MPC	D-2 (Treatment; DF < 10^3) D-3 (Treatment; 10^3 < DF < 10^6) D-4 (Treatment; DF > 10^6) D-5 (Containment)	4. Treatment with DF exceeding 10^6
E. Greater than 10^9 MPC	E-3 (Treatment; 10^3 < DF < 10^6) E-4 (Treatment; DF > 10^6) E-5 (Containment)	5. No treatment; containment

Note: DF = Decontamination factor.

TABLE C3. Categories of solid wastes according to activity concentration and half-life proposed by Gera (1968).

Category	Activity concentration, $\mu Ci/cm^3$ or $\mu Ci/g$		
	Group 1[a]	Group 2[b]	Group 3[c]
A	$<10^4$	$<10^5$	$<10^{-6}$
B	$10^{-4} - 10^{-1}$	$10^{-5} - 10^{-2}$	$10^{-6} - 10^{-3}$
C	$10^{-1} - 10^2$	$10^{-2} - 10$	$10^{-3} - 1$
D	$10^2 - 10^5$	$10 - 10^4$	$1 - 10^3$
E	$>10^5$	$>10^4$	$>10^3$

[a] $T_{1/2}$ <250 days, or $>10^{11}$ y.
[b] $T_{1/2}$ = 250 days to 10 y, or $10^8 - 10^{11}$ y.
[c] $T_{1/2}$ = $10 - 10^8$ y.

TABLE C4. Categories of solid wastes according to radiation index proposed by Gera (1968).

Category	Radiation Index $i = k^a D^b$
I	<0.002
II	0.002-0.02
III	0.02-0.2
IV	0.2-2
V	>2

[a]Surface area of waste item or package, perpendicular to the direction of measurement:

Area	k
<1 m^2	1
1-5 m^2	3
5-20 m^2	6
>20 m^2	19

Coefficient k reflects the different attenuation of exposure rate with the distance as a function of source dimensions; the values are taken from Regulations for the Safe Transport of Radioactive Materials, p. 69 (IAEA, 1967).

[b]D is exposure rate in air in R/hr, measured at a distance of 1 m from the waste.

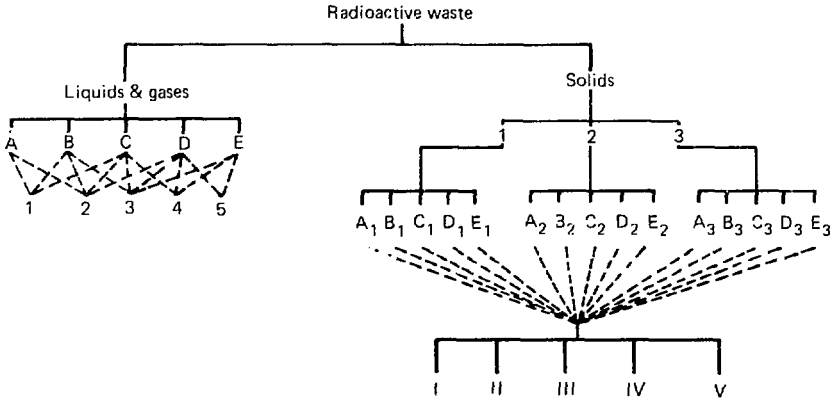

FIG. C4. Gera's 1968 waste classification system.

Classification System

Gera states that a succinct definition of the terminology in use today may constitute a sufficient classification system. His proposed definitions are as follows:

High-Level Waste. This category includes only wastes from the reprocessing of spent fuels or the spent fuels themselves. These wastes contain the bulk of the fission products and significant amounts of the long-lived alpha emitters. Liquid concentrations are in the thousands of curies per litre. Wastes are self-heating. Long-lived alpha emitters exist in concentrations too high to allow disposal for periods far exceeding the expected life of engineered storage structures.

Low-Level Waste. This category includes wastes with low hazard potential and with activity concentrations not much higher than MPC. Also included are large volumes that require no shielding and most of which may be treated with simple filtration or flocculation techniques, as well as solid wastes that can be handled without particular precautions.

Alpha-Bearing Waste. All wastes containing alpha emitters in excess of some appropriate threshold value are defined as alpha-bearing wastes; in particular, long-lived isotopes of transuranium elements.

Cladding Waste. This category includes cladding hulls and associated fuel assembly hardware from mechanical decladding of spent fuel. Most of the activation products and portions of the sorbed fission products and alpha emitters are present.

Intermediate-Level Waste. This includes all wastes not included in the preceding categories. These wastes cannot be released to the environment and usually require treatment and shielding, but not cooling.

The classification system requires a knowledge of the following waste characteristics: (1) source of the waste, (2) dominant radiation, and (3) general operational information.

It should be noted that the system (1) is simple and practical, and (2) provides guidance for the disposal of waste.

ANSI RADIOACTIVE WASTE CATEGORIES

To promote better communication on radioactive wastes between the nuclear energy industry and the public, the American National Standards Institute explicitly defined the terms used to describe broad categories of wastes.[C5] The basis for this system was a consideration of the sources of the waste in the fuel cycle and the physical nature of the radionuclides contained in the wastes. The waste categories specifically exclude (1) materials being stored for possible future recovery of radioactive contents of value, and (2) materials that are normally of use and are being stored for possible future removal of the radioactive contaminant(s). For liquid and gaseous wastes, the concepts used for specific concentrations are those of the Federal Radiation Council given in the Radioactivity Concentration Guide (RCG). Figure C5 diagrams the ANSI waste categories.

Waste Categories

The radioactive waste categories defined below include solid waste subcategories defined by parameters independent of possible mechanisms of exposure.

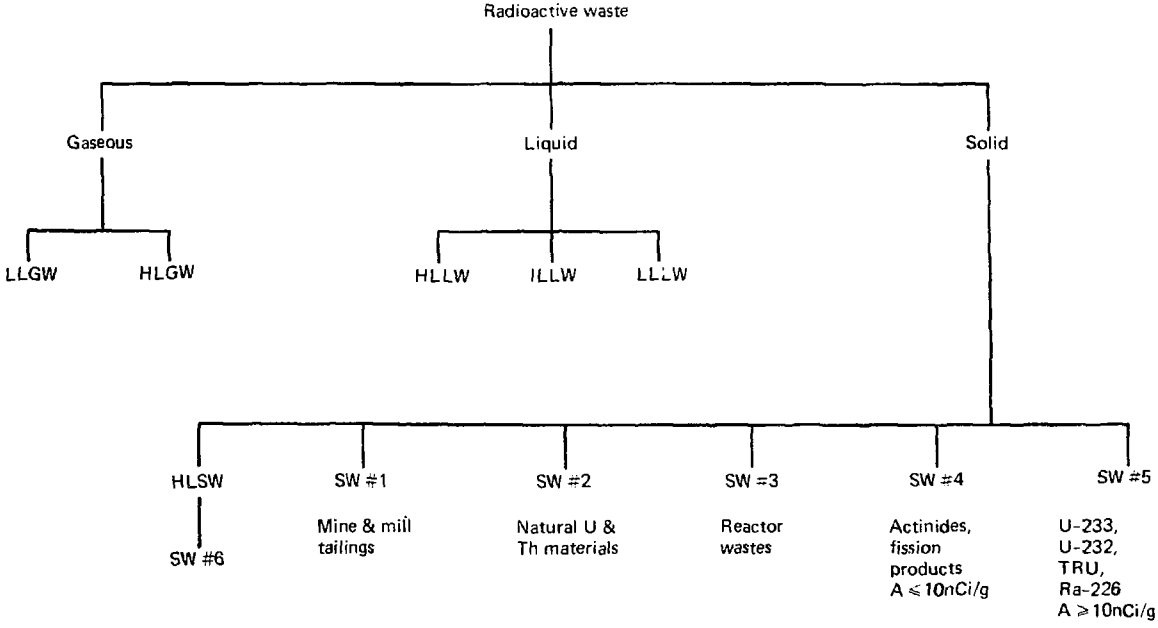

FIG. C5. American National Standards Institute waste categories.

High-Level Solid or Liquid Waste. Included in this category are the following:

1. Aqueous waste stream from first cycle of solvent extraction, including waste streams from subsequent cycles separate from or added to the first; solids or concentrates prepared from this first cycle and additions to it, where extraction cycles are intended for reprocessing (not including research reactors <1 MW).
2. Streams from partial separations where specific activity remains high.
3. Fuel elements from reactors (research reactors <1 MW).

Low-Level Liquid Waste. Wastes that after treatment (no more complex than filtration or ion exchange) can be released with reasonable assurance that the concentration in the water at the point of release will be less than the RCG value for the public (weighting individual RCG values by relative abundance of radionuclides present).

Intermediate-Level Liquid Waste. Liquid wastes not included in the high-level or low-level liquid waste categories. Wastes will eventually be treated to yield a processed fraction of low-level liquid wastes and a fraction that will be added to high-level liquid wastes.

Low-Level Gaseous Waste. Gaseous materials (including entrained or suspended volatile material, vapors, droplets, and particulate matter) which, after treatment no more complex than conventional filtration or scrubbing, can be released as specified in the low-level liquid waste category.

High-Level (Stored) Gaseous Waste. Gaseous or volatile materials not defined in the low-level gaseous waste category, which are stored either in gaseous form or sorbed in or on a solid medium.

Solid Waste. This category includes six groups:

1. Mine and mill tailings: tailings from mining or milling of uranium or thorium ores, in which daughter products are dispersed throughout the tailings so their concentration is no higher in any significant portion of a tailings pile than it was in original ores.

2. Natural uranium and thorium materials: solid waste with no radiation content exceeding the usual criteria for occupational radiation safety, except for natural uranium, enriched uranium, or thorium, which are the limiting potential hazard in the handling of the wastes.
3. Nuclear reactor waste: material normally nonradioactive containing no activity except that induced by neutron or other subatomic particle capture.
4. Material containing or contaminated with fission products or other radioactive materials not defined elsewhere, such that the concentration of selected actinides is less than the minimum concentration as stated in solid waste category 5.
5. Selected actinide waste. Material containing or contaminated with ^{233}U, ^{232}U, all transuranium radionuclides and ^{226}Ra at concentrations of $\geq$10 nCi/g (10 μCi/kg). (^{226}Ra is included with the actinides because of its comparable properties.)
6. Solid material included in the high-level solid- or liquid-waste category with the notation that relatively small amounts of such materials will not affect the status of solid waste category 4 or 5 (e.g., surface contamination on equipment that has been in contact with high-level wastes or samples removed for analytical control purposes).

Discussion

The following waste characteristics must be known to use this system to categorize any given waste:

- Physical state.
- Source of the waste.
- Activity concentrations (liquids and gases).
- Transuranic and ^{226}Ra concentrations.

In addition, it should be noted that the system (1) specifically excludes materials stored for future extraction of valuable material, and (2) does not provide guidance for the disposal of waste.

AEC RADIOACTIVE WASTE MANAGEMENT CLASSIFICATIONS

The AEC classifications[C6] are intended as definitions of wastes and as terminology for AEC-assigned responsibilities, authorities, and procedures for radioactive waste management. The definitions do not represent a rigorous, comprehensive classification system, but do show the work done by DOE's predecessor. Figure C6 is a diagram of the AEC waste classification system.

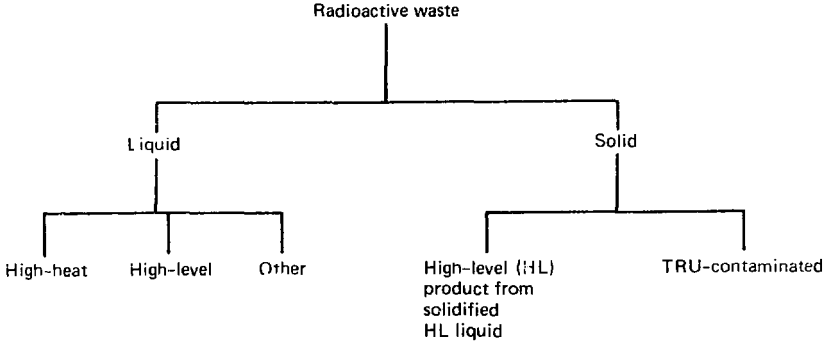

FIG. C6. AEC waste classification system.

Definitions

Radioactive wastes in general and liquid and solid wastes are defined as follows:

Radioactive Waste. Materials of no value consisting of, including, or contaminated with radioactive material in excess of the levels or concentrations permitted in AEC Property Management Instructions for unconditional release of excess property. These include: (1) stored liquid, solid, or gaseous residues from chemical or metallurgical processing of radioactive materials; (2) discarded items such as defective equipment and building rubble, not radioactive in themselves but contaminated with

radioactive materials; and (3) discarded items containing induced radioactivity. Treated as a separate category are (1) irradiated fuels stored for possible processing, (2) radioactive scrap stored for possible recovery of useful values, and (3) materials and equipment stored for possible future use following decontamination.

Liquid Radioactive Waste. Solutions, suspensions, and mobile sludges, contaminated with radioactive materials.

Solid Radioactive Waste. Material that is essentially dry but may contain sorbed radioactive fluids in sufficiently small amounts to be immobile when buried in dry soil.

Waste Classes

Classifications based on these definitions are specified for common terminology as follows:

High-Heat Liquid Waste. Liquid waste containing sufficient thermal energy to require some supplemental means of cooling, such as cooling coils.

High-Level Liquid Waste. The aqueous waste resulting from the operation of the first-cycle extraction system, or equivalent concentrated wastes from subsequent extraction cycles, or equivalent wastes from a process not using solvent extraction, in a facility for processing irradiated reactor fuels.

High-Level Waste. High-level liquid waste, or the products from solidification of high-level liquid waste, or irradiated fuel elements if discarded without processing.

Other Liquid Waste. Liquid waste, not within the definitions of high-level liquid waste.

Transuranium-Contaminated Solid Waste. Wastes contaminated with certain alpha-emitting radionuclides of long half-life and high specific radiotoxicity to greater than 10 nCi/g (10μCi/kg), subject to the following conditions and understandings:

1. The radionuclides included are ^{233}U (with its daughter products), plutonium, and transplutonium nuclides except ^{238}Pu and ^{241}Pu. (Note that ^{238}Pu and ^{241}Pu waste should be handled as transuranium-contaminated waste when so indicated by ^{239}Pu impurities or when required by local burial criteria.)
2. The activity density may be averaged over the contents of individual shipping containers, such as 55-gal drums, including materials added for shielding or sorption of liquids.

Discussion

To apply the AEC system, the following waste characteristics must be known:

- Physical state (solid, liquid).
- Thermal energy.
- Source of waste.
- Transuranic activity concentration.

It should be noted, also, that this system (1) considers physical state, (2) does not indicate the hazard potential, and (3) may be too restrictive.

BLOMEKE AND KEE WASTE DEFINITIONS

Blomeke and Kee provide a series of projections for the type and quantities of wastes that will be generated by the nuclear fuel cycle until approximately the year 2000.[C7]

Types of Waste

They identify and discuss the following types of radioactive wastes:

High-Level Wastes. These are composites of the liquid waste streams arising from the reprocessing of spent fuels. They contain more than 99.9% of the nonvolatile fission products, 0.5% of the uranium and plutonium, and all the other actinides formed by the transmutation of the uranium and plutonium in the reactors.

Cladding Wastes. These consist of solid fragments of Zircaloy and stainless steel cladding and other structural components of fuel assemblies that remain after the fuel cores have been dissolved.

Noble Gas Fission Products. These include krypton separated from reprocessing plant off-gas and packaged together with other collected noble gases.

Fission-Product Iodine. This is iodine whose radioactivity at the time of packaging and shipment is due solely to ^{129}I that has a 1.2×10^7 year half-life and is recovered at the reprocessing plant.

Tritium Wastes. These are wastes generated at nuclear power stations and reprocessing plants.

Carbon-14. This is carbon produced in oxide-fueled reactors principally by an (n,p) reaction with ^{14}N impurity in the fuels, but also as the product of an (n,α) reaction with ^{17}O.

Low-Level Transuranic Wastes. These are solid or solidified materials that contain plutonium or other long-lived alpha emitters in known or suspected concentrations greater than 10 nCi/g, and yet have sufficiently low external radiation levels after packaging that they can be handled directly.

Intermediate-Level Transuranic Waste. These solid or solidified materials contain long-lived alpha emitters at concentrations greater than 10 nCi/g, and have typical surface does rates of 10-1000 mrem/hr after packaging due fission-product contamination.

Nontransuranic Waste. This is waste composed of diverse materials contaminated with low levels of beta- and gamma-emitting isotopes, but containing less than 10 nCi of long-lived alpha activity per gram.

Ore Tailings. These are the results of mining and milling operations for the recovery of yellow cake, U_3O_8.

Discussion

The following properties of radioactive wastes must be specified when using the Blomeke and Kee classification system:

- Source of the waste.
- Physical properties of the waste.
- Activity concentration for long-lived α-emitters.

In addition, it should be noted that the system: (1) is clear-cut in its definitions of categories; (2) provides no guidance for the disposal of waste; and (3) gives little indication of relative hazard.

SUMMARY

The waste classification systems or methods reviewed provide useful insights on some of the important considerations for the further development of a suitable system. These considerations include:

- Degree of potential hazard.
- Measure of potential health hazards such as MPC.
- Physical state (solid, liquid, or gaseous).

Of equal emphasis and importance, but less often included in classification system, are the following considerations:

- Hazard duration; (half-life).
- Disposal options, containers, conditions, and permanence.
- Specific nuclide content.
- Specific emissions; alpha or beta-gamma emissions.
- Point of waste generation in the fuel cycle (i.e., waste stream of first cycle of reprocessing, hulls, etc.).

In many cases, the characteristics of radioactive waste included in the definitions of waste classes are overlapping or redundant.

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C1. International Atomic Energy Agency, Standardization of Radioactive Waste Categories (November, 1967).

C2. American Institute of Chemical Engineers, Proposed Definition of Radioactive Waste Categories (1967, since withdrawn).

C3. F. Gera, "The Classification of Radioactive Wastes," in Health Physics, Vol. 27 (Pergamon Press, 1974).

C4. G. Branca and F. Gera, Proposed Classification of Radioactive Waste, C.N.E.N. Internal Report, Rome (1968).

C5. American National Standards Institute, "Categories of Radioactive Waste: ANSI-N525" (Draft).

C6. U.S. Atomic Energy Commission, "Radioactive Waste Management," Part I, AEC Manual, Chapter 0511, Appendix 0511, (September 19, 1973). (Now a DOE Manual Chapter.)

C7. J. O. Blomeke and C. W. Kee, "Projections of Wastes to be Generated," International Symposium on the Management of Wastes from the LWR Fuel Cycle (July, 1976).

APPENDIX D

WC SYSTEMS PROPOSED BY TAP MEMBERS

INTRODUCTION

The classification systems presented in this appendix are those proposed by the panel members in the TAP's working sessions. A few of these systems were developed in some detail. Others represent the basic structure that a classification system might have, with only brief analysis of the details that may be necessary in applying such systems.

PROPOSED WASTE CLASSIFICATION SYSTEM 1

Description

The primary consideration in the development of waste classification system 1 (Rodger, 1975) is the method of waste disposal. The basic criterion applied is that the disposal method will provide reasonable protection to man and his environment.

The disposal methods are divided into four major categories:

Release of effluents. The release of radioactive wastes whose concentrations are in compliance or can be made to comply with guidelines given in 10CFR20[D1] for discharge directly to the environment.

Interim storage. The storage for an unspecified length of time of radioactive wastes to await radioactive decay; special treatment such as solidification, incineration, chemical change, heat dispersal, etc.; shipment, or any other operation that may require retention of the waste for some period of time.

Active confinement. Careful selection of sites where nuclear waste can be disposed of without significant harm to man and his environment. Shallow land burial would fall into this category.

Isolation. Methods that isolate the waste from man and his environment for long periods of time. Deep burial in geological formations, deep sea burial, and extraterrestrial disposal methods would fall in this category.

Discussion

All radioactive isotopes could be classified into two groups based on their potential hazard (see Fig. D1). These would be similar to the seven groups established for radioactive shipments as given in 49CFR173.389.* It was suggested that all radioisotopes that fall into transport [illegible] I and II be classified as Waste Group I. All other nuclides are classified as Waste Group II with the exception of ^{106}Ru, which should go into Waste Group I. This would place in G[illegible] I all the heavy element alpha emitters (including TRUs) plus hazardous be[illegible] emitters such as ^{90}Sr and ^{106}Ru.

In addition, it is proposed that concentration and total inventory limits be established for each individual site selected for active confinement. The limits for Waste Group I material would be more restrictive than those for Waste Group II material. The limits would be further defined as one limit for the immediate time and one limit for 100 y after use of the site. A period of 100 y is suggested because it seems reasonable that control of the site can be maintained for this length. It is also the time in which fission product nuclides decay significantly and TRU nuclides assume greater relative importance. The total inventory limits would be established after careful study of the site. Such factors as rainfall, subterranian water flow, ion exchange in the soil, and so on, would be used to establish both the immediate and the 100-y limits.

*Included as Appendix E in this report is Part 173.389 of Title 49 of the Code of Federal Regulations. This classification system is used for radioactive waste transportation purposes and has been quite satisfactory.

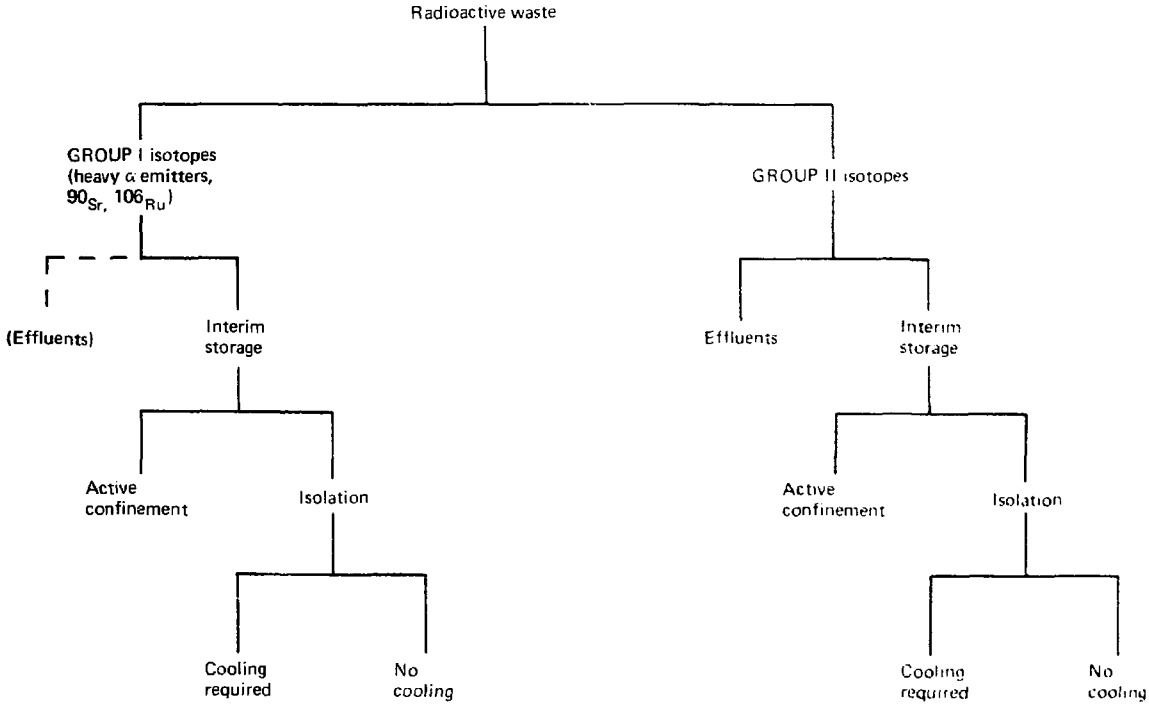

FIG. D1. Waste classification system 1.

On the other hand, the same concentration limits would be expected for all sites in the country. These limits should be established after careful study of the potential hazard, in the case of an accidental entry into the buried material. Some preliminary figures to illustrate the point are as follows.

Concentration limits for Waste Group I material could be:

1. For immediate burial -- 1.5 Ci/m^3.
2. For 100-y burial -- 0.15 Ci/m^3.

Concentration limits for Waste Group II material could be:

1. For immediate burial -- 15.0 Ci/m^3.
2. For 100-y burial -- 1.5 Ci/m^3.

For example, assume that a plant has a 55-gal drum (volume = 0.208 m^3) of waste containing 60% ^{90}Sr and 40% ^{106}Ru, by activity. The immediate burial limit of 1.5 Ci/m^3 implies that the drum may contain at burial:

$$^{90}Sr + {}^{106}Ru = (1.5\ Ci/m^3)\ (0.208\ m^3) = 0.312\ Ci \quad ,$$

with

$$(0.6)\ (0.312\ Ci) = 0.19\ Ci\ of\ {}^{90}Sr$$

and

$$(0.4)\ (0.312\ Ci) = 0.12\ Ci\ of\ {}^{106}Ru \quad .$$

The 100-y limit of 0.15 Ci/m^3 implies that the drum may contain at burial:

$$^{90}Sr\ (e^{-\lambda_1 t}) + {}^{106}Ru\ (e^{-\lambda_2 t}) = (0.15\ Ci/m^3)\ (0.208\ m^3) = 0.0312\ Ci \quad ,$$

where λ_1 and λ_2 are corresponding decay constants and t is time. Since at burial there is two-thirds as much ^{106}Ru as ^{90}Sr, then at burial:

$$^{90}Sr = \frac{0.0312 \text{ Ci}}{(e^{-\lambda_1 t}) + 2/3(e^{-\lambda_2 t})} = 0.34 \text{ Ci}$$

and

$$^{106}Ru = 2/3\ (0.35) = 0.23 \text{ Ci}\ .^{*}$$

Since the immediate burial limits are more restrictive in this case, the amount of ^{90}Sr and ^{106}Ru allowed per drum would be 0.19 Ci and 0.12 Ci, respectively.

For a mixture of plutonium isotopes, the governing criterion will be 100-y burial limits because of the long half-lives of the plutonium isotopes. This is obviously the case for ^{239}Pu with a 25,000-y half-life. For shorter lived ^{238}Pu ($T_{1/2}$ = 89 y), the following example leads to the same conclusion.

Assume that a plant has the same 55 gal drum containing waste with a composition of 80% ^{90}Sr and 20% ^{238}Pu, by activity. The immediate burial limit of 1.5 Ci/m^3 implies an activity limit of 0.25 Ci of ^{90}Sr and 0.062 Ci of ^{238}Pu. The 100-y burial limit of 0.15 Ci/m^3 implies activity limits of 0.15 Ci of ^{90}Sr and 0.038 Ci of ^{238}Pu. It can be seen that the 100-y limit is more restrictive in this case involving isotopes of longer half-lives, although both are less than 100 y.

For wastes in acceptable containers falling above these immediate and 100-y limits, isolation is required. Wastes destined for isolation from the environment should be divided into two subgroups depending on decay heat generation.

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*Note: In this example, the 100-y limit restricting the total activity of ^{106}Ru at burial is purely academic. Since ^{106}Ru has a half-life of 1 y, one could bury 7.5 × 10^{26} Ci of ^{106}Ru and at 100 y be within the activity limit for that drum.

The above scheme has merit from both a regulatory and an operational standpoint. Again, the limit figures used were somewhat arbitrary, and careful study would be needed to establish these limits.

PROPOSED WASTE CLASSIFICATION SYSTEM 2

Waste classification system 2 was developed to provide the maximum in utility with complete operational coverage.[D3] The system must satisfy the regulatory requirements and should not only be consistent with existing operational systems for hazardous materials, but should blend into them. The classification of a particular material should not change as it moves through an operational system unless the hazard associated with it changes. The system must be readily applicable to all radioactive materials.

Scope

The main consideration of the system was to serve the regulatory needs for radioactive wastes. There is a real need to move back into the nuclear fuel cycle in order to provide a system consistent with operation of the nuclear power plants. Wastes are generated at each point in the fuel cycle. The radioactive properties that define the hazard of radioactive wastes are also characteristic of many other materials in the fuel cycle, in the nuclear industry in general, and in the entire range of other activities using radioactive isotopes. Thus, the inclusion of these materials is virtually automatic in a system designed to meet operational needs by use of the properties and characteristics of radioactive nuclides.

Design Considerations

For the classification system to meet regulatory needs, it must relate the characteristics of the wastes to the hazards to be controlled. These characteristics include:

- Magnitude of the penetrating radiation.
- Hazard index.
- Life of the hazard.
- Mobility of the material.

In addition, the operational requirements should be recognized so that the operators may respond to the precautions implied by the classification.

The operations that will be affected include:

- Chemical processing operations.
- Handling operations.
- Emergency responses (e.g., during a fire).
- Transportation.
- Waste management--disposal or storage emplacement.

Detailed consideration of these factors leads to certain requirements or properties of the waste classification matrix. These are:

1. Simplicity--the system's classes must be easily recognized and the action implied by the classification readily defined and implemented without ambiguity.
2. Usefulness--the system must provide information needed in operations and regulations.
3. Comprehensiveness--the system must include all hazardous radioactive materials.
4. Flexibility--the system should permit changes in the classification of a material whose hazard has changed, and it should also be able to incorporate changes required by revisions in regulations, policy, and hazard assessment, or by new technology.
5. Multilevel requirements--since different properties differ in operational importance for different operations, it is not necessary to have all properties identified for each. A simpler system for these operations can be provided by including different levels of classification.
6. Precise definition--each class will be designated by a roman numeral and each subclass by one or more letters.

Primary Categories

The primary categories provide operating and control information needed in the conduct of each of the operations. These relate mainly to the handling

requirements and are distinguished by the need for cooling, shielding, and coping with an existing biological hazard. The most intense radioactive materials are those that require cooling. All such materials require shielding and present a large biological hazard. The next less hazardous materials are those requiring shielding but no cooling. The next less hazardous materials do not require shielding but contain materials that present a radioactive hazard if ingested or inhaled. These categories are summarized in Table D1.

TABLE D1. WC system 2 categories.

Category	Requirements	Criteria
I	Requires cooling and shielding	400 W/m^2
II	Requires shielding	200 mR/hr at surface; 10 mR/hr at 0.9 m
III	Requires no shielding	Would exceed MPC if released
IV	Requires no care	(Innocuous)

An additional category has been suggested for materials that would be diluted below MPC in leaving the controlled area. This category, if adopted, would become IV, and "Innocuous" would become V.

Subcategories

Subcategories are needed to delineate the other factors and properties of concern. One of the most important of these relates to the duration of the hazard. The hazards due to radioactive materials are complex, depending on many physical, chemical, and physiological phenomena, but they have been quantitatively defined by other authorities whose definitions we may use.

We need consider only the changes in hazard occurring with time. Each isotope decays in a precisely defined manner so that the rate of decrease or growth of each radioisotope and its attendant hazard can be readily calculated. Fairly accurate approximations for certain groups of isotopes, such as those from the fission process, are also available. We then define a subcategory that classifies the wastes according to the time the material stays within a category. When this time is exceeded, the waste then drops into the next lower category. This particular subcategory has a special meaning for waste management regulations in that it is probably the principal criterion for distinguishing between wastes that may require disposal with assured permanent isolation from the biosphere and those that may safely be disposed of with less severe restrictions. The time required for the radioactive hazard to decrease to acceptable levels, if short with respect to the anticipated duration of effective management control, would permit the less restricted emplacement system. Radioactive materials that would exist as serious hazards for times that are long compared with the anticipated duration of effective management control would probably be placed in geologic isolation or in some other system isolated from the biosphere. Note that the transuranics, when treated as wastes, would be included in this latter subcategory. Other long-lived isotopes, however, may also be included, even some with "infinite" half-lives (i.e., the stable isotopes), if sufficiently toxic.

The next subcategory defines the physical state or mobility of the waste material, and the final subcategory determines whether or not treatment is required for some regulatory consideration. These subcategories are summarized in Table D2.

Waste Classification

A symbol consisting of several letters can be used to identify the various categories and subcategories. These are summarized in Table D3.

TABLE D2. WC system 2 subcategories.

Symbol	Meaning	Criteria
a		<10 y
b	Lifetime for changing	10-100 y
c	category.[a]	100-1000 y
d		>1000 y
s	Solid	Immobile
m	Mobile	Will disperse or become airborne if container is spilled
l	Liquid	Will flow onto a flat surface from a small hole
g	Gaseous	Will mix with the atmosphere through a small hole
t	Treatment	Must be converted to another form before further processing
n	No treatment	---
e	Explosive	Not chemically stable
f[b]	Fissile	Special nuclear materials

[a]An alternative definition could be "lifetime until waste leaves category III (becomes innocuous)."

[b]May be added if desired for completeness, but waste materials other than the transuranics should have no significant fissile content.

TABLE D3. WC system 2 symbols and meanings.

Category	Symbol (Time / Physical State / Need for treatment / Explosive nature)	Description
I	a s t e b m n - c l d g	Any material with sufficient radioactivity to require cooling. The time required before it becomes type II is shown by a, b, c, or d. The physical state is shown as s, m, l, or g. The need for treatment is t or n. The explosive nature is blank-if chemically inert, e if a potential hazard.
II	a s t e b m n - c l d g	Any material with sufficient radioactivity to require shielding. The other elements are as described above.
III	a s t e b m n - c l d g	Requires control but no shielding - a, b, c, or d now become the consideration for complete release. Other elements as described.
IV		A material without hazard.

How wastes would be classified in this system can be illustrated by the following examples. Liquid waste from fuel reprocessing, which includes the unburned transuranics, as it is stored on site for eventual solidification, would be classified:

I (a, l, t) → remains in category I until 1985.

II (c, l, t) → remains in category II until 2100-3000.

III (d, l, t) → remains in category III indefinitely.

Upon solidification, this waste becomes:

I (a, s, n), etc.

Filters containing only ^{131}I would be classified as:

II (a, s, n).

A number of facets of the suggested system are arbitrary and can be adjusted to include other considerations or to improve overall consistency.

Figure D2 is a simplified diagram of waste cassification system 2.

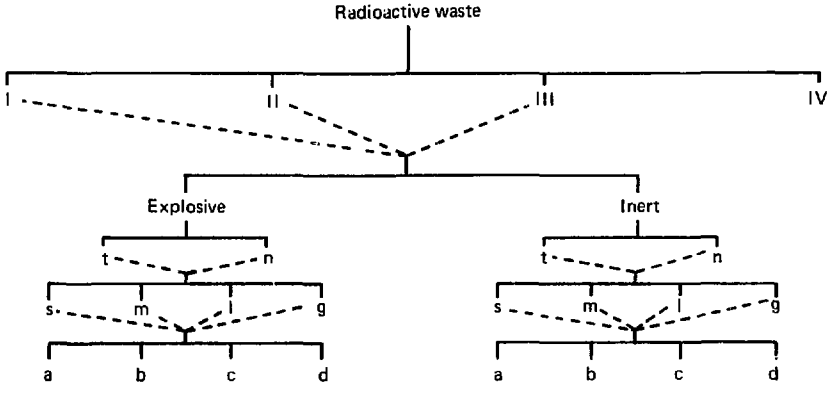

FIG. D2. Waste classification system 2.

PROPOSED WASTE CLASSIFICATION SYSTEM 3

Briefly outlined, the following simple classification system in based on thermal and radiation properties of the wastes (see Fig. D3).

Waste property		Current waste type
Thermal	Radiation	
Hot	Hot	HLW
Cool	Hot	Hulls
Cool	Cool	LLW

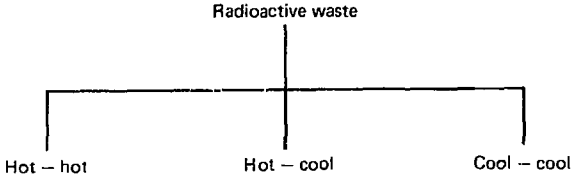

FIG. D3. Waste classification system 3.

PROPOSED WASTE CLASSIFICATION SYSTEM 4

The following format and criteria suggest a WC system based on environmental criteria (see Fig. D4).

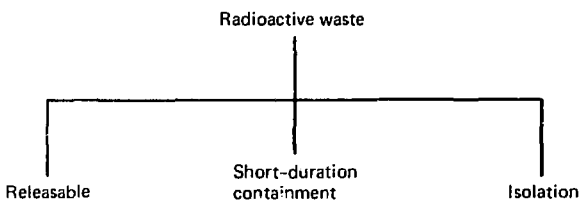

FIG. D4. Waste classification system 4.

Releasable

These are waste-satisfying conditions less than those set by "lower bound" criteria. These may be directly released as nonhazardous wastes such as normal trash, garbage, or sewage without further treatment.

Short-Duration Containment

These wastes satisfy the following conditions:

1. Criteria greater than those set by "lower bound" criteria where these criteria might be developed on the basis of: (a) comparision with natural deposits and acceptable natural-hazard limits and compliance with ALARA principles; (b) $1000 per man-rem (10 CFR 50, Appendix I)[D4]; and (c) MPC values less than 10 CFR 20[D1] MPC limits.
2. Wastes with hazard durations less than those specified by "middle bound" criteria.

Isolation

These wastes satisfy conditions greater than those set by "lower bound" criteria but with hazard durations greater than those set by "middle bound" criteria, which might be developed on the basis of:

1. Inventory commitment, as discussed in WASH-1539.[D5]
2. Age between 100 and 1000 y (Rodger, 1975).[D6]

PROPOSED WASTE CLASSIFICATION SYSTEM 5

Early in the tenure of the TAP, a WC system based primarily on two disposal modes was suggested. Wastes would be classified first as (1) those to be disposed of in shallow land burial sites, and (2) those to be disposed of in deep geologic formations.

Wastes destined for deep geologic disposal would be further categorized as high-heat or low-heat wastes. Low-heat wastes should be further classified as those requiring shielding and those not requiring shielding.

REFERENCES

D1. Code of Federal Regulations, Standards for Protection against Radiation, Title 10, Part 20 (10 CFR 20), U.S. Nuclear Regulatory Commission (Nov. 1960).

D2. Code of Federal Regulations, Title 49, Part 173.389 (49 CFR 173.389), "Transportation--Radioactive Materials; Definitions," General Services Administration (1975).

D3. M. Karol and R. Post, Comprehensive Operational Waste Classification System, University of Arizona, Tuscon, Ariz., EES Series, Report No. 48 (1977).

D4. Code of Federal Regulations, Licensing of Production and Utilization Facilities, Title 10, Part 50 (10 CFR 50), Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation To Meet the Criterion 'As Low as Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents" (Dec. 1975).

D5. Management of Commercial High Level and Transuranium Contaminated Radioactive Waste, Environmental Statement, USAEC Report WASH-1539 (Sept. 1974).

D6. W. A. Rodger, Critical Evaluation of the Limits of Transuranic Contamination of Low Level Waste, Nuclear Safety Associates, Bethesda, Maryland (Aug. 1975).

§ 173.389 Radioactive materials; definitions.

For the purpose of Parts 170–189 of this chapter:

(a) "Fissile radioactive material" means the following material: Plutonium-238, plutonium-239, plutonium-241, uranium-233, or uranium 235, or any material containing any of the foregoing materials. See § 173.396(a) for exclusions. Fissile radioactive material packages are classified according to the controls needed to provide nuclear criticality safety during transportation as follows:

(1) *Fissile Class I.* Packages which may be transported in unlimited numbers and in any arrangement, and which require no nuclear criticality safety controls during transportation. For purposes of nuclear criticality safety control, a transport index is not assigned to Fissile Class I packages. However, the external radiation levels may require a transport index number.

(2) *Fissile Class II.* Packages which may be transported together in any arrangement but in numbers which do not exceed an aggregate transport index of 50. For purposes of nuclear criticality safety control, individual packages may have a transport index of not less than 0.1 and not more than 10. However, the external radiation levels may require a higher transport index number but not to exceed 10. Such shipments require no nuclear criticality safety control by the shipper during transportation.

(3) *Fissile Class III.* Shipments of packages which do not meet the requirements of Fissile Class I or II and which are controlled to provide nuclear criticality safety in transportation by special arrangements between the shipper and the carrier.

Note 1: Uranium-235 exists only in combination with various percentages of uranium-234 and uranium-238. "Fissile radioactive material" as applied to uranium-235 refers to the amount of uranium-235 actually contained in the total quantity of uranium being transported.

Note 2: Radioactive material may consist of mixtures of fissile and non-fissile radionuclides. "Fissile radioactive material" refers to the amount of plutonium-238, plutonium-239, plutonium-241, uranium-233, uranium-235, or any combination thereof actually contained in the mixture. The "radioactivity" of the mixture consists of the total activity of both the fissile and nonfissile radionuclides. All mixtures containing "fissile material" shall be subject to § 173.396.

(b) "Large quantity radioactive materials" means a quantity the aggregate radioactivity of which exceeds that specified as follows:

(1) Groups I or II (see paragraph (h) of this section) radionuclides: 20 curies.

(2) Groups III or IV radionuclides: 200 curies.

(3) Group V radionuclides: 5,000 curies.

(4) Groups VI or VII radionuclides: 50,000 curies.

(5) Special form material: 5,000 curies.

(c) "Low specific activity material" means any of the following:

(1) Uranium or thorium ores and physical or chemical concentrates of those ores;

(2) Unirradiated natural or depleted uranium or unirradiated natural thorium;

(3) Tritium oxide in aqueous solutions provided the concentration does not exceed 5 millicuries per milliliter;

(4) Material in which the activity is essentially uniformly distributed and in which the estimated average concentration per gram of contents does not exceed:

(i) 0.0001 millicuries of Group I (see § 173.389(h)) radionuclides; or

(ii) 0.005 millicuries of Group II radionuclides; or

(iii) 0.3 millicuries of Groups III or IV radionuclides.

Note: This includes, but is not limited to, materials of low radioactivity concentration such as residues or solutions from chemical processing; wastes such as building rubble, metal, wood, and fabric scrap, glassware, paper and cardboard; solid or liquid plant waste, sludges, and ashes.

(5) Objects of nonradioactive material externally contaminated with radioactive material, provided that the

radioactive material is not readily dispersible and the surface contamination when averaged over an area of 1 square meter, does not exceed 0.0001 millicurie (220,000 disintegrations per minute) per square centimeter of Group I radionuclides or 0.001 millicurie (2,200,000 disintegrations per minute) per square centimeter of other radionuclides.

(d) "Normal form radioactive materials" means those which are not special form radioactive materials. Normal form radioactive materials are grouped into transport groups (see paragraph (h) of this section).

(e) "Radioactive material" means any material or combination of materials, which spontaneously emits ionizing radiation. Materials in which the estimated specific activity is not greater than 0.002 microcuries per gram of material, and in which the radioactivity is essentially uniformly distributed, are not considered to be radioactive materials.

(f) "Removable radioactive contamination" means radioactive contamination which can be readily removed in measurable quantities by wiping the contaminated surface with an absorbent material. The measurable quantities shall be considered as being not significant if they do not exceed the limits specified in § 173.397.

(g) "Special form radioactive materials" means those which, if released from a package, might present some direct radiation hazard but would present little hazard due to radiotoxicity and little possibility of contamination. This may be the result of inherent properties of the material (such as metals or alloys), or acquired characteristics, as through encapsulation. The criteria for determining whether a material meets the definition of special form are prescribed in § 173.393(a).

(h) "Transport group" means any one of seven groups into which normal form radionuclides are classified according to their radiotoxicity and their relative potential hazard in transportation, and as listed in § 173.390.

(i) "Transport index" means the number placed on a package to designate the degree of control to be exercised by the carrier during transportation. The transport index to be assigned to a package of radioactive materials shall be determined by either subparagraph (1) or (2) of this paragraph, whichever is larger. The number expressing the transport index shall be rounded up to the next highest tenth; e.g., 1.01 becomes 1.1.

(1) The highest radiation dose rate, in millirem per hour at three feet from any accessible external surface of the package; or

(2) For Fissile Class II packages only, the transport index number calculated by dividing the number "50" by the number of similar packages which may be transported together (see § 173.396), as determined by the procedures prescribed in the regulations of the U.S. Atomic Energy Commission, Title 10, Code of Federal Regulations, Part 71.

(j) "Type A packaging" means packaging which is designed in accordance with the general packaging requirements of §§ 173.24 and 173.393, and which is adequate to prevent the loss or dispersal of the radioactive contents and to retain the efficiency of its radiation shielding properties if the package is subject to the tests prescribed in § 173.398(b).

(k) "Type B packaging" means packaging which meets the standards for Type A packaging, and, in addition, meets the standards for hypothetical accident conditions of transportation as prescribed in § 173.398(c).

(l) "Type A quantity" and "Type B quantity" radioactive materials means a quantity the aggregate radioactivity of which does not exceed that specified as follows:

Transport group (see § 173.389(h))	Type A quantity (in curies)	Type B quantity (in curies)
I	0.001	20
II	0.05	20
III	3	200
IV	20	200
V	20	6,000
VI and VII	1,000	50,000
Special form	20[1]	6,000

[1] Except that for Californium-252 the Type A quantity limit for special form is 2 curies.

(m) *Containment system.* Containment system of a radioactive materials package means those components of the packaging including special form encapsulation where used, which have been specified by the package designer as intended to retain the radioactive contents during transport, whether or not individual vessels in the packaging retain their integrity of containment.

(n) *Maximum normal operating pressure.* Maximum normal operating pressure means the maximum pressure above atmospheric pressure at mean sea level

that would develop in the containment system in a period of 1 year, under the conditions of temperature and solar radiation corresponding to environmental conditions of transport in the absence of venting, external cooling by an ancillary system, or operational controls during transport.

[Amdt. 173-3, 33 F.R. 14022, Oct. 4, 1968; 33 F.R. 19823, Dec. 27, 1968, as amended by Amdt. 173-64, 37 F.R. 14589, July 21, 1972; Amdt. No. 173-60, 37 F.R. 17970, Sept. 2, 1972]

§ 173.390 Transport groups of radionuclides.

(a) List of radionuclides:

Element[1]	Radionuclide[2]	Transport group						
		I	II	III	IV	V	VI	VII
Actinium (89)	Ac-227	X						
	Ac-228	X						
Americium (95)	Am-241	X						
	Am-243	X						
Antimony (51)	Sb-122				X			
	Sb-124			X				
	Sb-125			X				
Argon (18)	Ar-37						X	
	Ar-41		X					
	Ar-41 (uncompressed)[3]					X		
Arsenic (33)	As-73				X			
	As-74				X			
	As-76				X			
	As-77				X			
Astatine (85)	At-211			X				
Barium (56)	Ba-131				X			
	Ba-133		X					
	Ba-140			X				
Berkelium (97)	Bk-249	X						
Beryllium (4)	Be-7				X			
Bismuth (83)	Bi-206				X			
	Bi-207			X				
	Bi-210		X					
	Bi-212			X				
Bromine (35)	Br-82				X			
Cadmium (48)	Cd-109				X			
	Cd-115m			X				
	Cd-115				X			
Calcium (20)	Ca-45				X			
	Ca-47				X			
Californium (98)	Cf-249	X						
	Cf-250	X						
	Cf-252	X						
Carbon (6)	C-14				X			
Cerium (58)	Ce-141				X			
	Ce-143				X			
	Ce-144			X				
Cesium (55)	Cs-131				X			
	Cs-134m			X				
	Cs-134			X				
	Cs-135				X			
	Cs-136				X			
	Cs-137			X				
Chlorine (17)	Cl-36			X				
	Cl-38				X			
Chromium (24)	Cr-51				X			
Cobalt (27)	Co-58			X				
	Co-57				X			
	Co-58m				X			
	Co-58				X			
	Co-60			X				
Copper (29)	Cu-64				X			
Curium (96)	Cm-242	X						
	Cm-243	X						
	Cm-244	X						
	Cm-246	X						
	Cm-240	X						
Dysprosium (66)	Dy-154			X				
	Dy-165				X			
	Dy-166				X			
Erbium (68)	Er-169				X			
	Er-171				X			
Europium (63)	Eu-150			X				
	Eu-152m				X			
	Eu-152			X				
	Eu-154		X					
	Eu-155				X			
Fluorine (9)	F-18				X			
Gadolinium (64)	Gd-153				X			
	Gd-159				X			
Gallium (31)	Ga-67			X				
	Ga-72				X			

See footnotes at end of table.

Element [1]	Radionuclide [2]	Transport group						
		I	II	III	IV	V	VI	VII
Germanium (32)	Ge-71				X			
Gold (79)	Au-193			X				
	Au-194			X				
	Au-195			X				
	Au-198				X			
	Au-199				X			
	Au-199				X			
Hafnium (72)	Hf-181				X			
Holmium (67)	Ho-166				X			
Hydrogen (1)	H-3 (see tritium)							
Indium (49)	In-113m				X			
	In-114m			X				
	In-115m				X			
	In-115				X			
Iodine (53)	I-124			X				
	I-125			X				
	I-126			X				
	I-129			X				
	I-131			X				
	I-132				X			
	I-133			X				
	I-134				X			
	I-135				X			
Iridium (77)	Ir-190				X			
	Ir-192			X				
	Ir-194				X			
Iron (26)	Fe-55				X			
	Fe-59				X			
Krypton (36)	Kr-85m			X				
	Kr-85m (uncompressed) [3]					X		
	Kr-85			X				
	Kr-85 (uncompressed) [3]						X	
	Kr-87		X					
	Kr-87 (uncompressed) [3]					X		
Lanthanum (57)	La-140				X			
Lead (82)	Pb-203				X			
	Pb-210		X					
	Pb-212		X					
Lutecium (71)	Lu-172			X				
	Lu-177				X			
Magnesium (12)	Mg-28			X				
Manganese (25)	Mn-52				X			
	Mn-54				X			
	Mn-56				X			
Mercury (80)	Hg-197m				X			
	Hg-197				X			
	Hg-203				X			
Mixed Fission Products	MF-P		X					
Molybdenum (42)	Mo-99				X			
Neodymium (60)	Nd-147				X			
	Nd-149				X			
Neptunium (93)	Np-237	X						
	Np-239	X						
Nickel (28)	Ni-59			X				
	Ni-63				X			
	Ni-65				X			
Niobium (41)	Nb-93m				X			
	Nb-95				X			
	Nb-97				X			
Osmium (76)	OS-185				X			
	OS-191m				X			
	OS-191				X			
	OS-193				X			
Palladium (46)	Pd-103				X			
	Pd-109				X			
Phosphorus (15)	P-32				X			
Platinum (78)	Pt-191				X			
	Pt-193				X			
	Pt-193m				X			
	Pt-197m				X			
	Pt-197				X			
Plutonium (94)	Pu-238 [4]	X						
	Pu-239 [4]	X						
	Pu-240	X						
	Pu-241 [4]	X						
	Pu-242	X						
Polonium (84)	Po-210	X						
Potassium (19)	K-42				X			
	K-43			X				
Praseodymium (59)	Pr-142				X			
	Pr-143				X			

See footnotes at end of table.

Element [1]	Radionuclide [2]	Transport group						
		I	II	III	IV	V	VI	VII
Promethium (61)	Pm-147				X			
	Pm-149				X			
Protactinium (91)	Pa-230	X						
	Pa-231	X						
	Pa-233		X					
Radium (88)	Ra-223		X					
	Ra-224		X					
	Ra-226	X						
	Ra-228	X						
Radon (86)	Rn-222				X			
	Rn-220		X					
Rhenium (75)	Re-183				X			
	Re-184				X			
	Re-187				X			
	Re-189				X			
	Re Natural				X			
Rhodium (45)	Rh-103m				X			
	Rh-105				X			
Rubidium (37)	Rb-83				X			
	Rb 87				X			
	Rb Natural				X			
Ruthenium (44)	Ru 97				X			
	Ru-103				X			
	Ru-105				X			
	Ru-106			X				
Samarium (62)	Sm-145			X				
	Sm-147			X				
	Sm-151				X			
	Sm-153				X			
Scandium (21)	Sc-46			X				
	Sc-47				X			
	Sc-48				X			
Selenium (34)	Se-75				X			
Silicon (14)	Si-31				X			
Silver (47)	Ag-105				X			
	Ag-110m			X				
	Ag-111				X			
Sodium (11)	Na-22			X				
	Na-24				X			
Strontium (38)	Sr-85m				X			
	Sr-85				X			
	Sr-89			X				
	Sr-90		X					
	Sr-91			X				
	Sr-92				X			
Sulphur (16)	S-35				X			
Tantalum (73)	Ta-182			X				
Technetium (43)	Tc-95m				X			
	Tc-96				X			
	Tc-96m				X			
	Tc-97				X			
	Tc-97m				X			
	Tc-99				X			
Tellurium (52)	Te-125m				X			
	Te-127m				X			
	Te-127				X			
	Te-129m			X				
	Te-129				X			
	Te-131m			X				
	Te-132				X			
Terbium (65)	Tb-160			X				
Thallium (81)	Tl-200				X			
	Tl-201				X			
	Tl-202				X			
	Tl-204			X				
Thorium (90)	Th-227		X					
	Th-228	X						
	Th-230	X						
	Th-231	X						
	Th-232			X				
	Th-234		X					
	Th Natural			X				
Thulium (69)	Tm-168			X				
	Tm-170			X				
	Tm-171				X			
Tin (50)	Sn-113				X			
	Sn-117m			X				
	Sn-121			X				
	Sn-125				X			
Tritium (1)	H-3				X			
	H-3 (as a gas, as luminous paint, or adsorbed on solid material).							X

See footnotes at end of table.

Element[1]	Radionuclide[2]	Transport group						
		I	II	III	IV	V	VI	VII
Tungsten (74)	W-181				X			
	W-185				X			
	W-187				X			
Uranium (92)	U-230		X					
	U-232	X						
	U-233[3]		X					
	U-234		X					
	U-235[4]			X				
	U-238		X					
	U-natural			X				
	U-enriched[5]			X				
	U-depleted			X				
Vanadium (23)	V-48				X			
	V-49			X				
Xenon (54)	Xe-123			X				
	Xe-131m			X				
	Xe-131m (uncompressed)[3]					X		
	Xe-133			X				
	Xe-133 (uncompressed)[3]						X	
	Xe-135		X					
	Xe-135 (uncompressed)[3]						X	
Ytterbium (70)	Yb-175				X			
Yttrium (39)	Y-88			X				
	Y-90				X			
	Y-91m			X				
	Y-91			X				
	Y-92				X			
	Y-93				X			
Zinc (30)	Zn-65				X			
	Zn-69m				X			
	Zn-69				X			
Zirconium (40)	Zr-93				X			
	Zr-95			X				
	Zr-97				X			

[1] Atomic number shown in parentheses.
[2] Uncompressed means at a pressure not exceeding 14.7 p.s.i. (absolute).
[3] Atomic weight shown after the radionuclide symbol.
[4] Fissile radioactive material.

(b) Any radionuclide not listed in the above table shall be assigned to one of the groups in accordance with the following table:

Radionuclide	Radioactive half-life		
	0–1,000 days	1,000 days to 10^4 years	Over 10^4 years
Atomic number 1–81	Group III	Group II	Group III.
Atomic number 82 and over	Group I	Group I	Do.

NOTE 1: No unlisted radionuclides shall be assigned to Groups IV, V, VI, or VII.

(c) For mixtures of radionuclides the following shall apply:

(1) If the identity and respective activity of each radionuclide are known, the permissible activity of each radionuclide shall be such that the sum, for all groups present, of the ratio between the total activity for each group to the permissible activity for each group will not be greater than unity.

(2) If the groups of the radionuclides are known but the amount in each group cannot be reasonably determined, the mixture shall be assigned to the most restrictive group present.

(3) If the identity of all or some of the radionuclides cannot be reasonably determined, each of these unidentified radionuclides shall be considered as belonging to the most restrictive group which cannot be positively excluded.

(4) Mixtures consisting of a single radioactive decay chain where the radionuclides are in the naturally occurring proportions shall be considered as consisting of a single radionuclide. The group and activity shall be that of the first member present in the chain, except if a radionuclide "x" has a half-life longer than that of that first member and an activity greater than that of any other member including the first at any time during transportation; in that case, the transport group of the nuclide "x" and the activity of the mixture shall be the maximum activity of that nuclide "x" during transportation.

[Amdt. 173–3, 33 F.R. 14923, Oct. 4, 1968]

APPENDIX F

CHARACTERISTICS OF EXISTING LOW LEVEL WASTE (LLW) DISPOSAL SITES

The current practice in the United States is to dispose of solid LLW mainly by shallow land burial. Hence, the reference containment facility (RCF) used in this study to analyze the degree of confinement given LLW after disposal is based on that practice. The parameters chosen to describe the RCF came from two sources. One was the measured values of those parameters at existing facilities; the other was conservative estimates of the values to be allowed at any future burial sites.

To gain a perspective on current LLW handling and provide a background for judging the reasonability of the RCF, we reviewed existing low-level waste burial facilities.[F1-F7] Six commercial* LLW burial facilities and five major active sites for burial of defense- and research-related radioactive wastes at DOE facilities now exist in the United States. Sources of the wastes may differ from site to site, but operational characteristics of the disposal facilities and waste compositions generally are similar.

LLW received at the disposal facilities are placed directly in pits or trenches excavated in the native soil or till at the site. Overburden removed during excavation is used to cover the wastes. Pits and trenches are sloped, and the cover is applied to control ground water and surface runoff from precipitation. Characteristics of the existing sites are summarized in tables F1 through F5.

Table F1 gives the capacity of the commercial sites, the sizes of trenches or pits, the cover, fill procedures, covering frequency, and provisions for water. Table F2 gives waste inventories.

*Three sites were not receiving waste at the time of publication.

Table F3 shows climatological information for each site, including data on climate, precipitation, geomorphology, permeability, and bedrock. Table F4 presents hydrogelogic site information including depth to aquifer zones, nearest surface water and water flow paths, radionuclide migration, downstream river flow rate, and seismic hazard zones. Table F5 lists demographic information, including downstream populations and distances.

TABLE F1. Capacities, covers, and water collection provisions at existing LLW sites.

Site	Commercial site capacity, 10^5 m^3 (Ref. E2)	Burial trench size (length × width × depth), m (Ref. E6)	Cover		Fill procedure	Covering frequency	Provisions for water collection and containment (Ref. E6)
			Type	Depth			
Hanford, Washington[a]	----	Variable × 1.5-5 × 4-8	Mounded earthfill	Min. 2.5 m, or to reduce to <1 mR/hr at surface	Filled from end	Daily	None
Richland, Washington	9	90 × 8 × 6	Earthfill	Min. 2 m total; mounded to 1 m above grade	Trench filled to 0.6 m of surface	As trench is filled	None
Beatty, Nevada	7	260 × 12-15 × 8	Earthfill	Min. 2 m total; mounded to 0.6 m above grade	Trench filled to 1 m of surface	As trench is filled	None
Inel, Idaho[a]	----	275 × 2-3 × 4	Reseeded earthfill	Min. 1 m to surface	Pits and trenches filled to 1 m of surface	As trench or pit is filled	None
Los Alamos, New Mexico[a]	----	120-180 × 9-30 × 8	Excavated tuff compacted	Min. 1.5 m; mounding 0.5-1 m above grade	Layered filling to 1 m of surface	Combustibles on day of delivery, other as required	None
Sheffield, Illinois[b]	2	150 × 15-18 × 6-8	Compacted clay reseeded	Min. 1 m final cover	Trench filled to 0.6 m of surface	Daily	Trenches sloped; sump and standpipe
Morehead, Kentucky[b]	31	60-150 × 24 × 6-8	1 m compacted clay; mounded; reseeded	1 m cover; mounded 0.6 m above grade	Trench filled to 0.6 m of surface	Daily	Trenches sloped, standpipe; may build berm around trench
Oak Ridge, Tennessee[a]	----	15 × 3 × 3-5	Reseeded earthfill	Min. 1 m to surface	Trench filled to 1 m of surface	As trench is filled	Trenches sloped; monitoring wells
Savannah River, South Carolina[a]	----	Variable × 6 × 6	Mounded earthfill	Min. 1.2 m, or to reduce to <6 mR/hr at surface	Random placement in trenches	After disposal	Monitoring wells
Barnwell, South Carolina	25	140 × 15 × 5-7	0.6 m clay additional mounded cover	3 m at centerline 1.5 m at trench edge	Trench filled to 1 m of surface	Daily	Trenches sloped 1°; sand at trench bottom
West Valley, New York[b]	2	180-210 × 10 × 6	Earthfill compacted topsoil added	Min. 3 m; mounded 1.5 m above grade	Trench filled to grade level	Daily	Trenches sloped 2°; sump with riser pipe
Values used in RCF	6	200 × 10 × 8	Mounded earthfill	1-2 m mounded	Filled to 1 m of surface	Daily	Trenches sloped; sump and standpipe

[a]DOE site
[b]Not receiving waste at time of publication.

TABLE F2. Waste inventories at existing LLW sites.

Site	Byproduct material Ci nondecayed	Special nuclear material, g	Source material, kg	Transuranics, g	Volume buried, 10^4 m^3	Cumulative through date
Hanford, Washington[a]	8.1×10^5	----	6.0×10^5	3.65×10^5	20.0	7/75
Richland, Washington	5.4×10^5	5.8×10^5	1.3×10^4	2.3×10^4	1.4	1/77
Beatty, Nevada	1.3×10^5	1.8×10^5	5.5×10^4	1.4×10^4	5.6	1/77
Inel, Idaho[a]	3.6×10^6	----	2.88×10^5	3.61×10^5	4.0	7/75
Los Alamos, New Mexico[a]	1.6×10^5	----	2.51×10^5	1.3×10^3	23.0	7/75
Sheffield, Illinois	4.6×10^4	4.8×10^4	1.9×10^5	1.3×10^4	6.8	1/77
Morehead, Kentucky	2.3×10^6	4.0×10^5	2.3×10^5	6.9×10^4	14.0	1/77
Oak Ridge, Tennessee[a]	$<6 \times 10^4$	----	100	1.3×10^3	18.0	7/75
Savannah River, South Carolina[a]	4.4×10^6	----	7.9×10^4	7×10^3	20.0	7/75
Barnwell, South Carolina	3.4×10^5	3.4×10^5	1.5×10^5	----	10.0	1/77
West Valley, New York	5.5×10^5	5.5×10^4	2.8×10^5	3.6×10^3	7.0	1/77
Values used in RCF	----	----	----	(DOE Sites only)	----	----

[a]DOE Site

TABLE F3. Climatological parameters at existing LLW sites.

Site	Climate (Ref. E6)	Precipitation: Mean annual, cm (Ref. E6)	Precipitation: Net, cm (Ref. E7)	Geomorphology (Ref. E4)	Classification (Ref. E6)	Total thickness, m (Ref. E6)	Interstitial permeability to water, cm/cov (Ref. E6)	Bedrock: Classification (Ref. E6)	Bedrock: Structure
Hanford, Washington[a]	Semiarid	200	-840	Columbia plateau semi-desert	Clay, sand, and gravel	>150	Variable	Basalt	Massive/flat-lying
Richland, Washington	Semiarid	200	-840	Columbia plateau semi-desert	Clay, sand, and gravel	>150	Variable	Basalt	Massive/flat-lying
Beatty, Nevada	Arid	100	1575	Basin and range deserts	Alluvial sand and gravel	>200	0.02-0.1	Metamorphic and sedimentary	Folded
Ingel, Idaho[a]	Semiarid	200	-600	Volcanic semi-desert	Alluvial sand and gravel	6	Moderate	Basalt	Massive/flat-lying
Los Alamos, New Mexico[a]	Semiarid	400	-870	Mountainous semidesert	Weathered tuff	2	Moderate	Volcanic tuff	Massive/flat-lying
Sheffield, Illinois	Humid	900	90	Glacial	Glacial drift; sand, silt, and gravel	20-30	0.04-40	Shale, sandstone, and coal	Flat-lying
Morehead, Kentucky	Humid	1200	300	Ridge and valley Appalachian	Weathered shale, clay, and sand	3-5	0.02	Shale	Flat-lying
Oak Ridge, Tennessee[a]	Humid	1300	460	Ridge and valley Appalachian	Weathered shale and fill	10	Very low	Shale	Folded
Savannah River, South Carolina[a]	Humid	1100	0	Coastal plain	Sand and clay	10	Very low	Clay, sand, and sandstone	Flat-lying
Barnwell, South Carolina	Humid	1100	0	Coastal plain	Sand and clay	10	0.2	Clay, sand, and sandstone	Flat-lying
West Valley, New York	Humid	1000	300	Glacial	Glacial drift; clay silt, and sand	20-30	0.5	Shale	Flat-lying
Values used in RCP	Variable	500	2000	-----	Sand and clay	10	-----	-----	Flat-lying

[a]DOE Site

TABLE F4. Hydrogeologic parameters at existing LLW sites.

Site	Depth to aquifer zones, m (Ref. E6)	Type	Nearest surface water (Refs. E5, E6)	Water flow paths from burial areas (Ref. E6)	Observed radionuclide migration (Ref. E6)	Downstream river flow rate normal annual mean, m^3/sec (Ref. E3)	Seismic hazard zone (Ref. E1)
Hanford, Washington[a]	100	Gradient	10 km perennial (Columbia River)	Pores in sand and gravel	Migration through uptake by deep-rooted plants	Columbia River McNary Dam 5320	2
Richland, Washington	100	Gradient	10 km perennial (Columbia River)	Unsaturated flow in sand and gravel pores	Not observed	Columbia River McNary Dam 5320	2
Beatty, Nevada	60	Gradient	3 km ephemeral (Amargosa River)	Unsaturated flow in sand and gravel pores	Not observed	None	3
Inel, Idaho[a]	60-300	Gradient	3 km ephemeral (Big Lost River)	Pores in sand and gravel	Possibly by on-site groundwater	Snake River Hagerman, Idaho 260	3
Los Alamos, New Mexico[a]	200	Gradient	1 km ephemeral 8 km	Bedrock fractures and sand and gravel pores	On-site vadose water zone	Rio Grande River Albuquerque, New Mexico 28	2
Sheffield, Illinois	5-20 100	Vadose gradient	Site boundary perennial	Pore spaces in fill	Not observed	Mississippi River St. Louis, Missouri 4935	1
Morehead, Kentucky	1-2	Vadose	500 m perennial	Shale fracture	On and off-site ground and surface water	Licking River Covington, KY ~100 Ohio River Louisville, KY 3445	1
Oak Ridge, Tennessee[a]	5	Vadose	On-site perennial	Shale fracture and pores in fill	On-site groundwater off-site surface water	Clinch River, Oak Ridge, Tenn. Tennessee River, Chattanooga 1045 Mississippi River Memphis, Tenn. 11,365	2
Savannah River, South Carolina[a]	10 200	Vadose gradient	On-site perennial	Pores spaces in sand	On-site groundwater	Savannah River Clyo, Georgia 335	2
Barnwell, South Carolina	10 200	Vadose grandient	2 km perennial (lower three run)	Pore spaces in sand	Not observed	Savannah River Clyo, Georgia 335	2
West Valley, New York	1-20	Vadose gradient not observed	On-site perennial	Shale fracture	On-site groundwater off-site surface water	St. Lawrence River Lake Ontario outlet 7080	2
Values used in RCP	18	Gradient	1 km perennial	Pores in soil underlying aquifer	-----	5320	----

[a]DOE Site

TABLE F5. Demographic data for existing LLW sites.

Site	Nearest population centers	Downstream population
Hanford, WA[a]	5 km (Richland)	1,000,000
Richland, WA	5 km (Richland)	1,000,000
Beatty, NV	8 km (Beatty) 180 km (Las Vegas)	6,000
Inel, ID[a]	26 km (Arco) 62 km (Blackfoot)	1,150,000
Los Alamos, NM[a]	4 km (Los Alamos) 78 km (Albuquerque)	2,250,000
Sheffield, IL	2.5 km (Sheffield) 17 km (Kewanee) 47 km (Peru)	6,000,000
Morehead, KY	2 km (Morehead) 24 km (Olive Hill) 27 km (Owingsville)	5,600,000
Oak Ridge, TN[a]	2.3 km (Oak Ridge) 24 km (Knoxville)	4,000,000
Savannah River, SC[a]	18 km (Jackson) 24 km (Barnwell) 37 km (Aiken)	350,000
Barnwell, SC	11 km (Barnwell) 13 km (Williston) 33 km (Aiken)	350,000
West Valley, NY	12 km (Springville) 27 km (Salamanca) 38 km (Olean)	6,000,000
Representative values	5 km	800,000

[a]DOE site

These tables provide a perspective on the RCF that has been mathematically modeled for determining the interfaces of the classification system. Parameters selected for the RCF appear at the bottom of each table. The RCF has been used for predicting radionuclide migration from a typical site to develop acceptable concentration levels in wastes requiring containment. Inclusion of these tables and the RCF parameters in this report should not, however, be construed as either acceptance of current practices or criteria for future confinement facilities. The intent is to provide guidance.

Because the release of radionuclides from a containment facility depends, in part, on the physical form of the waste, some characteristics of currently buried wastes are described in this appendix. LLW are generated in each stage of the nuclear fuel cycle. This study, however, does not include wastes from mining, milling, refining, and enriching operations. These large-volume, low-activity wastes, which generally are not buried, may be the topic of future evaluations. Included in the LLW that we describe here are those generated in fuel fabrication, reactor and reprocessing operations, and cleanup and decommissioning processes, and by non-fuel-cycle sources such as hospitals and industrial users of radioactive materials. Also included are dewatered solids and otherwise solidified LLW.

Decontamination and decomissioning of nuclear facilities produce large items of construction and structural materials that have been contaminated and/or activated during usage. They are currently disposed of by burial, with little packaging other than plastic sheeting.

In addition to the wastes generated in the nuclear fuel cycle, a number of other LLW sources exist. Among them are medical, university, and research users of radiation, who send radioactive wastes to burial facilities for disposal. These wastes include animal remains, contaminated glassware and laboratory supplies, failed equipment, trash, and small amounts of excess radioactive isotopes.

In the past, the radioactivity in wastes at the commercial sites has been characterized largely by use of broad categories such as special nuclear materials, source materials, and byproduct materials for fissile, fertile, and fission and activation products, respectively. Tables F6 through F10 use these categories to summarize the data for existing sites.

TABLE F6. Average concentration, Ci/m^3 of LLW buried at commercial sites, 1963 through 1976.

Site	Year														
	1963	1964	1965	1966	1967	1968	1969	1970	1971	1972	1973	1974	1975	1976	Average
Barnwell, South Carolina	--	--	--	--	--	--	--	--	3.52	0.27	2.68	18.03	0.96	0.97	4.91
Beatty, Nevada	1.62	2.28	3.21	3.39	3.39	1.90	2.28	2.98	1.20	1.22	1.40	5.83	4.40	1.16	2.49
Morehead, Kentucky	10.2	38.3	11.1	9.5	5.4	5.6	3.0	3.8	54.6	13.9	11.7	16.9	16.9	15.3	16.0
Richland, Washington	--	--	0.22	0.42	6.18	15.4	128.0	125.0	41.0	48.6	55.2	8.63	75.5	36.4	34.6
Sheffield, Illinois	--	--	--	--	1.52	0.88	1.09	1.92	1.78	0.82	0.28	0.26	0.43	0.57	0.67
West Valley, New York	2.63	1.78	4.56	8.74	10.35	11.46	5.44	7.12	6.67	8.67	22.74	6.47	5.01	--	8.66
Average	4.77	12.68	7.01	6.59	5.86	5.94	5.72	6.15	27.4	8.62	8.44	10.66	7.98	5.66	8.50

TABLE F7. Byproduct material buried at LLW sites from 1962 through 1976.

Site	Year															
	1962	1963	1964	1965	1966	1967	1968	1969	1970	1971	1972	1973	1974	1975	1976	Total
Barnwell, South Carolina																
volume, m^3	---	---	---	---	---	---	---	---	---	1,171	3,757	15,839	18,244	18,072	28,829	85,912
curies	---	---	---	---	---	---	---	---	---	4,118	997	42,500	329,043	17,420	27,890	421,968
Beatty, Nevada																
volume, m^3	1,860	3,510	2,840	1,990	3,530	3,210	3,560	2,480	4,130	3,580	4,300	4,080	4,100	4,179	3,864	53,042
curies	---	5,690	6,477	6,377	11,974	10,894	6,808	9,761	12,304	4,716	5,228	5,704	23,904	18,389	4,493	123,119
Morehead, Kentucky																
volume, m^3	---	2,210	3,870	5,750	5,560	7,820	8,180	10,400	12,500	13,200	15,600	10,100	8,520	17,098	13,783	134,591
curies	---	22,716	147,322	63,828	52,729	42,280	45,578	31,027	46,968	720,146	217,350	118,274	143,656	289,581	211,356	2,153,802
Richland, Washington																
volume, m^3	---	---	---	670	2,400	870	670	440	420	580	680	1,033	1,410	1,500	2,867	13,520
curies	---	---	---	144	1,006	5,378	10,330	55,964	52,820	23,916	31,809	57,037	12,173	113,341	104,306	468,224
Sheffield, Illinois																
volume, m^3	---	---	---	---	---	2,530	2,710	2,010	2,830	4,430	5,960	8,530	12,400	14,112	13,480	68,992
curies	---	---	---	---	---	3,850	2,381	2,192	5,427	7,895	4,857	2,834	3,229	6,104	7,744	46,513
West Valley, New York																
volume, m^3	---	520	6,390	4,720	4,700	4,950	4,500	4,270	5,100	6,360	7,060	7,500	8,580	2,049	---	66,726
curies	---	1,372	11,355	21,515	41,056	51,230	51,675	23,264	36,241	42,458	61,208	170,552	55,505	10,273	---	577,754
Total																
volume, m^3	1,860	6,240	13,100	13,100	16,200	19,400	19,600	21,400	25,000	29,301	37,285	47,046	53,242	57,010	62,823	422,607
curies	---	29,778	166,154	91,874	106,765	113,632	116,772	122,209	153,810	802,849	321,449	396,901	567,510	455,098	355,789	3,800,590

TABLE F8. Special nuclear materials (fissile materials) buried at LLW sites.

Site	Year															
	1962	1963	1964	1965	1966	1967	1968	1969	1970	1971	1972	1973	1974	1975	1976	Total
Barnwell, South Carolina																
g	--	--	--	--	--	--	--	--	--	13,220	46,718	99,800	110,444	64,425	92,800	427,407
g/m^3	--	--	--	--	--	--	--	--	--	11.3	12.4	6.30	6.05	3.56	3.22	4.97
Beatty, Nevada																
g	319	41,304	172,030	324,762	5,872	22,644	8,602	5,005	7,708	757	21,177	15,164	16,954	29,276	2,096	683,670
g/m^3	0.17	11.8	60.6	168	1.66	7.05	2.40	1.17	1.87	0.21	4.92	3.72	4.14	7.01	0.54	12.89
Morehead, Kentucky																
g	--	959	11,889	4,261	7,462	14,842	17,771	31,506	47,562	72,770	71,443	46,244	23,832	25,690	27,474	403,705
g/m^3	--	0.43	3.07	0.74	1.34	1.90	2.17	3.03	3.80	5.51	4.58	4.58	2.80	1.50	1.99	3.00
Richland, Washington																
g	--	--	--	3	1,418	0.16	0.27	32	200	15	832	6,558	4,884	18,978	24,378	57,298
g/m^3	--	--	--	4.0-3	0.59	2.0-4	4.04-4	0.07	0.47	0.03	1.27	6.35	3.46	12.65	8.50	4.24
Sheffield, Illinois																
g	--	--	--	--	--	1,238	1,754	3,843	5,649	9,934	5,898	6,126	6,198	5,285	1,738	47,663
g/m^3	--	--	--	--	--	0.49	0.65	1.91	2.0	2.24	0.99	0.72	0.50	0.37	0.13	0.69
West Valley, New York																
g	--	952	3,273	2,433	4,999	3,446	2,045	.7,301	8,273	4,816	7,321	7,710	2,984	--	--	56,003
g/m^3	--	1.82	0.51	0.52	1.06	0.70	0.45	1.71	1.62	0.76	1.04	1.03	0.35	--	--	0.84
Total																
g	319	43,215	187,192	341,459	19,751	42,170	30,172	47,687	69,392	101,512	153,389	181,107	166,296	143,654	148,486	1,675,801
g/m^3	0.17	6.93	14.29	26.06	1.22	2.17	1.54	2.23	2.78	3.46	4.11	3.85	3.12	2.52	2.36	3.97

TABLE F9. Source material (nonfissile uranium and thorium) buried at commercial sites.

Site	Year															
	1962	1963	1964	1965	1966	1967	1968	1969	1970	1971	1972	1973	1974	1975	1976	Total
Barnwell, South Carolina																
kg	--	--	--	--	--	--	--	--	--	12,586	1,606	45,305	26,961	46,005	16,259	148,682
kg/m^3	--	--	--	--	--	--	--	--	--	10.7	0.43	2.86	1.48	2.55	0.56	1.71
Beatty, Nevada																
kg	296	472	331	236	91	346	1,040	290	322	--	9,340	11,500	9,710	1,438	5,000	40,412
kg/m^3	0.16	0.13	0.12	0.12	0.03	0.11	0.29	0.07	0.08	--	2.17	2.82	2.37	0.34	1.29	0.76
Morehead, Kentucky																
kg	--	5,210	5,590	568	690	5,680	6,250	2,550	7,220	5,730	8,260	9,340	13,100	82,416	75,944	226,548
kg/m^3	--	2.36	1.44	0.10	0.12	0.73	0.76	0.25	0.58	0.43	0.53	0.43	1.54	4.82	5.51	1.70
Richland, Washington																
kg	--	--	--	0.9	253.0	0.9	2.7	88.4	31.3	606	3,113	2,250	20.3	215	5,011	11,592
kg/m^3	--	--	--	1.0-3	0.11	1.0-3	4.0-3	0.20	0.07	1.04	4.76	2.18	0.014	0.14	1.75	0.86
Sheffield, Illinois																
kg	--	--	--	--	--	3,930	8,703	6,330	2,000	212	3,600	2,410	13,900	35,950	3,854	80,889
kg/m^3	--	--	--	--	--	1.55	3.21	3.33	0.71	0.05	0.68	0.28	1.12	2.25	0.29	1.17
West Valley, New York																
kg	--	7,580	10,100	22,200	38,300	20,300	6,460	80,000	31,700	51,400	72,500	44,200	61,700	--	--	446,440
kg/m^3	--	14.52	1.58	4.70	4.15	4.10	1.43	18.69	6.22	8.07	10.27	5.89	7.19	--	--	6.69
Total																
kg	296	13,300	16,000	23,020	39,400	30,224	22,500	89,300	41,300	70,546	98,373	115,195	125,161	166,024	106,068	956,707
kg/m^3	0.16	2.13	1.22	1.75	2.43	1.65	1.15	4.18	1.65	2.41	2.64	2.45	2.35	2.36	1.69	2.26

TABLE F10. Inventory of nuclides as buried at INEL.

Nuclide	Half-life, d	Percentage of curie activity 1974	1975
^{144}Ce, ^{144}Pr	285	--	4.3%
^{144}Ce	285	4.1%	3.0%
^{60}Co	1,920	38.8%	55.3%
^{51}Cr	28	0.4%	1.7%
^{134}Cs	770	0.3%	0.3%
^{137}Cs	10,877	7.2%	4.3%
^{154}Eu	5,840	0.2%	0.3%
^{155}Eu	661	0.1%	0.2%
^{55}Fe	880	--	2.7%
^{59}Fe	45	7.3%	1.5%
MAP	--	0.5%	0.1%
MFP	--	3.8%	2.2%
^{54}Mn	312	0.4%	2.8%
^{56}Mn	0.1	--	4.2%
^{59}Ni	29,000,000	16.2%	--
^{238}Pu	31,536	1.6%	--
^{86}Rb	19	0.3%	--
^{106}Ru, ^{106}Rh	369	0.3%	0.6%
^{106}Ru	369	1.7%	1.3%
^{125}Sb	486	0.5%	0.6%
^{90}Sr, ^{90}Y	10,439	5.5%	11.7%
^{90}Sr	10,439	8.0%	1.6%
Unident. $\beta + \gamma$	--	0.1%	0.1%
^{65}Zn	245	1.8%	--
^{95}Ar, ^{95}Nb	65	0.8%	0.9%
Total curies buried, Ci		19,770	13,190
Total volume buried, m^3		3,694	5,685

Table F10 lists nuclide-specific data from the Idaho National Engineering Laboratory (INEL) for 1974 and 1975. The wastes were generated during all phases of the fuel cycle after irradiation in test reactors. It is expected that the activity and concentration ratios for these wastes are generally typical of LLW. No other nuclide-specific data characterizing LLW is readily available. This table has been included only for perspective concerning the nuclide content of typical wastes.

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APPENDIX G

REFERENCE CONTAINMENT FACILITY DETAILS

The RCF is based on shallow land burial. Nonetheless, selection of appropriate parameters makes the methodology suitable for calculations based on other disposal methods. A list of design assumptions and criteria was put together for the RCF used in the analyses. The following items have been assumed as typical of future waste-disposal facilities and appropriate for the RCF:

- The facility will use shallow land burial in open pits, covering the waste as filled with overburden removed during excavation.
- The wastes will arrive at the facility in DOT-approved shipping containers suitable for burial and sized for conventional materials-handling equipment.
- Wastes will be solidified at their source before shipment. No provisions will be made for handling liquid radioactive wastes at the RCF.
- A 121 hectare (300-acre) site will be considered. It is assumed to be in a semiarid locale, 5 km from a major highway, 2 km from a railroad main line, and 1 km from a large river. It is also assumed that between one-half and two-thirds of the total land area will be used for burial. The remaining area will allow for open terrain, structures, service roads, and highway and railroad access areas.
- The site will be fenced with an intrusion alarm to prevent unauthorized entry.
- Permanent identification markers will be used to locate burial trench boundaries.
- Wastes will be covered with soil every day.
- Migration of radioactive material will be monitored before, during, and after the operational phases of the facility.

- After the site is filled, unnecessary structures will be removed. The area will then be graded and seeded for erosion control, and a perpetual maintenance program will be established.
- A 40-y operational lifetime for burial operations is assumed. Institutional control and surveillance of the site will be maintained for as long as a few hundred years.
- About two-thirds of the trench volume is assumed to consist of soil mixed between waste containers as the trenches are filled.
- When calculating the concentrations in human exposure pathways, no credit will be taken for the retention of radionuclides in the buried containers.

Total site capacity is 569,000 m^3 of waste when 100 typical trenches are filled to capacity with wastes. This total is about 14,000 m^3 per year for a 40-y facility lifetime. At a uniform fill rate, about two hundred and eighty 55-gal drums per day are needed to fill the site in 40 y. This capacity is enough to handle the wastes projected to be generated by about 1000 GW(e)-y of nuclear power production.

Table G1 lists the factors used to arrive at the unit costs of \$100/m^3 of waste disposed. The costs are consistent with charges now assessed by commercial site operators across the country.

These data have been used as input for the cost-benefit analysis to evaluate cost-effectiveness as a function of interface level.

TABLE G1. Cost factors for RCF.

Cost factors		Costs, $
Capital		
Site acquisition and development		3,500,000
Structural construction		1,200,000
Site preparation		230,000
Operating equipment		1,140,000
Operationg costs		
Labor	367,500/y x 40 y	14,700,000
Supplies	7,000/y x 40 y	280,000
Materials	20,000/y x 40 y	800,000
Environmental impact statement		500,000
Architect-engineering		210,000
Site maintenance and perpetual care		770,000
Net costs		23,330,000
Financing charges		28,200,000
Total costs		51,530,000
Site capacity 569,000 m^3		
Unit costs \$90/m^3	Consider contingencies to make \$100/m^3 cost-effective.	

APPENDIX H

CALCULATIONS FOR AIRBORNE RELEASES

When determining the allowable concentration limits in the waste to be placed in the low-level burial RCF, we considered various potential airborne releases to identify limiting cases. Two potential occurrences that lead to limiting values of concentration are (1) an accident resulting in the release of a fraction of the contents of a barrel during the burial operation, and (2) the reclamation of land used as a burial site following loss of institutional control of the site after a few hundred years. The first case sets a limit on concentration considering both short and long half-life elements. The second case sets a limit based mainly on long half-life elements. In addition to these two cases, a continuous release was also investigated.

SINGLE-BARREL ACCIDENT SCENARIO

It is assumed that LLW will be packaged for disposal in 55-gal drums or barrels, as is the present practice. The single-barrel accident is defined as the instantaneous release of a fraction of the contents of a barrel during the burial operation. The main pathway of concern for dose to the off-site population is the airborne transport of the material released from the barrel. For this analysis, the following conservative yet realistic assumptions have been made:

- For the release, 0.1% of the barrel contents becomes airborne.
- A standard man is on the plume centerline at the site boundary 160 m (0.1 mi) from the source. He is engaged in light activity (20 l/min respiration rate).
- A Pasquill F stability level is used.
- The dose to any organ should not exceed 0.5 rem/y.

The concentration at the plume centerline for an instantaneous point source released at the surface is given by:

$$\chi(x,y,z,t) = \frac{Q(2)^{-1/2}(\pi)^{-3/2}}{\sigma_x\sigma_y\sigma_z} \exp\left[-\frac{(x-\bar{u}t)^2}{2\sigma_x^2}\right] \quad \text{(H1)}$$

where

$\chi(x,y,z,t)$ = concentration, Ci/m^3

Q = source strength, Ci

$\sigma_x, \sigma_y, \sigma_z$ = dispersion coeffecients, m

$\bar{u}$ = average wind speed, m/sec

x = direction of plume axis (wind).

This equation is integrated to get the center-line concentration-time exposure

$$I = \int_0^\infty \chi \, dt = \frac{Q}{\pi\sigma_y\sigma_z\bar{u}} \quad \frac{\text{Ci-sec}}{\text{m}^3} \quad \text{(H2)}$$

For an F stability level, $\sigma_y \cong 7$, $\sigma_z \cong 3{,}5$, and using

$\bar{u}$ = 1.56 m/sec (3.5 mph),

$$I = 8.33 \times 10^{-3} \, Q \quad \frac{\text{Ci-sec}}{\text{m}^3} \quad \text{(H3)}$$

Some of the activity released in an accident will fall out in the immediate vicinity and not reach the perimeter fence. To be conservative, however, we assume that all the activity that becomes airborne reaches the location of our maximum individual located directly downwind 160 m from the site of the accident. The amount of radioactivity he inhales is the product of the intergrated concentration at that location and his respiration rate. Thus, the amount inhaled is

$$I_o = \left(8.33 \text{ E-03 } Q \quad \frac{\text{Ci-sec}}{\text{m}^3}\right)\left(20 \frac{1}{\text{min}}\right)\left(10^{-3} \frac{\text{m}^3}{\text{l}}\right)\left(\frac{1 \text{ min}}{60 \text{ sec}}\right)$$

$$= 2.8 \text{ E-06 } Q \text{ Ci.} \quad \text{(H4)}$$

The acceptable dose to an organ at risk from internally deposited radioactivity was 0.5 rem/y. Appendix B, Table II, of 10 CFR 20 shows that ^{239}Pu in soluble form is the most radiotoxic isotope listed,* with a maximum permissible concentration of 6×10^{-14} $\mu Ci/cm^3$.

The exposure to the organs of the body was calculated with the LLL AERIN computer code. This code tracks the radionuclide from the moment of initial deposit in the body through the translocation pathways to various organs of the body, calculating the dose to the organs as a function of time. The code accepts intake into the body of any nuclide as a single acute exposure, as a series of acute exposures received at any selected time intervals, or as a continuous exposure of varying concentrations.

The code was developed mainly for inhalation exposures, but it can be used to describe other modes of intake by altering certain constants (i.e., mass of organs, fractions of the radionuclide moving to and from the blood, biological half-life of the radionuclide in the organs, and so forth).

In the case of plutonium inhalation, the code uses the ICRP Lung Model. The particle size of the inhaled nuclide is an important parameter in this model.

Table H1, resulting from the AERIN Code runs for soluble and insoluble ^{239}Pu of different particle sizes, shows the value of I_o necessary to give a maximum organ dose of 0.5 rem/y.

*^{248}Cm is the single exception. The MPC for it is 2×10^{-14} $\mu Ci/cm^3$; however, the quantity of this isotope in waste will be very much less than one third the quantity of ^{239}Pu. Therefore, we shall use ^{239}Pu in the calculation of internal dose to our maximum exposed individual.

TABLE H1. Values of I_o in μCi to give 0.5 rem/y maximum dose to body organ indicated for inhalation of soluble and insoluble ^{239}Pu.

	Particle size (AMAD); μm			
Organ	0.1	1.0	5	10
Bone	8.2 E-03 (S)[a]	9.8 E-03 (S)	8.2 E-03 (S)	8.2 E-03 (S)
	1.4 E-02 (I)	2.6 E-02 (I)	4.1 E-02 (I)	5.3 E-02 (I)
Lung	5.1 E-03 (S)	1.1 E-02 (S)	2.1 E-02 (S)	3.2 E-02 (S)
	1.3 E-03 (I)	2.7 E-03 (I)	5.4 E-03 (I)	8.1 E-03 (I)

[a]S = Soluble "W" class; I = Insoluble "Y" class

Particle sizes in Table H1 are normalized to an activity median aerodynamic diameter (AMAD). From Stokes' law on the terminal velocity of particles settling in a medium, the approximate AMAD of a particle with density ρ can be given by

$$AMAD = D\left(\frac{\rho - \rho_o}{1 - \rho_o}\right)^{1/2}$$

where D is the physical diameter of the particle in question and ρ_o is the density of the medium.

For example, a 1-μm-diameter particle of PuO_2, which has a density of 11.46 g/cm^3, would have an AMAD of 3.4 μm in air. A reasonably conservative assumption for particle size in the accident scenario would be 1 μm AMAD.

From Table H1, the most restrictive value of I_o for a 1 μm AMAD is 2.7 E-03 μCi. Using equation (H4) we solve for Q and get

$$Q = \frac{2.7\ E\text{-}03}{2.8\ E\text{-}06}\ \mu Ci = 1\ E{+}03\ \mu Ci.$$

Thus, the maximum release of ^{239}Pu should not exceed 1 E+03 μCi if exposure of the maximum exposed individual at the perimeter fence is not to exceed 0.5 rem/y.

If we assume that the waste container is a 55-gal drum (volume - 0.208 m^3) and that 0.1% of the barrel contents becomes airborne as a result of the accident, then the maximum concentration (MC) of ^{239}Pu permitted in the barrel is:

$$MC_{(waste)} = \frac{1.0\ E+03\mu\ Ci}{0.1\ E-02 \times 0{,}208\ E+06\ cm^3}$$

$$MC_{(waste)} = 4.8\ \mu Ci/cm^3.$$

The objectives of this study do not include determination of the correct or appropriate guidelines for acceptable risk. We have used 0.5 rem/y on the basis of present regulatory guides. The value is used to illustrate the applicability of the methodology for qualifying the WC system interfaces. Other guidelines could be used and the results will scale proportionally to the limiting dose accepted.

CONTINUOUS AIRBORNE SCENARIO

The effect of a continous airborne source model was considered in determining limiting cases. We developed a scenario in which natural erosion has exposed the buried waste. The radioactive nuclides are resuspended, and they contaminate the air downwind of the RCF. Populations of the public are continuously exposed by inhaling the contaminated air. In developing the calculations for this scenario, we shall assume that our RCF of 120 hectares is a plot 1200 m by 1000 m and that one-half of this land area was used to bury the radioactive waste. The concentration of radioactivity in the soil of the buried waste is $(Q_B/30)$ $\mu Ci/cm^3$, where Q_B is the maximum concentration of the activity permitted in the waste container at the time of burial. The factor 30 is the product of the soil dilution (3) and the peak-to-average ratio (10).

The average long-term soil erosion rate of 25 g/m^2-y will be assumed. We shall assume also that this is the rate for wind erosion alone.

After enough time, the natural eroding forces of wind and water will expose the contents of the burial site. It is most unlikely that exposure would occur during the 100 y of institutional control. In fact, it is reasonable to believe that many years will pass after controls are removed before erosion will work down the overburden to expose the radioactivity. Only radionuclides with long half-lives will still be present. We shall assume that the exposed radioactivity is ^{239}Pu and that no radioactive decay has occurred since burial. The source term then is given by the following equation:

$$Q = \left(\frac{\frac{\mu Ci}{Q_B \ cm^3}}{30}\right)\left(\frac{25g}{m^2\text{-}y}\right)\left(\frac{A \ m^2}{\rho \ g/cm^3}\right) 10^{-6} \ \frac{Ci}{\mu Ci}$$

where

A = surface area of exposed waste material = $1.2 \times 10^6 \times 0.5$, or 6×10^5 m^2

ρ = density of soil = 2 g/cm^3.

The source term then is

$$Q = 0.25 \ Q_B \text{ Ci/y or } 7.9 \times 10^{-9} \ Q_B \text{ Ci/sec.}$$

Pasquill's continuous point-source diffusion equation for a surface release is:

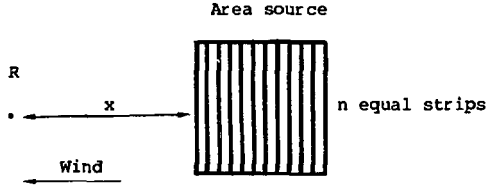
$$\chi = \frac{Q}{\pi \sigma_z \sigma_y \bar{u}} \ e^{\left[\frac{-y^2}{2\sigma_y^2}\right]} \text{ Ci/m}^3 \ ,$$

where σ_z and σ_y are the standard deviations of the distribution of material in a plume in the vertical and crosswind directions, respectively. Their values are functions of the downwind distance, x, from the point of release and the stability of the atmosphere into which the material is released.

To obtain a value of χ downwind from an area source we take advantage of the property that Gaussian diffusion models possess--if the source and the receptor

locations are interchanged, the numerical value of the concentration, χ, is not affected. This means that the concentration at a point downwind from a number of sources can be computed by assuming that all the sources are combined at the receptor point and by summing the computed values of χ at the actual source points.

To apply this in the solution of an area source, we assume that the area is made up of n crosswind strip sources of length 2y. The receptor point, R, is x meters downwind from the near edge of the source area as sketched.



Thus, the concentration, χ, at R is the sum of the contribution of each strip source and is given by the equation,

$$\chi = \frac{Q'}{\pi \bar{u}} \sum_{i=1}^{n} \frac{1}{\sigma_{z_i}} \times \frac{1}{\sigma_{y_i}} \int_{-y}^{y} e^{-\left(\frac{y^2}{2\sigma_{y_i}^2}\right)} dy, \text{ Ci/m}^3 \quad . \tag{H5}$$

Q' has been redefined as the actvity per unit length emitted per unit time along each strip, or,

$$Q' = Q/2yn, \text{ Ci/m-sec}$$

where n is the number of strips chosen to represent the area of the RCF. Thus,

$$Q' = \frac{7.9\text{x}10^{-9} Q_B \text{ Ci/sec}}{1.2\text{x}10^{3} \quad n} = 6.6\text{x}10^{-12} \quad Q_B/n, \text{ Ci/m-sec.}$$

The solution of

$$\int_{-y}^{y} e^{-\left(\frac{y^2}{2\sigma_y^2}\right)} dy$$

is obtained from the table of normal probability function for each value of σ_y.

We assumed that (1) the exposed public was located 1 km downwind from the nearest edge of the RCF, (2) the average wind velocity over 50 y was 3.5 mph, and (3) the average Pasquill stability category was "D" or neutral. These are conservative values. Higher wind velocity or more instability in the lower atmosphere will lower the 50-y dose commitment for a given source term.

Setting n equal to 50, we obtain

$$Q' = \frac{6.6 \times 10^{-12} Q_B}{50} = 1.3 \times 10^{-13} Q_B, \text{ Ci/m-sec} \quad .$$

The solution of

$$\sum \frac{2}{\sigma_z \sigma_y} \int_{o}^{y} e^{-\left(\frac{y^2}{2\sigma_y^2}\right)} dy$$

yielded a value

$$3.35 \times 10^{-2}, \text{ m}^{-1} \quad .$$

Therefore, from equation H5,

$$\chi = \frac{1.3 \times 10^{-13} \ Q_B \ \text{Ci/m-sec}}{\pi (1.56) \ \text{m/sec}} \ (3.35 \times 10^{-2}) \text{m}^{-1}$$

$$\chi = 8.9 \times 10^{-16} \ Q_B, \ \text{Ci/m}^3 \quad .$$

Since a person will inhale 20 m^3/d, the intake rate, I_o, is

$$I_o = 1.8 \times 10^{-14} \ Q_B \ \text{Ci/d or } 1.8 \times 10^{-2} \ Q_B \ \text{pCi/d} \quad .$$

From the AERIN Code, the intake rate, I_o, of 10-μm AMAD particles of ^{239}Pu aerosol for 50 y to give a maximum dose-rate of 5 mrem/y to the bone is 3.3×10^{-2} pCi/d. (The bone receives the highest dose rate and total dose from long-term chronic exposure.)

Equating the two expressions for I_o, and solving for Q_B, we obtain:

$$Q_B = \frac{3.3 \times 10^{-2}}{1.8 \times 10^{-2}} = 1.8 \text{ μCi/cm}^3 \quad .$$

Table H2 lists values of Q_B, the concentration of ^{239}Pu in the waste at time of burial, to give 5 mrem/y at various distances downwind from the edge of the RCF. A particle size of 50 μm AMAD is also shown.

TABLE H2. Values of Q_B, in μCi/cm^3, that will give a dose rate of 5 mrem/y to the bone after 50 y of inhalation of contaminated dust.

Distance downwind	10 μm AMAD	50 μm AMAD
1 km	1.8	2.8
10 km	5.2×10^1	7.7×10^1
100 km	7×10^3	1.0×10^4

RECLAMATION SCENARIO

Another potential occurrence we considered is the reclamation of the burial facility site after institutional control is removed. For this case, workers are considered to be exposed to radioactively contaminated dust while moving earth at the site. The following assumptions are used in the calculations:

- The working atmosphere to which workers are exposed contains 2 mg/m^3 of dust.
- The workers spend an average of 1 mo working at the site, and they do not reside there. Consequently, a 176-hr exposure time is used.
- The respiratory rate is 20 l/min.
- The maximum allowable organ dose is 0.5 rem/y.
- The LLW is assumed to be diluted by a factor of 3 by earth during burial.
- The average interface level concentration is 1/10 the maximum concentration of radioactivity in the wastes at time of burial.

At a respiration rate of 20 l/min, a worker will inhale approximately 10 m^3 in an 8-hr work day. At a dust loading of 2 mg/m^3, 20 mg of contaminated dust will be inhaled each day.

The dust is contaminated with radioactivity at a concentration 1/30 that of the peak concentration of activity in the waste container when it was buried 100 to 300 y earlier. The amount of radioactivity inhaled by the worker per day, I_o, in relationship to the concentration in the container at burial given by

$$I_o = \frac{20\ E\text{-}03\ g/d}{\rho_s} \times \frac{Q_B}{30}\ ,\ \mu Ci/d\ ,$$

where

ρ_s = density of the soil in g/cm^3

Q_B = concentration in the waste container at burial in $\mu Ci/cm^3$.

Since

$\rho_s = 2\ g/cm^3$

$$I_o = 3.33\ E\text{-}04\ Q_B\ \mu Ci/d\ . \qquad (H6)$$

After contact with the soil for 100 or more years, the radioactive nuclides, particularly with heavy alpha emitters, will be firmly attached to the dust particles. Therefore, a reasonably conservative particle AMAD for this contaminated dust would be 10 μm. We can also assume that the radioactive material would be relatively insoluble in the body.

The AERIN Code was run for inhalation of ^{239}Pu at a daily intake for 30 d. Particle sizes of 1, 5, 10, and 50μm AMAD were analyzed.

Table H3 lists the results of these calculations. It shows the value of I_o in μCi inhaled each day for 30 d that would result in a maximum dose of 0.5 rem/y for the lung and bone.

TABLE H3. Calculated inhalation rate, I_o in μCi/d of ^{239}Pu inhaled for 30 d to give 0.5 rem/y maximum dose to worker at site ("Y" class).

Organ	Particle size (AMAD), μm			
	1	5	10	50
Lung	1.3 E-04	2.6 E-04	3.8 E-04	1.0 E-03
Bone	1.2 E-03	1.9 E-03	2.5 E-03	3.7 E-03

In this scenario the lung is the critical organ and, for 0.5 rem/y exposure, the maximum permissible inhalation rate is 3.8 E-04 μCi/d for 10μm AMAD particles.

From equation (H6) and the calculated value of I_o we get

$$3.33 \text{ E-04 } Q_B = 3.8 \text{ E-04}$$

or

$$Q_B = 1.1 \ \mu Ci/cm^3 \ .^*$$

This quantity is the maximum permissible concentration of ^{239}Pu in the waste. Values for other isotopes can be found by taking the ratio of their respective MPC in air with ^{239}Pu and multiplying by the value of A_B above.

Comparing the three scenarios on inhalation, we see that the most restrictive one is that of reclaiming the burial site 100 to 300 y after institutional control is removed.

*Note that 1.1 μCi/cm^3 in a waste container will end up averaging approximately 18 nCi/g of soil (ρ = 2 g/cm^3). This is comparable to the 10 nCi/g currently used to define TRU waste.

APPENDIX I

CALCULATIONS FOR WATERBORNE RELEASES

INTRODUCTION

Migration of radionuclides from their burial site through the geosphere represents an important exposure pathway to the environment. This migration should be considered in any environmental assessment of nuclear waste management alternatives. Several processes influence migration rates and change the rates at which nuclides are released to water systems in the environment. Among them are erosion and suspension of waste particles in surface water flows, leaching of the buried inventory (the process wherein the nuclide is taken up from its original form and becomes suspended or dissolved in water), and convection (nuclide movement caused by gross water movement). Others are dispersion (nuclide movement caused by the nuclide concentration gradient), ion exchange between the nuclides in the water and the matrix through which the water flows, and radioactive decay.

The present analysis does not consider the effect of ingrowth of a particular nuclide from decay of a parent. The effects of radioactive decay have been included to show the decrease in amount of radioactivity with time.

This appendix describes the processes and parameters that we evaluated in this investigation. Also described are a parametric sensitivity analysis and the results of base cases for several nuclides.

LEACHING MODEL

Leaching is the process by which buried materials are dissolved and enter the groundwater. The rate, in Ci/y, of addition of a nuclide to water is given by:

$$\frac{dM_w}{dt} = M_{I_o} \lambda_\ell \exp\left[-(\lambda_\ell + \lambda_d)t\right],$$

where M_{I_o} is the initial inventory, λ_ℓ is the leach constant, and λ_d is the decay constant.

The inventory, in Ci, remaining in the burial location at any time is described by:

$$M_I(t) = M_{I_o} \exp\left[-(\lambda_\ell + \lambda_d)t\right].$$

Values of leach rates for the several nuclides considered in this study are obtained using data from drill cores taken in soil below actual LLW disposal sites at INEL.[I1] We assume that any concentrations of a given nuclide below the pit or pond result from leaching. Thus, by integrating the concentration over the contaminated depth, correcting for radioactive decay, and knowing the burial and sample times, we get an order-of-magnitude estimate for the leach rates.

NUCLIDE MIGRATION

Nuclide migration is determined from a second-order differential mass-balance equation that considers longitudinal dispersion, convection, sorption, and radioactive decay. Based on a one-dimensional homogeneous medium, this equation describes the concentration as a function of space and time.[I2]

$$D \frac{\partial^2 C}{\partial x^2} - V \frac{\partial C}{\partial x} - K \frac{\partial C}{\partial t} - \lambda_d C = 0 \quad \text{Ci/m}^3\,\text{y},$$

where C is the nuclide concentration, D is the dispersion coefficient, V is the groundwater velocity, and K is the retardation factor. The first term in the equation represents longitudinal dispersion. For conservatism and simplicity, lateral dispersion has been ignored. The second term represents convection, the third term sorption processes, and the fourth term radioactive decay.

Using Laplace transform techniques on the above equation with boundary conditions characterized by a series of unit step functions, the solution is:

$$C(x,t) = 1/2 \exp\left(\frac{Vx}{2D}\right)\left[\sum_{j=1}^{N} C_j \, f(t - \tau_j) \, \mu \, (t - \tau_j)\right] \quad Ci/m^3 ,$$

where N is the number of steps in the boundary condition, τ_j is the time at which the jth step "turns on," μ is the unit step function, and

$$f(t - \tau_j) = \left\{\exp \; (-ab) \; \operatorname{erfc}\left[\frac{a - 2b(t - \tau_j)}{2(t - \tau_j)^{1/2}}\right] + \exp \; (ab) \; \operatorname{erfc}\left[\frac{a + 2b \; (t - \tau_j)}{2(t - \tau_j)^{1/2}}\right]\right\}$$

with

$$a = \left(\frac{K}{D}\right)^{1/2} x$$

and

$$b = \left(\frac{V^2}{4DK} + \frac{\lambda d}{K}\right)^{1/2}$$

and

$$\operatorname{erfc}(\theta) = 1 - \frac{2}{\sqrt{\pi}} \int_0^{\theta} e^{-z^2} \, dz$$

The sorption processes are characterized by the equilibrium sorption coefficient, or time transformation factor, K,[13] which is also expressed as a relative nuclide velocity, i.e.,

$$K = \frac{V_{water}}{V_{nuclide}} \quad .$$

Alternately, K is related to the distribution coefficient (the ratio of the concentration of the nuclide in the soil to the concentration of the nuclide in the water) by the equation

$$K = 1 + \frac{\rho}{\varepsilon} K_d \quad ,$$

where K_d is the distribution coefficient, ρ is the soil density, and ε is soil porosity.

The time of arrival of the contamination front is given approximately by

$$t_a = \frac{KL}{V_{water}} \quad y \quad ,$$

where L is the length of the aquifer.

SOIL AND SITE CHARACTERISTICS

In this study, the soil has been characterized by the dispersion coefficient, D, and the equilibrium sorption coefficient, K. The value of D is 0.42 m^2/y [12], and the values of K for the various nuclides are shown in Table 11.

TABLE 11. Nuclide-specific parameters and values.

Nuclide	Initial inventory, Ci	Sorption coefficient	Leach rate, y^{-1}
^{90}Sr	24000	100	1×10^{-2}
^{129}I	0.73	1	1×10^{-2}
^{137}Cs	35000	1000	1×10^{-2}
^{237}Np	12000	100	6×10^{-4}
^{239}Pu	65000	10000	6×10^{-4}
^{241}Am	65000	1000	6×10^{-4}

Source: Ref. 15.

Other parameters characterizing soils have been implicitly assumed in the selection of groundwater velocities.[I2] Included are permeability, porosity, and water pressure gradient.

As has been noted elsewhere,[I4] the information is available about the sorption coefficient for soils. The values of Table I1 are for a typical western desert soil.[I5]

The parameters representative of the RCF are site area, number of pits, distance from pit bottoms to aquifer, aquifer flow area, aquifer velocity, distance from site to nearest surface waters, and rain rate. Table I2 shows their values.

TABLE I2. Site-specific parameters and values.

Parameter, units	Value
Site plan area, m^2	2×10^6
Number of pits	100
Distance between pit and aquifer, m	10
Water velocity from pit to aquifer, m/y	10
Rain rate, m/y	0.5×10^{-2}
Aquifer flow area, m^2	1000
Distance from site to surface water, m	1000
Aquifer water velocity, m/y	110
Dispersion coefficient, m^2/y	0.42
River volume flow rate at nuclide inlet, m^3/sec	2550
River volume flow rate at nuclide outlet, m^3/sec	3790

BASE CASES

Table I3 contains the results of the base case analysis. Strontium-90, ^{137}Cs, and the ^{241}Am initially present do not appear at the aquifer outlet, because the initial inventories essentially have decayed completely before they reach the aquifer outlet. The effects of ingrowth of radioactive daughter products may be important,[I6] but they have not been considered. Matching calculations performed by The Analytic Sciences Corp. are also given in Table I3.

TABLE I3. Preliminary dose calculations.

Isotope	Burial facility inventory, kg	Burial facility inventory, Ci	Average burial facility activity, Ci/m^3	Arrival Time of contamination front, y (half-lives)	Peak Fraction of original inventory released to river, y^{-1}	Activity into river per year after arrival of contamination front, Ci/y	Average individual 50-y body dose, rem	Total population dose, man-rem 50 y
BASE CASE model								
^{239}Pu	1060	6.5×10^{4}	0.11	10^{5} (4.1)	2.0×10^{-7}	1.3×10^{-2}	1.2×10^{-7}	8.9×10^{-2}
^{241}Am	1995	6.5×10^{6}	11.0	10^{5} (22)	--	--	--	--
^{237}Np	1.72×10^{4}	1.2×10^{4}	2.1×10^{-2}	10^{3} (4.7×10^{-4})	6.0×10^{-6}	7.2×10^{-2}	1.1×10^{-6}	8.8×10^{-1}
^{129}I	4.2	0.73	1.3×10^{-6}	10^{1} (6.3×10^{-7})	9.0×10^{-3}	6.6×10^{-3}	9.9×10^{-8}	7.9×10^{-2}
^{137}Cs	40	3.5×10^{6}	6.2	10^{4} (34)	--	--	--	--
^{90}Sr	17	2.4×10^{6}	4.2	10^{3} (35)	--	--	--	--
TASC model								
^{239}Pu	1060	6.5×10^{4}	0.11	1.3×10^{5} (5.3)	1.8×10^{-8}	1.1×10^{-3}	1.0×10^{-8}	8.0×10^{-3}
^{237}Np	1.72×10^{4}	1.2×10^{4}	2.1×10^{-2}	10^{3} (4.7×10^{-4})	6×10^{-6}	7.2×10^{-2}	1.1×10^{-6}	8.8×10^{-1}
^{129}I	4.2	0.73	1.3×10^{-6}	17 (1.1×10^{-6})	10^{-2}	7.3×10^{-3}	1.1×10^{-7}	8.8×10^{-2}

EROSION

It is conservative to assume that future confinement facilities will not be sited in areas where substantial erosion is likely to occur. Nontheless, some contamination would ultimately be released to surface waters or dispersed into the atmosphere if the RCF were located in an area where wind or water erosion were occurring. Design features, such as covering the filled burial area with pebbles through which grass vegetation could be established, would tend to minimize erosion processes.

To determine whether erosion may pose a substantial health risk, we performed a simplistic yet conservative calculation based on a representative erosion rate. A number of site-specific parameters influence erosion rates. Among them are surface slope, amount of precipitation, distances to watercourses, distances from peaks, amount and type of vegetation, and soil properties. A typical sheet erosion rate is 6 tons of soil per acre per year.[17] Using this rate, the RCF pit surface area of 50 acres, and a soil density of 2 g/cm^3, one determines that 1500 y are needed for a surface covered 1 m thick to be eroded away before erosion of the buried wastes begins. The concentration in the eroded material that will give guideline doses to maximum individuals and populations can be determined if three assumptions are made: (1) a dilution factor of 3 accounts for the mixing of the wastes with soil during burial, (2) a factor of 10 accounts for the difference between the interface and average concentrations of activity in the waste, and (3) dilution and holdup of eroded material between the pits and the river are ignored.

If all of the eroded material goes directly into the river (with a volumetric flow rate of 2550 m^3/sec), and if sedimentation is neglected, it would take an initial waste concentration of 10^5 $\mu Ci/cm^3$ of ^{239}Pu to give a maximum individual dose of 0.5 rem/y. This calculation presumes that the erosion takes place uniformly over the total surface and that gullying and preferential erosion do not occur. For the erosion case, where potentially large populations may be involved, guideline doses of 5 mrem/y to maximum individuals may be more appropriate. This would allow concentrations of ^{239}Pu in the wastes of 1000 $\mu Ci/cm^3$.

It is evident that, even with overly conservative assumptions, erosion does not give the most restrictive limitation of concentration. Factors such as establishment of a dense vegetative cover over the buried wastes and siting the containment facility where sheet erosion does not occur would reduce even further any possible consequences of erosion.

PARAMETRIC VARIATIONS

Figure I1 shows the effect of variation of the leach constant, λ_ℓ, hence the leach rate, $\frac{dM_w}{dt}$, on the concentration at the pit bottom. As the value of the leach rate decreases significantly below the decay rate, the decay rate becomes more dominant in determining the rate at which the concentration changes. The converse is also true. Figure I1 demonstrates also that the concentration increases as the leach rate increases.

The influence of the dispersion coefficient, D, on the response at the aquifer outlet is shown in Fig. I2. As the dispersion of the pulse increases (i.e., as the dispersion coefficient increases), the pulse becomes broader and is reduced in magnitude.

Figure I3 shows that, as the hypothetical pulse boundary condition is decreased in duration (but not in magnitude) with a constant dispersion coefficient, the magnitude of the concentration at the aquifer outlet is accordingly decreased. That is, as the pulse is shortened, the output has less time to equilibrate, and it achieves a smaller fraction of equilibrium concentration.

Figure I4 shows the effect on individual doses of varying the volumetric flow rate of the stream into which the wastes flow. This study used a base case of a rather large flow typical of the Columbia River. As long as the relative uses of water along the stream remain constant, the population doses do not depend on the specific volumetric flow rate.

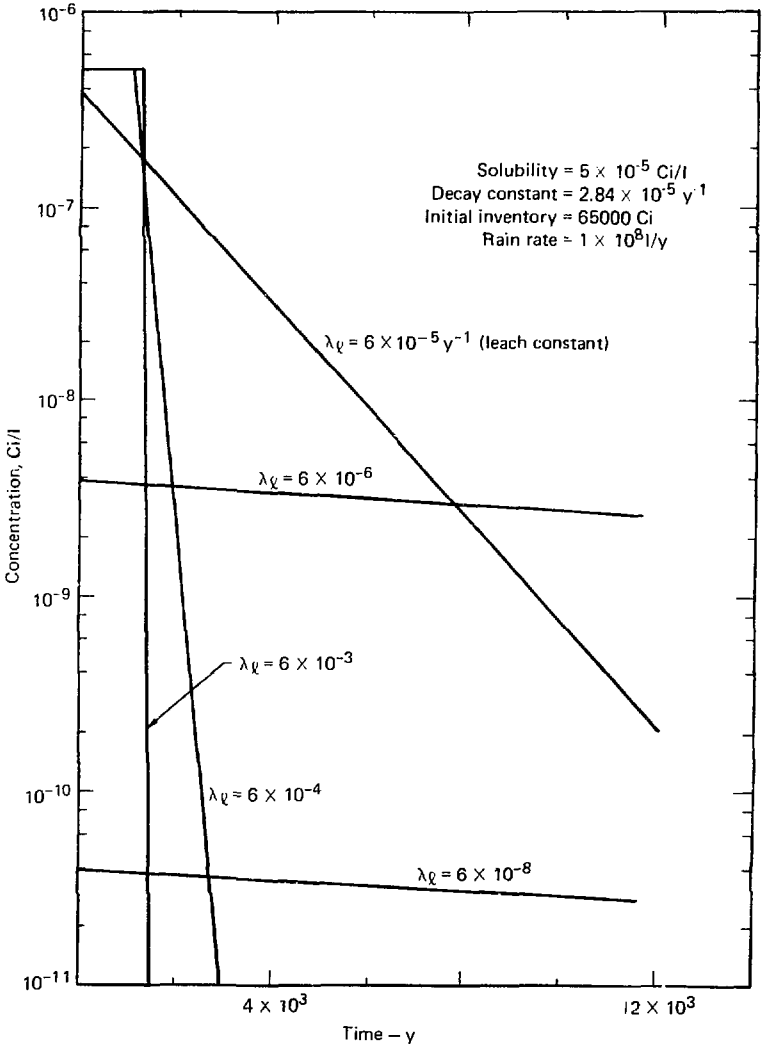


FIG. 11. Concentration at pit bottom for ^{239}Pu vs time, with leach constant as a parameter.

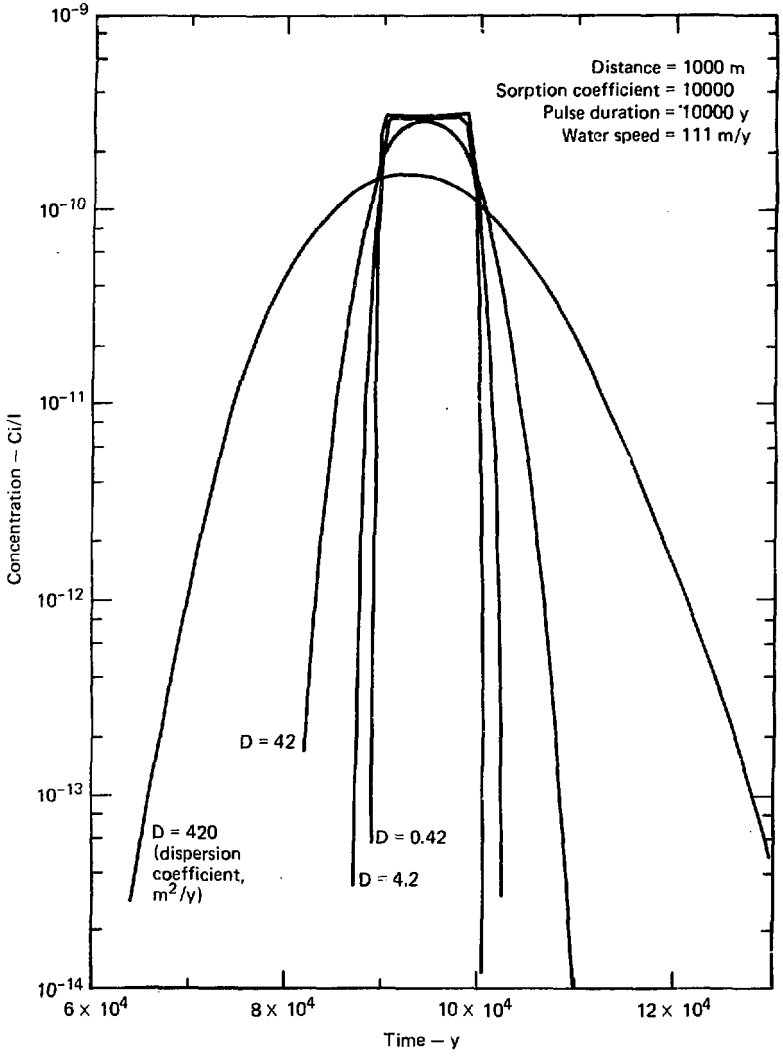


FIG. I2. Concentration at aquifer outlet vs time with dispersion coefficient as a parameter for ^{239}Pu.

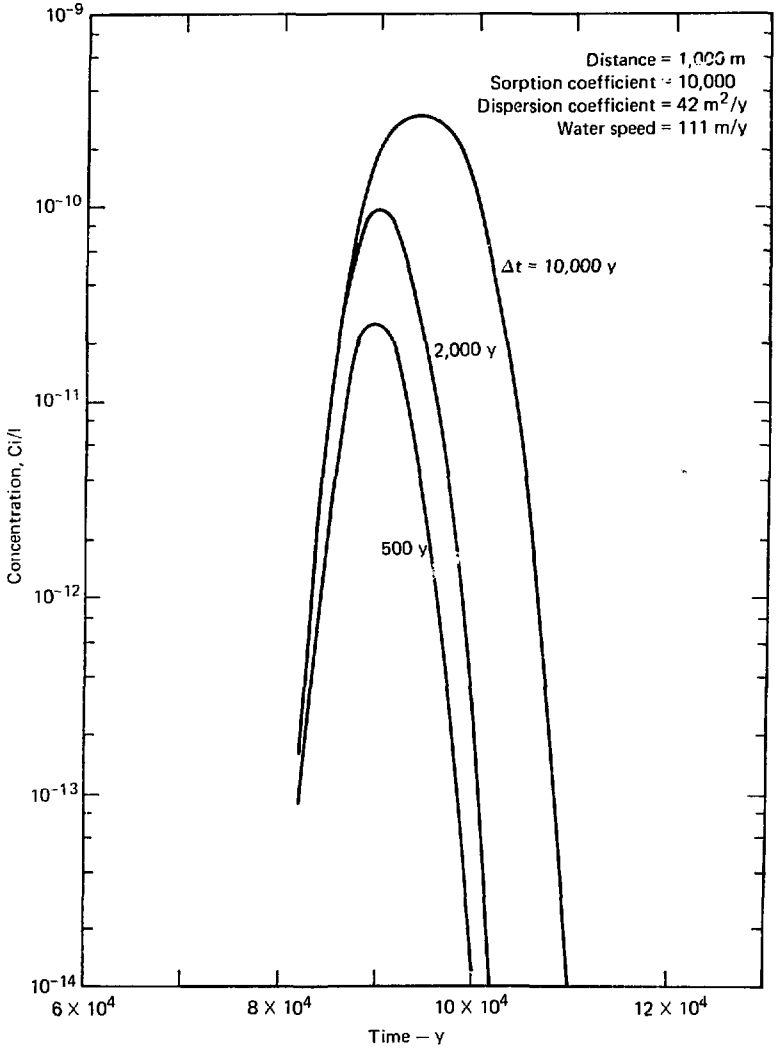


FIG. I3. Concentration at aquifer outlet vs time for ^{239}Pu with boundary condition pulse length as a parameter.

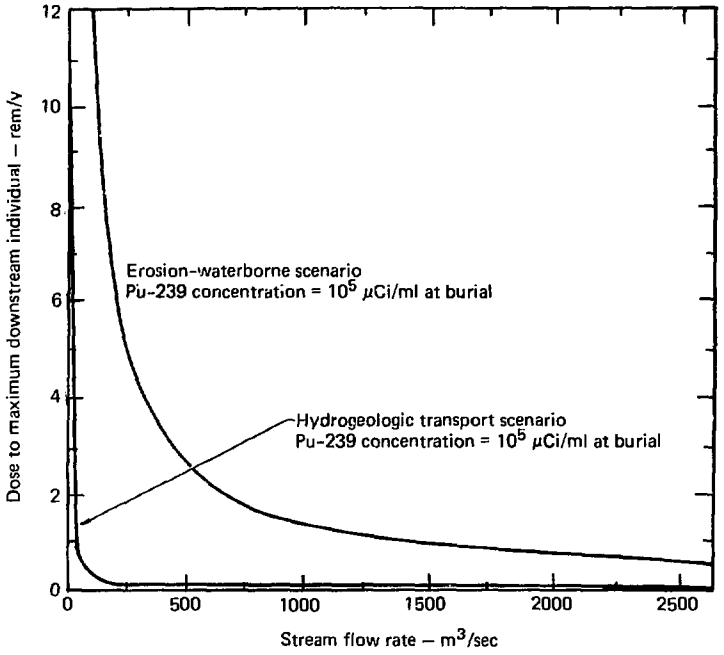


FIG. I4. Effects of water volumetric flow rate in stream on maximum individual doses.

DRILLED WELL SCENARIO

In this scenario, we assume that institutional control of the waste burial site is removed 100 y after burial operations have ceased. A well is drilled some time later into the aquifer under the RCF, and individuals then drink the water continually for 50 y. The question then is: What concentration of ^{239}Pu could be allowed in the buried waste such that persons drinking the contaminated water at the maximum concentration continually for 50 y would accumulate just enough ^{239}Pu in the body to produce an internal dose of 0.5 rem/y?

Based on the work of Isaacson et al.[18] on the transport of soil moisture, it is unlikely that any radionuclides will reach the aquifer if the radioactive burial site is properly located in an arid region. We shall assume for this scenario, however, that water percolates slowly from the surface through the buried waste into the aquifer. The transport processes include leaching of the material from the site of burial and downward movement through the soil, where the nuclide is sorbed and desorbed, reaching the aquifer where the concentration increases to a maximum depending on the various parameters used in the transport equation.

The following assumptions are used in the calculations:

- Average annual rainfall is 0.1 m/y.
- The fraction of precipitation that percolates through the buried waste is 0.001. The remainder goes to surface runoff, evaporation, and plant evapotranspiration.[18]
- The surface area of the RCF is 1.2×10^6 m^2.
- The volume of the waste burial pits is 6×10^5 m^3.
- The dilution factor of the waste with the soil is 3.
- The average concentration of the waste at burial is 1/10 of the peak concentration limit imposed.
- The depth of the adsorbing medium between the waste and the aquifer is 10 m.
- The average static volume of the aquifer directly under the RCF is 10^7 m^3.

- The annual flow of water in the aquifer entering the region directly under the RCF from adjacent areas is 1.1×10^6 m^3/y (based on a flow rate of 100 m/y and a cross sectional area of the aquifer of 10^4 m^2).
- Due to the effects of temperature, evaporation, and capillary action, the net effective downward transport velocity of the moisture is 1/3 m/y.

Since we are interested in finding only the maximum concentration of the nuclide in the aquifer, we shall ignore the effects of dispersion and convection as the nuclide passes through the adsorbing soil. This approach reduces the calculating effort, but it overestimates the value of the maximum concentration. The error introduced is on the conservative side, however.

The adsorbing medium slows up or delays the transport of the nuclide through the medium. As shown previously, the transit time for any given molecule of the nuclide through the medium is given by

$$t_a = \frac{K\,L}{V_w} \quad ,$$

where

$$K = 1 + \frac{\rho}{\varepsilon} K_d \quad ,$$

and

L = thickness of the adsorbing medium

V_w = average net transport velocity of the water through the medium

ρ = density of the medium

ε = porosity

K_d = distribution coefficient.

Eventually all the nuclides will pass through the adsorbing medium into the aquifer except for what is lost by radioactive decay during the delayed transit time. Since we have ignored the effects of dispersion and convection, the rate of flow of the nuclide into the aquifer will be equal to the rate of flow of the nuclide into the adsorbing medium, except that radioactive decay during the delay will appropriately reduce the quantity.

We shall assume a two-compartment model, with a delay line between the two, to describe the transport between the burial pits and the aquifer. If we let Q_A be the quantity, in Ci, of any radionuclide in the aquifer at any time t, the rate of change in Q_A per unit time is given by the equation

$$\frac{dQ_A}{dt} = \lambda_\ell Q_{B_o} \exp -(\lambda + \lambda_\ell)t \exp(-\lambda t_a) - (\lambda + \mu)Q_A ,$$

where

λ_ℓ = leach constant (fraction/y)

λ = radioactive decay constant (y^{-1})

μ = turnover rate in the aquifer resulting from inflowing water (y^{-1})

$$\mu = \left(\frac{f_r + f_a}{v_s}\right)$$

f_r = fraction of precipitation entering the aquifer (m^3/y)

f_a = aquifer flow from adjacent area (m^3/y)

v_s = static volume of the aquifer (m^3)

$$t_a = \frac{K\ L}{V_w}$$

Q_{B_o} = total quantity, in Ci, of the nuclide in the burial ground at t = 0.

Substituting the value of t_a and applying the boundary condition that $Q_A = 0$ at t = 0, the solution of the differential equation is

$$Q_A \approx \frac{\lambda_\ell Q_{B_o} e^{-\lambda \frac{K L}{V_w}}}{\mu - \lambda_\ell} \left[e^{-(\lambda+\lambda_\ell)t} - e^{-(\lambda+\mu)t} \right] . \qquad (11)$$

One must remember that Q_A, as given in the above equation, is the quantity of the radionuclide in the aquifer at a time $t + \frac{K L}{V_w}$. For any given molecule containing the radioactive atoms entering the adsorbing medium from the burial pit, a delay of $\frac{K L}{V_w}$ y occurs before that molecule moves into the aquifer.

After the delay time, the value of Q_A will increase to a maximum some time later depending on the parameters used in the equation. Figure I5 is a plot of Q_A/Q_{B_o} vs time after arrival of ^{239}Pu at the aquifer. In the scenario we used, one sees that the quantity of ^{239}Pu in the aquifer attains its maximum about 45 y after the ^{239}Pu first reaches the layer.

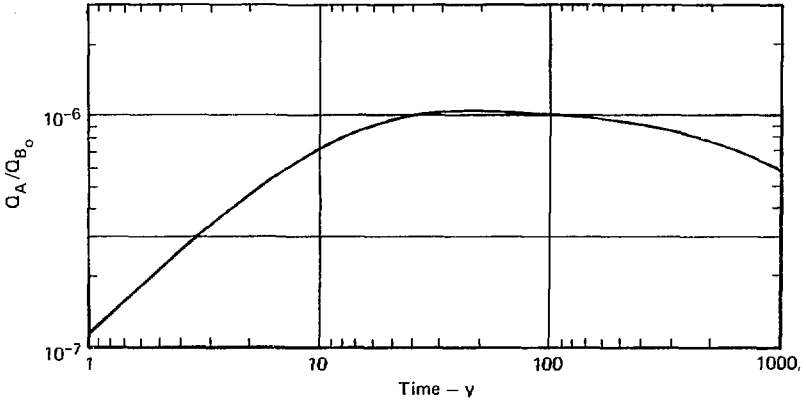


FIG. I5. Q_A/Q_{B_o} vs time for ^{239}Pu.

The concentration of the radionuclide in the well water equals the concentration in the aquifer. It is given by

$$C_A = \frac{Q_A}{v_s} \tag{I2}$$

If we now set C_A equal to the MPC_w (maximum permissible concentration in water) that will result in a dose equivalent of 0.5 rem/y after people have been drinking such water for 50 y and solve for Q_{B_o}, we get the amount or total inventory of the specific nuclide that can be permitted in the burial ground.

Equation I1 can be solved for the ratio Q_A/Q_{B_o}. Since

$$Q_A = \frac{Q_A}{Q_{B_o}} \times Q_{B_o} \quad ,$$

and since we want to set $MPC_w = C_A$, we can solve for Q_{B_o} as follows:

$$Q_{B_o} = \frac{MPC_w \times v_s}{(Q_A/Q_{B_o})} \quad , \tag{I3}$$

The peak concentration, C_c, permitted in the waste container at the time of burial is given by

$$C_c = \frac{Q_{B_o}}{v_b} \times 30 \quad ,$$

or

$$C_c = \frac{MPC_w \times v_s \times 30}{(Q_A/Q_{B_o}) \times v_b} \quad , \tag{I4}$$

where v_b = original volume of the buried waste (assumed to be 6×10^5 m^3).

With the previous assumptions, we substituted the following values in Eq. (I10) and Eq. (I14) to get the value of C_c for the ^{239}Pu:

$$\lambda = 2.84 \times 10^{-5}\ y^{-1} \qquad MPC_w = 1.7 \times 10^{-6}\ Ci/m^3$$

$$\lambda_\ell = 6 \times 10^{-4}\ y^{-1} \qquad v_s = 1 \times 10^{7}\ m^3$$

$$\lambda + \lambda_\ell = 6.3 \times 10^{-4}\ y^{-1} \qquad v_b = 6 \times 10^{5}\ m^3$$

$$K = 1 \times 10^{4}$$

$$\mu + \lambda = 1.1 \times 10^{-1}\ y^{-1} \qquad \mu = 1.1 \times 10^{-1}\ y^{-1}$$

$$\exp\left(-\lambda \frac{K\ L}{v_w}\right) = 1.99 \times 10^{-4}$$

$$\lambda_\ell/(\mu - \lambda_\ell) = 5.4 \times 10^{-3}$$

With these values, we calculate $Q_A/Q_{B_o} = 1.04 \times 10^{-6}$, which is the maximum value occurring 45 y after first arrival in the aquifer. Therefore, we see that the concentration, C_c, of ^{239}Pu permitted in the waste container at the time of burial is 820 $\mu Ci/cm^3$ in the well-water scenario, using the parameters we have assumed. The most critical parameter in the equations is the value of v_w, the net average velocity of the water in the soil. As mentioned previously, the work of Isaacson et al.[I8] would indicate that the value of v_w is equal to or very near zero for an arid region when the site is properly located. If so, the radioactive nuclides would never reach the level of the aquifer.

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APPENDIX J

APPLICATION OF THE RELATIVE HAZARD INDEX (RHI) CONCEPT FOR INDIVIDUAL NUCLIDES AND MIXTURES

Let C_1, C_2, C_3, ... C_n = the concentration of various nuclides in the waste. For a single nuclide, we have by definition

$$RHI = \frac{C\,K}{MPC_a},$$

where

C = concentration in μCi/ml

$K = \frac{\text{dust loading}}{\text{soil density}}$

MPC_a = maximum permissible concentration in air.

For any mixture of nuclides, the total hazard index is given by the sum of the individual RHIs for each nuclide; that is,

$$(RHI)_T = (RHI)_1 + (RHI)_2 + (RHI)_3 + \ldots (RHI)_n$$

The $(RHI)_T$ must not exceed the interface value of 1.7×10^4 as shown in Fig. 9. (An activity-weighted average half-life could be used to increase the acceptable total RHI. If the average half-life cannot be reasonably established, one should assume that it is long enough to give 1.7×10^4 for the total RHI.) Rewriting the above equation,

$$(RHI)_T = \frac{C_1\,K}{MPC_1} + \frac{C_2\,K}{MPC_2} + \frac{C_3\,K}{MPC_3} + \frac{C_n\,K}{MPC_n} \ldots \qquad (J1)$$

The interface concentration, here designated by C_I, is found by solving for C when given K and RHI_T.

$$C_I = \frac{RHI_T \; MPC_a}{K} \qquad (J2)$$

Rearranging terms in Eq. (J1) yields

$$\frac{C_1 \; K}{MPC_1 \; RHI_T} + \frac{C_2 \; K}{MPC_2 \; RHI_T} + \frac{C_3 \; K}{MPC_3 \; RHI_T} + \cdots \frac{C_n \; K}{MPC_n \; RHI_T} = 1 \quad . \qquad (J3)$$

Substituting Eq. (J2) into Eq. (J3) gives

$$\frac{C_1}{C_{I1}} + \frac{C_2}{C_{I2}} + \frac{C_3}{C_{I3}} + \cdots \frac{C_n}{C_{In}} = 1 \quad . \qquad (J4)$$

Therefore, if the sum of the left-hand side of Eq. (J4) is ≤ 1, the material may be considered low-level waste. For example, if the waste contains the following mixture of nuclides,

^{90}Sr	2.3×10^{2}	µCi/ml
^{90}Y	2.3×10^{2}	µCi/ml
^{137}Cs	8.8×10^{3}	µCi/ml
^{239}Pu	4.2×10^{-1}	µCi/ml

is the material low-level waste?

Using Eq. J6 with the values of C_I calculated from values in Table 9 yields:

$$\frac{2.3 \times 10^{2}}{2 \times 10^{3}} + \frac{2.3 \times 10^{2}}{2 \times 10^{5}} + \frac{8.8 \times 10^{3}}{4 \times 10^{4}} + \frac{4.2 \times 10^{-1}}{1.0} = y \quad .$$

y = 0.76, which is less than 1. The answer to the above question is yes; the material is low-level waste.