LA-9395-MS

c. 3

Overview Assessment of Nuclear Waste Management

Los Alamos
Los Alamos National Laboratory
Los Alamos, New Mexico 87545

Overview Assessment of
Nuclear Waste Management

B. W. Burton J. G. Steger
V. P. Gutschick T. K. Thompson
B. A. Perkins L. K. Trocki
C. L. Reynolds E. M. Wewerka
J. C. Rodgers M. L. Wheeler

CONTENTS

APPENDIX

DEFINITIONS OF TERMS

ADU	Ammonium diuranate	MAC	Maximum Acceptable Concentration
AEC	Atomic Energy Commission	MCD	Mined Cavity Disposal
BWIP	Basalt Waste Isolation Program	μCi	microcurie, 10^{-6} curies
BWR	Boiling-Water Moderated Reactor	MED	Manhattan Engineering District
CFR	Code of Federal Regulations	μg	Microgram, 10^{-6} gram
Ci	Curie	mℓ	milliliter, 10^{-3} liter
cm	Centimeter, 10^{-2} meter	μm	micrometer (micron), 10^{-6} meter
d	days	mm	millimeter, 10^{-3} meter
DDC	Dry Direct Conversion Process	MPC	Maximum Permissible Concentration
de minimus	Radioactivity level below which a waste may be handled as nonradioactive waste	mrem	millirem, 10^{-3} rem
		MT	metric ton
DOE	Department of Energy	MTU	metric tons of uranium
DOT	Department of Transportation	MWD	Megawatt days
EDP	Environmental Development Plan	NAS	National Academy of Science
EIS	Environmental Impact Statement	nCi	nanocurie, 10^{-9} curie
EMAD	Engine Maintenance Assembly/Disassembly Facility	NCRP	National Committee on Radiation Protection
EPA	Environmental Protection Agency	NEPA	National Environmental Policy Act
ERDA	Energy Research and Development Administration	NNWSI	Nevada Nuclear Waste Storage Investigations Project
ESE	Environmental Safety and Engineering Division	NPDES	National Pollutant Discharge Elimination System
GCD	Greater Confinement Disposal	NRC	Nuclear Regulatory Commission
GDP	Gaseous Diffusion Plant	NSD	Near-Surface Disposal
g	gram	NSTF	Near Surface Test Facility
HEPA	High-Efficiency Particulate Air Filter	NTS	Nevada Test Site
HLW	High-Level Waste	NWTS	National Waste Terminal Storage Program
hr	hour	ONWI	Office of Nuclear Waste Isolation
HW	Hanford, Washington	ORNL	Oak Ridge National Laboratory
ICRP	International Commission on Radiological Protection	OWI	Office of Waste Isolation
INEL	Idaho National Engineering Laboratory	pCi	picocurie, 10^{-12} curie
in situ	In (its original) place	PDCF	Pathway Dose Conversion Factor
in vivo	Within the living organism	ppm	parts per million
IRG	Interagency Review Group	PWR	Pressurized-Water Moderated Reactor
ISFS	Independent Spent Fuel Storage facility	R	Roentgen
kCi	kilocuries, 10^3 curies	RCF	Reference Containment Facility
km	kilometer, 10^3 meters	RCRA	Resource Conservation and Recovery Act
ℓ	liter	RCW	Recirculating Cooling Water system
Los Alamos	Los Alamos National Laboratory	SLB	Shallow Land Burial
LLNL	Lawrence Livermore National Laboratory	TRU	Transuranic
LLS	Low-Level Solids	TVA	Tennessee Valley Authority
LLW	Low-Level Waste	USGS	United States Geological Survey
L-S	Liquid Scintillation	W	Watts
LSA	Low Specific Activity	WIPP	Waste Isolation Pilot Plant
LWR	Light-Water Moderated Reactor	yr	year
m	meter		

OVERVIEW ASSESSMENT OF
NUCLEAR WASTE MANAGEMENT

by

B. W. Burton, V. P. Gutschick, B. A. Perkins, C. L. Reynolds, J. C. Rodgers,
J. G. Steger, T. K. Thompson, L. K. Trocki, E. M. Wewerka, and M. L. Wheeler

ABSTRACT

After reviewing the environmental control technologies associated with Department of Energy nuclear waste management programs, we have identified the most urgent problems requiring further action or follow-up. They are listed, in order of decreasing importance, below.

1. Shallow Land Disposal Technology Development

There is insufficient understanding of radionuclide release and transport mechanisms, which is compounded by the location of many burial sites in areas of complex geology. Regulatory standards and criteria seem to be conflicting and may not be adequate to provide safe disposal. In addition, there seems to be no coherent plan for ensuring site integrity after closure. We recommend further research in the areas of hydrogeologic conditions at existing sites and radionuclide release and transport mechanisms. A close evaluation of limiting concentrations in burial grounds (maximum and minimum allowable concentrations) is needed. Also needed are programs to address waste treatment and volume reduction. Land use evaluation schemes to be used in new facility siting should be developed, as well as a specific definition of low-level waste.

2. Active Uranium Mill Tailings Piles

Remedial action and the new Nuclear Regulatory Commission (NRC) licensing requirements addressing old and new tailings piles are promising, but their long-term effectiveness has yet to be proven. Environmental controls for currently active piles are not adequate, but these facilities are licensed and needed improvements may not be required. It is reasonable to assume that successful developments in the remedial action programs and NRC regulations regarding stabilization/rehabilitation at the close of operations will be applied to these active piles. However, currently active milling operations, which were not subject to the new NRC requirements at the time they were licensed, should be comprehensively studied to determine the extent to which their current practices are affecting the environment and to determine the cost/feasibility of ameliorative action.

3. Uranium Mine Dewatering

Pumpage of uranium mine water results in changes in the aquifer, dispersal of contaminants (both radioactive and toxic), and loss of water, itself a valuable resource.

Mine water control is not very effective and these activities are poorly documented. Environmental health and safety regulations and division of authority are not clear. Carefully coordinated field studies to determine the overall potential health and environmental impact of current uranium mining practices with particular attention to implications of the gaps in regulatory authority and enforcement capacity of both state and federal authorities should be undertaken.

4. Site Decommissioning

Some of the decontamination and site decommissioning activities scheduled for the near future (that is, decommissioning of the Gunite tanks at Oak Ridge National Laboratory and the plutonium facilities at Mound National Laboratory) are potentially quite hazardous. They will involve large quantities of radionuclides, will take place in populated areas, and are novel. Relevant documents do not address methods of establishing priorities, development of cleanup criteria, or methods for disposal of the waste. These issues must be addressed before operations are allowed to begin.

5. Exhumation/Treatment of Transuranic Waste at Idaho National Engineering Laboratory

This activity is discussed in some documents but not in others, so the current status of the project is not clear. The operations will be novel, the quantity of radionuclides involved is large, and the environmental controls are not known. This activity is potentially hazardous and, if it is still scheduled, the US Department of Energy (DOE) should ensure careful planning and adequate environmental controls before exhumation is allowed to begin.

6. Uranium Mine Spoils

Mine spoils piles are a potential source of wind- and waterborne contamination (both radioactive and toxic). There are very little data available regarding mine spoils. Although spoils piles are poorly regulated, an effective reclamation program can probably be developed under the Resource Conservation and Recovery Act (RCRA). These spoils piles are outside DOE jurisdiction, but DOE should pay close attention to developments in this area.

7. Medical/Institutional Wastes

Large volumes of very low activity wastes are produced by the medical/institutional community. Enforcement of environmental controls has been ineffective, and much of the waste is disposed to municipal sewer and refuse systems. Because of the low activities involved, the hazards of institutional wastes are biological and chemical rather than radiological. Large volumes of toxic and organic liquids are buried in commercial low-level burial grounds, presenting the potential hazard of chelating and mobilizing radionuclides. Although these waste generators are not under DOE jurisdiction, there is a compelling need for increased development of waste treatment and volume reduction systems.

We feel that the following areas need not be of immediate concern.

- *Conversion/Enrichment/Fabrication.* Because of the small amounts of waste produced by these facilities, we feel that they should receive a low priority. These facilities have been in operation for some time and no environmental releases of any consequence have been reported. We perceive the major problem to be on-site holding ponds, but these will not become a concern until the facilities are decommissioned.

- *Reactor/Fuel Storage Operations.* These operations have been under licensed supervision for many years and we feel that there is no immediate problem in these areas other than the need for more efficient waste volume reduction practices at reactors.

- *Terminal Isolation.* It seems to us that postoperational monitoring and accident recovery are not adequately covered. There may be conflicts in the guidelines. We are somewhat concerned about the potential misuse of probability estimates of future events. These programs are relatively new, however, and we feel that it is probably too early to assess the potential control technologies.

I. INTRODUCTION

In mid-1979, the Environmental Safety and Engineering (ESE) Division inquired into Los Alamos National Laboratory's interest in performing an assessment of environmental control technology in nuclear waste management. After some discussion, we began our evaluation in October 1979. Our role in this assessment is to assist ESE to conduct an independent evaluation of the environmental control technologies applied to nuclear waste management and to advise ESE where these controls seem to be inadequate or where further studies are needed to determine adequacy.

Our initial approach was to assemble relevant information on nuclear waste generation, US Department of Energy (DOE) waste management programs, and the results of previous evaluations. This report covers activities to date, provides preliminary conclusions, and makes recommendations for future work.

There is considerable activity in the waste management field (20 000 references in TID-3311) and the rate of this activity is increasing. After the initial program was underway, actions by the Administration and Congress to balance the budget caused a substantial rearrangement of the waste management program. The ultimate effects are still unclear. The Nuclear Regulatory Commission (NRC) has released their draft 10 CFR Parts 60 and 61, containing the criteria for disposal of high-level and low-level waste, respectively. These documents are not final, and any attempts to evaluate activities against their requirements must be preliminary. Another interesting event that occurred during this initial assessment was the assignment of a large role to the public in the decision making process by the President's message to Congress on waste management of February 12, 1980. The effects of this event, as well as the establishment of the State Planning Council, cannot be adequately evaluated at the present time.

In many of the areas we reviewed, information from one source may be inconsistent with that from another source. In some cases the data were totally unusable. We did not attempt to resolve all of the conflicts in the data during this overview assessment; that will be addressed in more detailed individual assessments. Thus, there may be minor inconsistencies (for example, compare Tables XVIII and XXVII).

To ensure a broad perspective we assembled a team of scientists to take part in this initial evaluation. Before we could begin a well-coordinated overview assessment, we had to make several assumptions and decisions from which to work. Some of these are:

1. Evaluations must be clinical, and not based upon opinion.

2. Effects of nonradiological constituents in the waste must also be considered.

3. Criteria used will be limited to those already existing or are proposed by appropriate agencies.

4. Occupational effects will not be addressed.

5. Activities not under DOE jurisdiction should only be monitored.

An additional factor that must be treated carefully is the position taken by Congress that DOE weapons facilities will not be licensed. There is a remarkable lack of available data from unlicensed facilities compared to that from licensed facilities. Because of this, no assessment of any weapons facility is now planned.

We reviewed relevant planning, program, and budget documents to ensure that important factors are not being overlooked during the planning stages.

Our review of the nuclear fuel cycle is somewhat detailed. We feel that this level of detail is necessary because the fuel cycle produces the largest volume of waste and a superficial evaluation is not sufficient. We also included an assessment of medical/institutional wastes. Institutions produce a substantial volume of waste, but have not attracted much attention from regulators.

Waste disposal methods were also treated in some detail. If the ultimate disposition of all radioactive wastes is burial (either in shallow trenches or deep geologic formations), then the adequacy of burial as an environmental control technology must be carefully evaluated. We reviewed existing and proposed standards and criteria relevant to waste disposal because these are the measures against which adequacy is judged. We must ensure that standards and criteria are appropriate to the technologies being evaluated.

II. DOE WASTE MANAGEMENT PLANS. PROGRAM AND BUDGET DOCUMENTS*

A. Introduction

Because the waste management program is controlled by its program documents, these documents were evaluated to determine if they were consistent, comprehensive, and emphasized the most important areas.

---

*Source document is the FY '81 DOE Waste Management Budget submission.

This evaluation was based on the waste management environmental development plans (EDP Commercial Waste, 1980; EDP Defense Waste, 1980), program plans (DOE/NE-0008, 1980), and budget (DOE Budget, 1980). We did not have the decommissioning and decontamination EDPs to review, therefore some of our conclusions may not be relevant.

Because the documents we evaluated are similar, they were considered as one unit. Any comment about one is equally appropriate to all of them.

B. Commercial Waste Management

1. Terminal Isolation R&D

The terminal isolation program provides research and development for identification of repository sites and for the development of technology necessary for design, licensing, and operation of a repository. The repository will be designed to accept spent fuel or solidified high-level waste and transuranic wastes. The primary emphasis is directed toward the disposal of unreprocessed spent fuel into geologic facilities, and the development of facilities for packaging spent fuel elements.

Transportation issues are also addressed as part of the program through the Transportation Technology and Information Center (Albuquerque Operations Office/Sandia Laboratory). Detailed characterization of specific potential repository sites will be continued in several geologic environments. Program resources are managed to develop a list of fully characterized sites from which one can be chosen for development of the first regional repository for high-level waste. It is anticipated that the first commercial repository could be operational in the 1990-1995 time period. Studies of salt formations include consideration of salt domes and bedded salt. The interior Gulf Coast domes (Mississippi, Louisiana, and Texas) are of interest, as are bedded salt portions of the Permian (Texas, New Mexico), Paradox (Utah), and Salina (Michigan, Ohio, and New York) Basins. Another major effort is directed toward the geologic and hydrologic evaluation of the thick basalt flows underlying the Columbia River Plateau, with emphasis on the Hanford Site. Following identification of multiple candidate sites in early FY 1980, a suitable site could be identified during FY 1981. A third project will continue to determine if a potential site location exists at NTS. The project has examined Syncline Ridge (argillite), Calico Hills (granite and shale), Walmonie

Stock (granite), and Yucca Mountain (tuff). Work on all but one site has been discontinued because of technical questions regarding the structural integrity of the formations. Yucca Mountain remains under active consideration. Its suitability has been investigated through a series of geologic and hydrologic holes drilled in FY 1980 and FY 1981. A fourth project will determine the technical and environmental feasibility and develop the technological capability for disposal of high-level wastes or spent nuclear fuel by implantation beneath the ocean floor. The areas under investigation are low in biological activity and essentially devoid of natural resources.

2. Waste Treatment Technology

This program is responsible for the development of technology for immobilizing waste from nuclear fuel cycles and converting it to forms that provide safety and economy of management, satisfy regulatory requirements, and are acceptable for disposal. The processes deal with high-level and low-level liquid, low-level solid, and TRU-contaminated solid and gaseous wastes. Each of these waste types requires processes that reduce the volume and/or convert the materials into more stable, less dispersible forms for safe transport and/or long-term isolation. The technology generally is not specific to a given fuel cycle. Much of the technology is similar to that required for defense waste immobilization and is being developed in close coordination with related defense waste management programs to assure timely transfer of knowledge and experience. Airborne waste technology development is concerned with the immobilization, containment, and safe storage of radioactive gases to reduce releases of these gases to the atmosphere. The principal technologies under development are methods for storage of krypton-85 and processes for immobilizing other volatile emissions from spent fuel and the thorium fuel cycle. High-level waste technology is to develop processes that satisfy the provisions of 10 CFR 50 that high-level waste be solidified before it is shipped to a Federal repository for terminal storage. Processing produces less dispersible, more stable forms such as crystalline forms, glass, concrete, etc. Processes for producing various waste forms are being adapted to existing commercial wastes. The low-level waste program is to develop technology for treatment and disposal of low-level waste and make this technology available to the private sector. Also included is comprehensive national planning for an adequate number of regional disposal sites for commercial low-level wastes under state management. Technology development for solid waste contaminated with transuranic elements (TRU waste) focuses on incorporating contaminated wastes into forms acceptable for safe storage, transportation, and disposal.

3. Remedial Actions

The objectives of the remedial actions program under the Office of Nuclear Waste Management are to keep radioactively contaminated sites and facilities (government-owned and designated non-government) that are no longer required or used, from becoming an actual or potential health, safety, or environmental hazard, and to release property for alternative productive use with minimum or no restrictions. Government-owned property that is now surplus to programmatic needs and private property contaminated by government-contracted activities are included in the remedial action program. The major program areas are (1) remedial action at formerly used Manhattan Engineering District and Atomic Energy Commission (MED/AEC) sites, (2) remedial action at Grand Junction, Colorado, (3) remedial action at inactive uranium mill tailings sites, (4) management of all radioactively contaminated DOE facilities declared surplus before October 1, 1976 and management of surplus radioactively contaminated DOE facilities from Nuclear Energy Programs subsequent to 1976, and (5) remedial actions at the Nuclear Fuel Services Facility at West Valley, New York.

C. Defense Waste Management

1. Decontamination and Decommissioning

The defense decontamination and decommissioning (D&D) program is to provide centralized planning for decommissioning of surplus facilities, to conduct engineering, safety, and environmental studies on specific projects, and to undertake required disposition of surplus facilities used for defense program activities. Disposition projects under this program in FY '81 include resumption of D&D activities at Mound Facility Buildings PP and R to coincide with defense program plans, and initiation of D&D for INEL Process Cell, ORNL Waste Facilities, and Savannah River high-level waste tank decommissioning.

2. Interim Waste Operations

The interim waste operations program provides safe handling and storage or disposal of DOE radioactive waste pending implementation of the long-term waste management program. The program involves operation and maintenance of high-level waste (HLW) storage tanks, bins, and related processing and storage facilities at Savannah River, Richland, and Idaho; the operation and maintenance of low-level waste burial grounds at Savannah River, ORNL, INEL, Hanford, Los Alamos National Laboratory, Sandia, and Nevada Test Site (NTS); the placement of transuranic wastes into retrievable storage at various sites; the operation of low- and intermediate-level liquid waste facilities; and the handling of low-level gaseous waste streams. Although the basic waste operations continue, efforts are also underway to upgrade the waste handling, storage, and disposal operations, such as transfer of HLW, from old existing tanks at Savannah River and Hanford into new double-shell tanks, construction of new improved facilities, and improving burial ground operations.

3. Long-Term Waste Management Technology

The long-term waste management technology program is directed toward the development and evaluation of alternatives for the long-term management of DOE radioactive wastes. It consists of developing alternative technologies for each waste type and assessing the associated hazards, costs, and environmental impact to select and implement the optimum long-term disposition alternative. In FY '81 the program has continued development of various site–specific waste processing technologies and has emphasized the development and evaluation of various final waste forms other than glass for high-level waste.

a. High-Level Waste Technology. The high-level waste technology program is to provide for the selection and implementation of alternatives for the long-term disposition of DOE high-level waste at Savannah River, Richland, and Idaho. Efforts before FY '80 focused on a limited number of final waste forms, primarily monolithic borosilicate glass. Comments and criticism from the scientific community and recommendations by the Interagency Review Group (IRG) on nuclear waste management led to an expanded program in FY '80 to pursue development of additional alternative waste forms to the degree required to provide a sound technical and engineering choice of the final waste form(s) most suitable for defense high-level waste. This accelerated effort will be further expanded in FY '81.

b. TRU Waste Technology. The components of this activity are to (1) conduct analyses of the alternative available for long-term management of DOE's buried and retrievably stored TRU waste, (2) develop technology to treat DOE TRU wastes and demonstrate that they can be placed in a form suitable for safe storage, transportation, and final disposal, (3) reach decisions on the proper final disposition of these wastes, (4) design and construct TRU waste handling and treatment facilities, and (5) reduce volume generation rates at current operating sites.

c. Low-Level Waste Technology. This activity includes (1) development of improved handling, treatment, storage, disposal, and monitoring techniques for low-level solid waste to minimize reliance on long-term maintenance and surveillance, (2) final criteria for disposal by shallow land burial, (3) development of methods for reducing the volumes of waste generated, and (4) development of alternatives to shallow land burial.

d. Airborne Waste Technology. The airborne waste technology program is to develop technology needed to meet current and projected requirements for removing airborne waste from effluents of DOE-operated nuclear plants, to assure the quality of filter systems used in DOE facilities, and to provide methods of preparing the waste product for long-term management.

4. Transportation R&D

The consolidation of DOE's nuclear materials transportation activities enables an integrated systems approach to be used in transportation planning and execution. Waste transportation activities in support of DOE programs include the design, development, testing, and procurement of DOE waste transport systems; development of transportation and information data bases; risk and environmental impact analyses; development of transportation interface parameters for input to waste acceptance criteria, facility designs, handling procedures, and hardware design; and continued coordinated testing and evaluation of current and future generation packaging systems under normal and abnormal conditions.

D. Conclusions

As a result of our review, we have concluded that the documents controlling the DOE Waste Management Program are clear, well done, coordinated, and comprehensive. They are quite adequate for their intended purposes. We have not received a copy of the decommissioning and decontamination EDP. The following comments may be modified after our review.

1. The programs all seem to be oriented to disposal only. This is too narrow. The ever increasing cost and consumption of resources required by waste disposal could exceed public acceptance and lead to the curtailment of the program. We think the goals should be expanded to include such things as recycle, use of natural phenomena to separate resources, storage of resources, multiuse facilities, codisposal with other wastes, etc. Waste material is a national problem, and it is questionable that disposal alone can provide a solution.

2. There are no coherent programs to develop criteria and standards. Much effort is being expended on developing technology, but almost no effort on developing the criteria and standards necessary to determine the adequacy of the technology. This is especially true in the overlapping areas of low-level waste disposal, remedial action, and decommissioning and decontamination. The existing criteria and standards are at best inadequate, contradictory, inappropriate at times, and vague. Criteria and standards will ultimately affect all aspects of waste management and therefore must not be allowed to develop in a haphazard manner. A formal criteria and standards development project should be started.

3. The treatment of the commercial LLW issue was not complete in the documents. More work is in progress than is reflected. Because commercial LLW is quite an issue, it would probably be worth the effort to provide better coverage of the area.

4. Most of the DOE waste management activities do not seem to be a cause for concern for one or more of the following reasons.

   a. Technologies are still under development and will not be available for some time (HLW solidification, geologic disposal, etc.).

   b. The activities are routine and remote from the public (HLW tank storage, TRU pad storage, burial of LLW).

   c. Only low levels of radioactivity are involved (remedial action).

We recommend several activities for a close review because of the following factors: the operations are novel and the experience base is weak, a substantial inventory of radionuclides is involved, they will establish precedents for many other activities, they will take place in populated areas, and previous releases from the sites involved have received some adverse publicity. Our concerns are based upon the *potential* for problems and not a prediction that something is wrong. We recommend a close look before the fact rather than trying to explain what went wrong after the fact. The activities of concern are

★ decommissioning of West Valley,

★ decommissioning the Gunite tanks at ORNL,

★ decommissioning the plutonium facilities at Mound, and

★ The exhumation of TRU wastes at INEL.

III. NUCLEAR FUEL CYCLE

A. Introduction

Because the nuclear fuel cycle is the largest source of radioactive waste, we feel it is necessary to consider each compartment in some detail. The use of nuclear energy to generate electric power is a complex process, involving many individual steps, beginning with the extraction of uranium ore and ending with the disposal of radioactive wastes. It should be noted that, until some form of terminal disposal (geologic, seabed, etc.) becomes a reality, the nuclear fuel cycle is incomplete. Although the fuel cycle is often considered as a whole (and sometimes identified with reactors only) each individual step is a separate entity with its own process and unique environmental effects.

Figure 1 shows the commercial nuclear fuel cycle as it is presently conceived. There are some differences between the commercial and military fuel cycles, though some facilities (such as enrichment plants) are common to both. For example, military fuels may be enriched to a much higher $^{235}U/^{238}U$ ratio and spent fuel from govern-

Fig. 1. Nuclear fuel cycle.

concentration of U_3O_8. Most uranium mills use an acid leach process, but in some mills an alkaline leach solution is used. The waste rock (tailings) is slurried with process water and impounded on site. The decay of radium in the tailings produces radon gas. Radon diffusion is probably the greatest single contributor to risk from mill tailings.

After milling, the U_3O_8 must be converted to UF_6 for enrichment. Conversion is accomplished by either of two processes. The hydrofluor, or "dry" process consists of continuous, successive reduction, hydrofluorination, and fluorination of ore concentrates, followed by fractional distillation to purify the crude UF_6. The "wet" process uses a wet chemical solvent extraction step to purify the uranium feed before reaction to UF_6; distillation is not used. Wastes are largely composed of CaF_2 ash from dry process plants and sludges from wet process plants. Both types of conversion plants produce CaF_2 chemical wastes by heating scrub liquors. Conversion plant solid wastes may be buried on site or shipped to a commercial burial ground. Liquid wastes containing uranium precipitates are ponded on site.

Before uranium can be used as fuel in a nuclear reactor it must be enriched in fissionable ^{235}U. The current technology used for uranium enrichment is gaseous diffusion. At a given temperature, volatile $^{235}UF_6$ diffuses through a porous barrier at a slightly higher rate than $^{238}U_6$. By allowing UF_6 to diffuse through many successive stages, a product highly enriched in ^{235}U can be obtained. Gaseous diffusion plant wastes are primarily from decontamination and uranium recovery operations. Solid wastes are buried on site. Liquid wastes are collected in holding ponds to allow precipitation of uranium compounds. Holding pond sludges are periodically removed and buried on site.

At the uranium fuel fabrication plant UF_6 is converted to UO_2, which is formed into pellets and loaded into fuel rods. The fuel rods are then made into fuel assemblies, ready for use in the reactor. Two processes are used to convert UF_6 to UO_2. The ammonium diuranate (ADU), or "wet" process, involves hydrolysis of UF_6 with water to form UO_2F_2 in solution, followed by precipitation with NH_4OH, yielding ADU. The ADU is calcined by heating in a partially reducing atmosphere to form UO_2. In the dry direct conversion (DDC) process, UF_6 is hydrolyzed by steam in a fluidized bed rather in bulk solution. The resulting UO_2F_2 is reduced with H_2 in a fluidized bed. The DDC process generates smaller volumes of liquid waste, but is used less commonly. The most significant fabrication plant waste is CaF_2 sludge, produced by treatment of liquid wastes.

ment-owned reactors is recycled. No Federal repository for high-level and transuranic wastes or spent commercial reactor fuel currently exists. Efforts to identify a suitable repository site are discussed in Sec. V.

The first step in the nuclear fuel cycle is uranium mining. Uranium ore is obtained primarily by conventional underground and open pit mining methods, though other methods such as solution mining, heap leaching, and uranium recovery from low-grade stockpiles have become increasingly popular in recent years. The major wastes produced by uranium mining operations are overburden from open pit mines, waste rock (gangue) from underground mines, and water from mine dewatering activities. Waste rock and overburden from some mines may have high concentrations of nonradioactive toxic elements.

Uranium ore is sent from the mine to a mill for processing. The ore is processed by crushing or grinding, chemical dissolution (leaching), physical separation of the dissolved ore from the gangue, and separation and

The fuel assemblies are transferred to reactors for use in generation of electric power. Power reactors in current use are light-water moderated reactors (LWRs). Light water reactors are classified as boiling water reactors (BWRs) or pressurized water reactors (PWRs). Wastes from these two types of reactors differ in both quantity and radioactivity. The radionuclides contaminating reactor wastes are fission products from the fissioning of the uranium fuel, activation products from exposure of fuel assembly hardware to high neutron flux in the reactor core, and transuranic elements produced by neutron capture in the fuel matrix. The primary wastes are ion exchange resins, filters and filter sludges, evaporator bottoms produced by cooling water cleanup, and compactible and noncompactible dry solid wases contaminated during reactor operation. Commercial power reactor wastes are shipped for commercial burial. Wastes from government-owned reactors may be buried in commercial or DOE burial grounds.

An additional waste stream from commercial reactors consists of spent fuel assemblies. Currently, spent fuel is stored in water basins, either at the reactor site or at Independent Spent Fuel Storage (ISFS) facilities. Storing spent fuel under water provides cooling and shielding until a Federal repository becomes available. Wastes produced by fuel storage operations are essentially the same as those produced at reactors. Reactor basin wastes are included in general reactor wastes. Wastes from ISFS facilities may be buried commercially or stored on site.

Most radioactive wastes generated by the commercial nuclear fuel cycle are classified as low level and are buried in commercial or on-site shallow land burial facilities. The term low-level is somewhat misleading because the activity in some of these wastes may be quite high. Shallow land burial may be inappropriate. Transuranic wastes are produced during reprocessing of spent reactor fuel and by DOE research- and weapons-related activities. TRU wastes are presently stored retrievably, awaiting the opening of a repository. High-level wastes are generated by fuel reprocessing and are, like TRU, limited to government sources. High-level wastes are currently stored at DOE facilities until a repository becomes available. Shallow land burial and efforts to develop a suitable Federal repository are discussed in Sec. V.

B. Uranium Mining

1. Introduction

Uranium is required for nuclear power reactors and the Nation's defense programs. The recovery of uranium ore is accomplished primarily by conventional underground and open pit mining. Underground mines produced 30% of the recovered uranium metal in 1979 and open pit mines produced 45% (GJO-100, 1980). Although open pit and underground mining have accounted for the overwhelming share (75 to 98%) of the uranium ore extraction over the last five years (GJO-100, 1976-1980), it is significant that during this same period other methods (heap leach, mine water, solution mining, low-grade stockpiles) have increased their share of the total annual concentrate (expressed as U_3O_8) production from 2% (GJO-100, 1976) during 1975 to 25% (GJO-100, 1980) during 1979. New Mexico and Wyoming produced 62% of all ore mined in the United States in 1979, whereas Arizona, California, Colorado, Florida, Texas, Utah, and Washington combined produced the remaining 38% (GJO-100, 1980). It is estimated that New Mexico, Wyoming, Texas, Arizona, Colorado, and Utah have ore reserves that account for 97% of the total U_3O_8 recoverable at a forward cost of \$66/kg U_3O_8 using 1979 dollars. Of the estimated \$66/kg U_3O_8 reserves, 65% is projected to come from underground mines, of which approximately 45% will come from depths greater than 240 m (GJO-100, 1980). In comparison, only 19% of the 1977 production came from depths greater than 240 m (GJO-100, 1978).

2. Process

Both underground and open pit uranium mining are accomplished using conventional technologies established over many decades. Open pit mining is generally used for ore deposits less than 120 m below the surface (Clark, 1974). Overburden and ore are removed by heavy equipment after blasting has loosened well-consolidated formations. If blasting is not required, shovels, loaders, and draglines can be used to remove the overburden. Open pit uranium mines range in size from 11 km² of disturbed land at the world's largest open pit uranium mine (New Mexico Energy and Minerals De-

partment, 1979) to very small mines, which are worked using backhoes (Clark, 1974).

Underground uranium mining is generally necessary if the ore deposit is deeper than 120 m (Clark, 1974). Compared with open pit mines, the surface land commitment for an underground mine is small, approximately 0.15 km², and includes the mine head frame, ore storage and loading facilities, associated buildings, mine water treatment and settling ponds, and waste rock disposal areas (New Mexico Energy and Minerals Department, 1979).

Besides the conventional underground and open pit technologies, uranium is also being recovered by *in situ* leaching of ore bodies, heap leaching on the surface of low-grade ores, mine residues or mill wastes, and leaching of residues in mined-out stopes. *In situ* leaching can be accomplished by naturally occurring ground waters that are pumped to the surface from mines and processed through ion exchange resins to remove the solubilized uranium, or with the controlled pumping of acid or alkaline solutions, which contain oxidants such as sodium chlorate, oxygen, or hydrogen peroxide, into the ore body (Merrit, 1971). This solution containing solubilized uranium is then recovered from other wells in the field and processed for recovery of the uranium. A more detailed discussion of the process can be found in "Environmental Development Plan, Uranium Mining, Milling and Conversion," U.S. Department of Energy, DOE/EDP-0058, August 1979.

The heap leach process generally involves the leaching of open or confined piles of low-grade ores that have previously been stockpiled on the surface. The low-grade ore is usually arranged in an array of piles, each of which may be approximately 100 m long, 6 to 8 m high, and 20 m wide. Retaining ridges, or berms, separate the piles in the array and are used to form ponds on top of the piles. From the ponds, a leach solution (most commonly sulfuric acid, but sometimes sodium or ammonium carbonate) percolates through the ore, solubilizes the uranium, and is collected by a network of perforated collection pipes underlying the piles and collection troughs at the toe of the piles (DOE/EDP-0058, 1979).

Leaching of residual uranium values from mined-out stopes need not be discussed separately as the process impacts and controls are similar to those encountered in recovery of uranium from mine pumpage and *in situ* leaching.

3. Wastes Produced

The major wastes from mining include overburden from open pit mines, waste rock from underground mines, and water pumped from the mines. Information regarding mine wastes is scant but does indicate the scale of the disposal problem, although not the severity. It is reported that the overburden-to-ore ratio for open pit mines varies from 8:1 to 35:1 (Clark, 1974; Reynolds et al., 1976) and that the waste rock-to-ore ratio for an underground mine varies from 1:20 to 1:1 (DOE/EDP, 1979). Using 1979 production figures, this amounts to between 70 and 307 million metric tons of overburden from open pit mines and between 0.24 and 4.9 million metric tons of waste rock from underground mines during the 1979 production year alone. Using 1979 ore and U_3O_8 production figures (GJO-100, 1980) and the above ranges of overburden- and waste rock-to-ore ratios, we calculate that for open pit uranium mines, between 10 000 and 42 000 metric tons of overburden are stripped for every metric ton of uranium metal produced. At underground uranium mines, between 50 and 1000 metric tons of waste rock are removed for every metric ton of uranium metal produced. It should also be noted that, of the uranium metal produced, only 0.71% is fissionable ^{235}U. The chemistry of this overburden and waste rock is not well documented. It is not likely to have significant activity for it would then be stockpiled as low grade ore. However, some waste rock and overburden have been found to contain trace elements (for example, Se, As, Mo) in concentrations exceeding those of the ore (Squyres, 1970). Adverse effects on soils, plants (Dreesen et al., 1978), and cattle (Dollahite et al, 1972; Chappel, 1975) have been noted in the vicinity of surface storage areas for uranium mine spoils and ore. Another study (New Mexico Energy and Minerals Dept., January 1979) states "It appears from preliminary surveys that because of air and wind transport there are areas of fenced mine property where the concentrations of gamma emitting radionuclides is sufficient to cause such excessive gamma levels that people occupying these areas continuously would receive above the maximum allowed whole body dose for unrestricted areas."

Underground uranium mines are becoming increasingly deeper, which increases the amount of water pumped from the mines. It is reported that dewatering rates for underground mines vary from 760 to 11 350

ℓ/min (Clark, 1974). Although some underground mines are dry, these figures are bracketed by very recent data (New Mexico Health and Environment Dept., July 1980) collected for New Mexico uranium mines, which indicate water production rates from 76 to 14 380 ℓ/min for active underground uranium mines and from 380 to 15 140 ℓ/min for underground uranium mines being developed. Water quality data for uranium mine pumpage have limited availability. However, Table I, compiled from six sources, gives an indication of the variation encountered.

Well-documented, conclusive information regarding possible environmental effects of uranium mine water pumpage is sparse, if it exists at all. However, a draft memo report from the Public Health Service, Center for Disease Control (1980) indicates a potential problem. This report was generated after the New Mexico Envi-ronmental Improvement Division requested assistance in assessing the health implications of a large uranium mill tailings pond dam break that occurred in July 1979. Among many other samples collected and analyzed were tissue samples from local livestock. Although certainly not conclusive, the report states "The animal radio-nuclide concentrations may be due to chronic exposure to radionuclides in the Rio Puerco, in soil, on vegetation and in air. This point is supported by the fact that the cow analyzed....had comparatively high radionuclide concentrations, but was exposed to [uranium mine] dewatering effluent rather than spill materials."

Because *in situ* extraction of uranium does not involve bringing the ore to the surface, volumes of waste are greatly reduced. However, because this operation typi-cally involves the pumping of more solution from the ore body than is injected, and because the buildup of

TABLE I

CHEMICAL ANALYSES OF URANIUM MINE WATERS[a]
(Data in mg/ℓ unless otherwise noted)

pH (units)	6.6 - 8.8	Chloride	9 - 1597
Conductivity (μmhos)	589 - 2456	Copper	0.1
Solids, total	612 - 3712	Fluoride	1.28
Solids, suspended	<1.0 - 420[b]	Iron	0.16
Solids, dissolved	363 - 2962	Lead	<0.005 - 0.017
Solids, volatile	38 - 164	Magnesium	5 - 106
Hardness	181	Molybdenum	<0.01 - 1.33
Alkalinity, total	420	Phosphorus, total as P	0.05 - 2.3
Alkalinity, phenol	7.5	Potassium	1.17 - 12.1
BOD 5-day	0 - 67	Selenium	0.01 - 1.0
COD	0 - 2.4	Sodium	28 - 338
Kjeldahl nitrogen	0.22 - 1.42	Silica	14.2
Nitrate (as N)	0 - 1.06	Vanadium	<0.01 - 1.4
Ammonia (as N)	0.22 - 1.60	Zinc	<0.01 - 0.3
Bicarbonate	33 - 441		
Phosphate	0.20	Total α (pCi/ℓ)	54 - 5300
Sulphate	61 - 922	Total β (pCi/ℓ)	77 - 168
		Lead-210 (pCi/ℓ)	9.7 - 15
Aluminum	<0.250 - 16	Polonium-210 (pCi/ℓ)	2.2 × 10⁻⁵
Arsenic	<0.005 - 0.202	Radium-226 (pCi/ℓ)	0.1 - 460[b]
Barium	<0.10 - 2.13	Thorium-230 (pCi/ℓ)	6
Cadmium	<0.001 - 0.008	Uranium (pCi/ℓ)	17 - 4900[b]
Calcium	5 - 222		

[a]Compiled from Tsivoglou and O'Connell, 1962, Texas Dept. of Health, 1977, Whicker, 1972, Kaufmann et al., 1976, USAEC, 1974, New Mexico Health and Environment Dept., 1980.
[b]In excess of EPA effluent limitations guidelines for existing sources, uranium, radium, and vanadium ores subcategory (Federal Register, Vol. 3, No. 133, July 11, 1978).

dissolved solids can interfere with the dissolution process, a bleed is usually required. The bleed solution is ponded for evaporation or injected into wells for disposal and is expected to contain constituents such as arsenic, selenium (Rouse, October 1978), and molybdenum (Wienke, July 1979).* If leach solution escapes the mining zone, surrounding ground water can become contaminated. Other wastes include spent ion exchange resins and brine from ground water restoration operations.

After heap leach operations, the remaining barren ore contains ^{226}Ra and ^{230}Th and is therefore a source of radon, as well as airborne particulates and surface and ground water contamination.

4. Environmental Controls

Waste rock from underground uranium mines and overburden from surface mines are not uniformly regulated. For instance, in New Mexico, where a great deal of uranium mining occurs, there is no state requirement for a reclamation effort pertinent to uranium mining, although there is an active and enforced coal mine reclamation program. In Colorado, Texas, Utah, Washington, and Wyoming there is a reclamation requirement for uranium mines. However, most inactive uranium mine sites do not show any evidence of an effective reclamation effort. Lack of uranium mine reclamation places ground and surface water quality, air quality (particulates and radon), safety, wildlife habitat, and land resources at higher risk. Mine water pumpage presents a much more complex picture. Although uranium mining, with the exception of *in situ* leaching operations, uses little water, it may be necessary to pump great quantities continuously from the mines during development and to permit mining operations. Water pumped to the surface and largely lost to evaporation depletes a natural resource, particularly valuable in the west, and is a potential source of contamination for shallow ground water supplies and surface streams used by livestock. It is sometimes assumed that the ground water pumped from mines is of the same chemical quality as natural ground water. This is not a valid assumption. The opening of mines creates oxidizing conditions that change the valence state of some elements in the ore body exposed to the air. This change in valence increases the solubility of some toxic and radioactive elements in

the mine water. One of the elements so affected is uranium, and in some cases the mine water contains economic values of dissolved uranium, as well as other elements of environmental concern. Another potentially complicating factor in mine water quality is the use of uranium mill tailings sands to backfill mined-out stopes. Although this appears to be a good solution to significantly reducing the volume of wastes disposed in mill tailings piles and also is effective in preventing subsidence of old underground mines, the effect of drainage from and through the backfill on the quality of water pumped from active portions of the mines has not been determined. There is a need, therefore, for control of this water contaminated, but not actually used, by the mining process.

Mine facilities differ in their control of mine water; however, control technology for pumpage from active mines usually involves use of lined or unlined settling ponds. Typically, the water is discharged into settling ponds, a flocculent is added to promote settling of suspended solids, and barium chloride is added to precipitate radium. In addition, if the water has elevated levels of uranium, it is treated by ion exchange to recover the uranium. In some cases water in the ponds is totally contained except for potential seepage, but may be discharged to arroyos or washes. These are sometimes tributaries to streams. The stream water may be used in drilling operations, recirculated in old stopes to solubilize more uranium, used as process water in mills (eventually disposed with mill tailings), used for irrigation, or used on dirt haulage roads for dust control (New Mexico Health and Environment Dept., July 1980).

In many cases, treatment with barium chloride and use of retention ponds has occurred only recently, and arroyos and washes remain contaminated from previous poor management of mine water. In some cases they continue to be contaminated. For instance, many uranium mines under development in New Mexico have a surface discharge to a watercourse (New Mexico Health and Environment Dept., July 1980). Even with barium chloride treatment of mine waters to remove radium, the chemical quality with regard to elements other than radium is questionable. There is evidence (New Mexico Health and Environment Dept., July 1980) that gross alpha activity may remain high even after treatment with barium chloride. Other toxic trace elements such as selenium, arsenic, and molybdenum are not expected to be removed by barium chloride.

National Pollutant Discharge Elimination System (NPDES) permits, under the Federal Water Pollution

---

*Memorandum from Caroline Wienke to File, Subject: "Corpus Christi Field Trip 6/19," July 3, 1979, H12-79-266.

Control Act as administered by the Environmental Protection Agency (EPA), are required for mining activities. However, there are some potentially environmentally significant administrative problems with this system. The EPA guidelines for issuing NPDES permits contain limitations for total suspended solids, chemical oxygen demand, zinc, ^{226}Ra (dissolved and total), uranium, and pH (Federal Register, July 11, 1978). Also of particular interest, but not directly regulated within the NPDES system, are gross alpha, selenium, arsenic, molybdenum, and vanadium. If a state does not have authority to issue NPDES permits, but has its own state regulations pertaining to surface water discharge or ground water contamination, compliance with these state requirements may become a condition for issuance of the NPDES permit by EPA. In some cases, however, the requirements are merely for reporting of discharge concentrations with no means of regulating. Thus in New Mexico for instance, molybdenum, vanadium, and selenium concentrations in pumpage are reported, but there are no limitations on the concentrations that can be discharged.

Of the six major uranium producing states, three (New Mexico, Texas, and Utah) are not approved to issue NPDES permits. Application must, in these cases, be made directly to the EPA. There is some indication that the lack of direct state involvement may result in less efficient enforcement of effluent limitations because of travel constraints and insufficient enforcement personnel. For instance, because of ongoing adjudication based on the mining companies' contention that discharge to a dry arroyo does not constitute discharge into "navigable" water, there are six mines or mine complexes in New Mexico alone for which NPDES permits have never been in effect.

With regard to potential ground water contamination from uranium mine pumpage, it should be noted that some uranium producing states (Utah, Washington, Wyoming) do not have regulations for protection of ground water. Although EPA's Safe Drinking Water Act provides for protection of potable supply, it is not clear how any one mine could be regulated if ground water contamination occurred in an area where there were many mines. In cases where mine water is allowed to percolate into soil or alluvium of dry washes etc., the sink of contamination is likely to be large enough to preclude any feasible effective remedial action should a drinking water standard be exceeded. This situation has a potential for creating long-term contamination of shallow ground water, a resource that is particularly valuable in the west.

In situ leaching operations also face problems associated with potential ground water contamination. Monitoring wells around the perimeter of the in situ leach well fields are operated to detect excursions from the mining zone. Withdrawal from the producing wells is also greater than injection volume to help prevent loss to aquifers. At the close of operations, ground water must be restored based on native water composition, potable water quality standards, water quality required for probable future uses, and treatability of waters. This is accomplished by pumping through the leached ore body to remove contaminants left by the leach solution or solubilized from the ore body. Based on relatively limited data, restoration of ground water may take "6 to 30 months and require the pumping of a minimum of 8 pore volumes" (Kasper and Engelmann, July 26, 1978). The flushing solution contains dissolved salts and trace elements and must itself be cleaned if it is to be re injected. This can be done with reverse osmosis, elec trodialysis, distillation, or ion exchange. However, it is not clear that these methods are in current use at in situ leaching operations.

5. Conclusions

Although waste rock from underground uranium mines and overburden from open pit mines are at present poorly regulated for inactive mines sites in particular, there is some expectation that under provisions of EPA's Resources Conservation and Recovery Act of 1976 (RCRA), an effective reclamation program will develop. At the time of this writing (October 1980), land reclamation pertinent to uranium mining was expected to be specifically addressed by EPA in the fall of 1980 (Federal Register, May 19, 1980).

Of greatest concern, in our opinion, is the potential impact on surface and ground waters by large amounts of contaminated water pumped from uranium mines. The situation is worthy of considerable attention now, and there is a particular urgency as mines are developed deeper and deeper thus producing more and more water. Although adequate controls (lined ponds, chemical treatment, reuse, mine grouting, backfilling) are technically feasible, lack of appropriate state regulations in some cases and administratively complex Federal-State regulatory interactions have permitted gaps that could be environmentally significant.

It should also be pointed out that EPA, as mandated by the Uranium Mill Tailings Radiation Control Act of 1978 (PL95-604) "shall provide a report to the Congress

which identifies the location and potential health, safety, and environmental hazards of uranium mine wastes together with recommendation, if any, for a program to eliminate these hazards." This report is at this time (October 1980) in draft form and is expected to be completed very early in 1981.

C. Uranium Milling

1. Introduction

Uranium milling is a series of processes that begin with mined uranium ore and through variable combinations of crushing, grinding, thickening, leaching, liquids-solids separation, ion exchange, elution, precipitation, filtration, and drying results in a semirefined uranium concentrate commonly called "yellowcake." As of January 1, 1976, there were 15 conventional operating uranium mills in the United States with a total nominal capacity of 25 800 metric tons of ore per day (GJO-100, 1976). As of January 1, 1980, there were 21 conventional operating uranium mills with a total nominal capacity of 44 500 metric tons of ore per day (GJO-100, 1980). This represents a growth in capacity of over 72% in four years. As of January 1, 1980, at least three more conventional mills were expected to begin operation in 1980 (GJO-100, 1980), and it is estimated that 83 such mills may be required by the year 2000 if there is no recycling of plutonium or uranium (DOE/EDP-0058, 1979).

2. Process

Conventional milling is accomplished primarily by using an acid leach solution to extract the uranium from the crushed ore, though a few mills use an alkaline leach solution.

Flow diagrams for both acid and alkaline leach mill circuits are presented in Figs. 2 and 3. Most uranium mills in the United States use a sulfuric acid leach solution because chemical costs are less and because the ore need not be ground as finely as for the alkaline process (carbonate leach). However, the alkaline process is more specific for uranium, solubilizes a lesser range of metals (Merrit, 1971), and can be used for ores that contain large amounts of limestone.

In the acid leach process, the crushed ore is typically ground to less than 0.1 mm (Clark, 1974). At this stage in both processes the ore is slurried and may be thickened and/or roasted to increase the process effi-

ciency. Leaching with acid or alkaline solutions is sometimes combined with use of oxidants to change the valence of the mineralized uranium and render it more soluble in the leach solution. In the alkaline leach process, pressurized leaching tanks are frequently used. After leaching, the solids ("barren") are separated from the liquids ("pregnant" with dissolved uranium). This process is sometimes preceded by a sands/slimes separation. Separation of sands and slimes is important, particularly if the sands are later to be used for mine backfilling. Solids are eventually separated from the pregnant liquid fraction using countercurrent decantation (acid process) or filtration (alkaline process). Strong base anionic exchange resins or, more commonly, solvents, and sometimes a combination of both, are then used to concentrate the uranium from the acid leach solutions. In strictly alkaline leach circuits, there is no concentration step; however, this solution may be clarified before precipitation. After the uranium has been concentrated, it is stripped from the solvent using, most typically, a tertiary amine, sometimes used in combination with an alcohol (decanol or isodecanol). In both acid and alkaline circuits the uranium in solution is then precipitated. Ammonia is most commonly used in acid circuits, giving ammonium diuranate. Sodium hydroxide is used in the alkaline circuits, giving sodium diuranate. The final product (sometimes purified further) is then dewatered, dried, and packaged in drums as yellowcake. Yellowcake composition is typically referred to as U_3O_8, although, strictly speaking, it is not. By-product streams may include recovery of molybdenum or vanadium.

3. Wastes Produced

The milling of uranium ore results in large volumes of solid, liquid, and gaseous wastes. It is estimated that, partially because ore grades will continue to decrease, the actual amount of ore processed will have to be doubled from 1977 to 1983 to meet the demand (Brown and Williamson, 1977). This of course results in accelerated production of milling wastes.

The average ore grade in the United States during 1979 was 0.11% U_3O_8 (GJO-100, 1980). Herein lies the reason for the large solid waste problem associated with uranium mills. There is essentially as much solid waste generated as ore processed. Thus, using milling extraction efficiency of 90.9% (GJO-100, 1980), for every "average" metric ton of ore processed during 1979, less than 1.0 kg of uranium metal is produced. Or, every metric ton of uranium metal produced as U_3O_8 from

ORE RECEIVING

CRUSHING ——→ DRYING —→

SAMPLING

GRINDING, CLASSIFICATION, AND THICKENING

Sulfuric Acid

Fresh Water
Makeup

LEACHING

LIQUID-SOLIDS SEPARATION
AND COUNTER CURRENT
WASHING

SAND-SLIME SEPARATION
AND COUNTERCURRENT
WASHING

Pregnant
Solution

Solids

Sands

Slimes

CLARIFICATION

RESIN-IN PULP
ION EXCHANGE

ION EXCHANGE OR
SOLVENT EXTRACTION

Barren
Slimes

High-Grade
Solution

Barren
Solution

High-Grade
Solution

TAILINGS DISPOSAL

Ammonia ——→

PRECIPITATION

FILTRATION

DRYING

Yellow Cake
Product

Fig. 2. General schematic diagram of an acid mill leach circuit (DOE, August 1978).

"average" ore in 1979 resulted in about 1180 metric tons of tailings solids. It is projected that by the year 2000 there will be a total of 620 million metric tons of uranium mill tailings in the United States (NUREG-0511, April 1979).

Mill tailings solids consist of the crushed fractions of the uranium ore after the uranium has been removed (chemically leached). The tailings can generally be considered to consist of two major fractions—slimes and sands. Slimes are <74 μm in diam and the sands are >74 μm (Swift et al., January 1976). Although tailings are roughly one-quarter slimes and three-quarters sands by weight, it is the slimes fraction that contains roughly three-quarters of the radioactivity (NUREG-0511, April 1979; Fry, September 1975; Borrowman and Brooks,

1975). These tailings are slurried with process water to impoundments on site. Water consumption for this purpose is about 550 m³ per metric ton of uranium metal produced (NUREG-0511, April 1979) in an acid process mill. The alkaline leach process uses 30 to 80% the amount of water used in the acid process (NUREG-0511, April 1979). This is because solids are separated by filtration in the alkaline process instead of washing. A mill tailings pile can be chemically characterized by analyzing the solids (sands and slimes) and the tailings water. Elements of concern vary somewhat with the ore processed but always include the radionuclides of the ^{238}U decay series. The principal branch of this decay series is depicted in Fig. 4 (DOE/EDP-0058, August 1979). Assuming secular

15
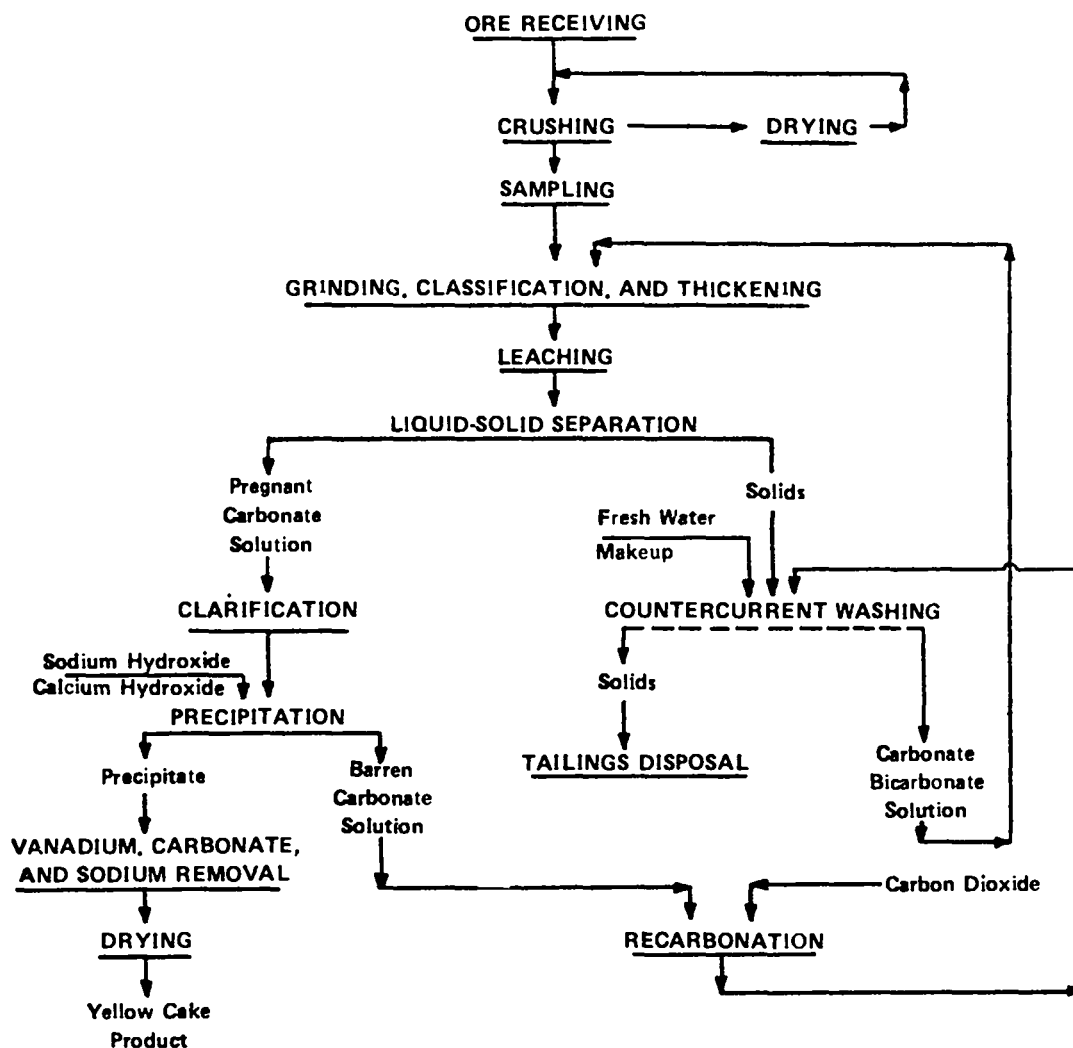
Fig. 3. General schematic diagram of an alkaline mill leach circuit (DOE, August 1979).

equilibrium in the ore and using the activities for the six β
-emitting daughters and the six α-emitting daughters that
remain after uranium extraction, we calculate that for
every metric ton of uranium metal produced as U_3O_8,
4.02 curies of α- and β-emitting daughters are in the
waste. Using 1979 U_3O_8 production figures (GJO-100,
1980), this results in about 58 000 curies in the uranium
mill wastes generated during 1979. It should be noted,
however, that a portion of the ^{222}Rn, an α-emitting gas,
will escape the mill tailings. Therefore, the above esti-
mate of 58 000 curies is high for solid tailings. However,
the number remains a good estimate of the total activity
released to the environment from uranium milling during
1979. One recent study (New Mexico Health and
Environment Dept., July 1980) reports chemical analy-
ses for active tailings pond liquor (Table II). There are no
available data to indicate that these particular ponds are
seeping.

Tailings from alkaline leach mills generally contain
contaminants in lesser concentrations, with the exception
of the anionic salts such as selenium, arsenic,
molybdenum, and vanadium (NUREG-0511, April
1979; Dreesen et al., 1978; Dreesen and Marple, 1979).
The tailings particles from an alkaline leach process also
tend to be more finely ground than those from an acid
leach process.

Of primary concern with regard to tailings from either
an acid or alkaline leach mill is the radon emanation
from the piles. The amount of radon exhalation from a
tailings pile varies considerably with the condition of the
tailings (wet vs dry, covered vs uncovered), meteo-
rological conditions, particle size, and homogeneity of
the tailings (sands and slimes). Radon is, however,
considered to be the radionuclide that is the single
greatest contributor to risk from uranium mill tailings
(NUREG-0511, April 1979). The flux of ^{222}Rn from

16

U = URANIUM
Th = THORIUM
Pa = PROTACTINIUM
Ra = RADIUM
Po = POLONIUM
Pb = LEAD
Bi = BISMUTH
α = ALPHA PARTICLE
β = BETA PARTICLE
γ = GAMMA RAYS
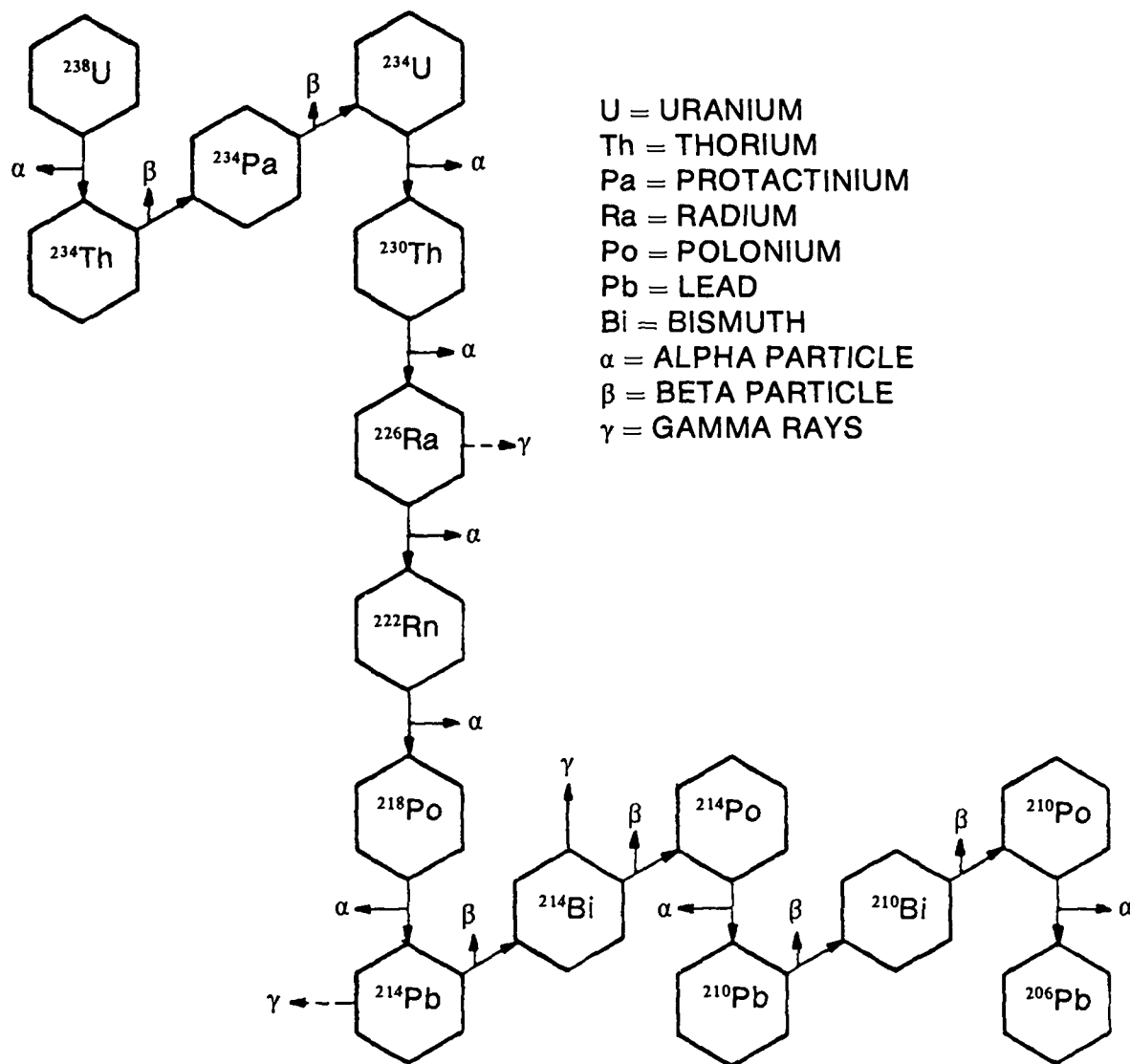
Fig. 4. Principal branch of ^{238}U decay series.

uranium mill tailings has received much attention recently and is the primary reason for much more stringent controls (cover material) on newly licensed milling operations. It is interesting to note, however, that the amount of radon exhaled from uranium mines is calculated (Bishop and Miraglia, October 1976) to be more than five times the amount from milling and that from a "worst case mill cluster situation" the radiological health risk is much smaller than from exposure to radon from mines in the region (NUREG-0511, April 1979).

Also of concern is the potential for seepage of mill process water or water from natural precipitation through the tailings materials and into surface or ground water. There is evidence of this having occurred at active sites. With regard to a mill in New Mexico, it was stated, "The upward trend in concentrations began before the disposal well was drilled, because of leakage from the tailings pond" (West, 1972). A shallow potable water supply downgradient from another active uranium mill in New Mexico was contaminated with selenium (EPA-906/9-75-002, 1975). More recently, contamination of ground waters with Mo and U by seepage from a Colorado tailings pond was directly implicated (Dreesen and Gladney, August 2, 1979).* The concentrations exceed suggested thresholds, irrigation standards, and/or ambient level goals based on health and ecological effects

*Letter from D. R. Dreesen, Los Alamos National Laboratory to Ray Cooperstein, December 18, 1979.

TABLE II

ACID MILL TAILINGS POND LIQUOR[a]

	No. of Samples	Range	$\bar{x}$	S
TSS mg/ℓ	6	20.5 - 435[b]	207 [b]	175
TDS mg/ℓ	7	17 850 - 46 104	36 013	9019
Cond μmhos	8	19 636 - 89 376	57 308	21 898
pH	6	0.87[b] - 2.15	1.27	0.47
As mg/ℓ	10	0.62 - 5.59[b]	2.21	1.48
Ba mg/ℓ	9	0.11 - 0.55	0.26	0.13
Se mg/ℓ	10	0.006 - 6.97	1.57	2.36
Mo mg/ℓ	10	0.16 - 21.82	3.02	6.64
NH_3 mg/ℓ	10	3.32 - 507	266	191
Na mg/ℓ	10	550 - 2118	1356	555
Cl mg/ℓ	10	297 - 3112	1335	1050
SO_4 mg/ℓ	10	304 - 57 824	24 465	17 564
Ca mg/ℓ	7	224 - 688	439	156
K mg/ℓ	8	82 - 182	109	33
Cd mg/ℓ	8	0.0094 - 0.097	0.04	0.04
Nitrate + nitrite mg/ℓ	8	<0.01 - 15.64	<6.6	5.8
Mg mg/ℓ	4	1205 - 2101	1590	426
V mg/ℓ	8	39 - 107	66	27
Zn mg/ℓ	8	<0.25 - 12.39[b]	<7 [b]	3.5
Al mg/ℓ	4	1110 - 1250	1175	70
Pb mg/ℓ	8	0.055 - 2.15	1.21	0.72
Gross α pCi/ℓ	8	2200 - 73 000	34 000	26 220
Ra-226 pCi/ℓ	9	15 - 1800[b]	257 [b]	580
Pb-210 pCi/ℓ	2	1200 - 1800	1500	424
U mg/ℓ	10	1.1 - 53	20	17

[a]New Mexico Health and Environment Department, July 1980.
[b]In excess of EPA effluent limitations guidelines for existing sources, uranium, radium, and vanadium ores subcategory (Federal Register, Vol. 43, No. 133, July 11, 1978).

(Dreesen, August 23, 1979).* It should also be noted that this contaminated ground water was used for irrigation and that in those areas, soil and vegetation analyses revealed elevated concentrations of these same contaminants (Dreesen, December 18, 1979).**

4. Environmental Controls

It is now generally recognized that control of uranium mill tailings is important. The Uranium Mill Tailings Radiation Control Act of 1978 requires that "every reasonable effort be made to provide for the stabilization, disposal, and control in a safe and environmentally sound manner of such tailings in order to prevent or minimize radon diffusion into the environment and to prevent or minimize other environmental hazards from such tailings." The DOE remedial action program generated by this act will address the 22 inactive sites and

*D. R. Dreesen and E. S. Gladney, "Report on Water Samples Collected in the Lincoln Park Area, Canon City, Colorado," transmitted by cover letter to Ray Cooperstein, New Facilities Section, Uranium Recovery Licensing Branch, Division of Waste Management, US NRC, Washington, DC from D. R. Dreesen, Los Alamos National Laboratory, August 2, 1979.
**D. R. Dreesen, "Final Report, Investigation of Environmental Contamination, Canon City, Colorado," transmitted by cover letter to Ross A. Scarano, New Facilities Section, Uranium Recovery Licensing Branch, Division of Waste Management, US NRC, Washington, DC from D. R. Dreesen, Los Alamos National Laboratory, August 23, 1979.

take appropriate action where necessary. Research and development programs are in place with regard to liners and caps for these tailings at inactive sites as well as potential conditioning technologies that could be used to make them less hazardous.

Environmental controls at active uranium mill sites vary considerably. New mill licenses now fall under the provisions of NRC's "Uranium Mill Licensing Requirements" (Federal Register, October 3, 1980). Criteria to be met by licensees include siting of tailings to isolate them from man and the environment, below-grade disposal or above-grade disposal with reasonably equivalent isolation, slope of tailings embankments, vegetative or rock cover on the tailings, avoidance of "capable fault" areas, seepage control, wind erosion control, and sufficient earth cover (not less than 3 m) on the tailings at the end of milling operations. This last criterion is to result in a "calculated reduction in surface exhalation of radon emanating from the tailings or wastes to less than two picocuries per square meter per second." It thus appears that environmental controls for new tailings sites will be sufficient to preclude health risks and significant environmental degradation. However, technologies developed and being developed to meet specific criteria, if they have been field tested, have not been observed over long periods.

Presently-operating uranium mills that were licensed before NRC's new "Requirements" became effective are the area of greatest concern. These operating mills do fall under provisions of the "Uranium Mill Tailings Radiation Control Act" and NRC's new "Requirements." However only certain requirements representing minimum levels of protection may be considered "practicable" at existing, active sites. Most of the tailings piles at these sites do not have specially constructed seepage barriers and some are seeping. Wind erosion of tailings particles from others is obvious and breaks in tailings pond dikes can lead (and have recently led) to environmentally significant contamination. On July 16, 1979 a tailings pond retaining dam breach released approximately 3.6×10^8 ℓ of acidified mill effluent and approximately 1100 tons of tailings slurry. This waste reached a watercourse. The mill was recently constructed and began operation in 1977.

5. Conclusions

Although there is an abundance of recent literature regarding the environmental aspects of uranium mill tailings, there is surprisingly little analytical data. Envi-ronmental reports and statements usually provide the most definitive data for a particular site. However, data often lack analyses for chemical species, which are important considerations for the proposed process. Often, the waste concentrations of radionuclide and toxic trace element species of interest are projections or are derived from "tests" and are not necessarily representative of actual operations. In some cases actual operating data are presented in a way (no net flow rates) such that they cannot be normalized and used in any meaningful way for comparison with different or similar processes.

Remedial actions taken at inactive sites as a result of the "Uranium Mill Tailings Radiation Control Act of 1978" and control measures taken to meet the new NRC "Uranium Mill Licensing Requirements" should be carefully studied to determine their long-term effectiveness.

Of greatest concern is the potential health risk and environmental degradation posed by existing uranium milling operations that were licensed before NRC's requirements were effective and for which remedial action may not be "practicable." This situation is not well documented and should be carefully assessed to quantify potential risks if remedial actions are not taken. If remedial actions are advisable, cost and feasibility should be determined.

D. Uranium Hexafluoride Conversion

1. Introduction

Two commercial plants in the United States convert the mills' output of moderately purified yellowcake to uranium hexafluoride for subsequent enrichment and fuel fabrication. These are the Kerr-McGee plant at Gore, Oklahoma and the Allied Chemical Plant at Metropolis, Illinois. In addition, several DOE facilities can or do convert U inputs (DOE/EDP-0058, 1979), generally not from "virgin" uranium mill products, but from reprocessed fuel, reclaimed enrichment scrap, etc. The gaseous diffusion plants (GDPs) at Paducah, Kentucky and Portsmouth, Ohio are equipped for conversion but do not operate. Former conversion operations at the Feed Materials Production Center at Fernald, Ohio, the refinery at Weldon Springs, Missouri, and the Oak Ridge GDP are currently inactive. The total throughput *capacity* of the two commercial conversion plants is about 24 000 MTU/yr. Domestic *input* is about 14 000 MTU/yr, and the remainder of throughput is for other nations.

2. The Conversion Process and its Waste Production

To appreciate the potential environmental concerns from conversion processes, it is useful to review the process technology and its associated waste production and environmental control technology. Several publications provide descriptions and flow charts. For generic but detailed and quantitative engineering descriptions that treat environmental and waste management concerns as specific aspects, there are two recent publications (ORNL/NUREG/TM-7, 1977; Schneider and Kabele, 1979). Partly because proprietary information is protected and unavailable, these publications use aggregated measurements or computed flows and activities, and refer to no specific existing plants; they are useful primarily for assessing future expansion of conversion operations. A companion volume to this report (Perkins, 1982a) is focused on environmental and waste management concerns, while also giving detailed flow charts and quantitative measurements. It is based on specific conversion facilities and reflects current practices and concerns.

a. Input Material. Input to the conversion process is nominally yellowcake or U_3O_8. Currently, it is diuranate salts of (principally) ammonium or of alkali/alkaline earth metals, and it averages 70% U metal by weight. Pure ammonium diuranate (ADU) would be 76% by weight (similarly for other uranates), thus about 10% consists of impurities. Table III quotes analyses for concentrate supplied to Kerr-McGee's facility. Alkali metals are removed for process engineering reasons (lot by lot when above a tolerance level). Heavy metal impurities of environmental concern are rejected in various process waste streams. The radioactivity is embodied in about 10 radionuclides when virgin ore (vs reprocessed fuel or scrap) is used. Table IV quantifies the seven major nuclides. Note that uranium itself as ^{238}U carries slightly less than one-fourth of the total activity. Because of the ^{238}U decay chain, three other nuclides are in secular equilibrium with ^{238}U and with each other, and thus have equal activity.

Later nuclides in the chain are very slow to equilibrate because half-lives increase markedly beyond ^{234}U; they grow in at negligible rates in purified uranium materials. The small activity of ^{230}Th left in semipurified concentrate is nonetheless of some concern in effluent (less so in waste) from conversion plants. It is a bone-seeking material of low maximum permissible concentration (Code of Federal Regulations, Title 10, Part 20). The

TABLE III

CONCENTRATE IMPURITIES
(1980 average of all lots samples)

All reported values in per cent on a uranium basis[a]

Nitric Acid Insoluble U^6	0.02
Molybdenum	0.08
Vanadium	0.05
Calcium	0.25
Thorium	0.06
Zirconium	<0.03
Boron	<0.01
Phosphorus	0.06
Halogens (Cl, Br, I as Cl)	0.13
Fluoride	0.01
Carbonate (CO_3)	0.07
Sulfur	1.00
Arsenic	<0.05
Sodium	0.72
Magnesium	0.09
Iron	0.53
Silicon	0.53

[a]Sequoyah Facility Lab results.
^{226}Ra is not routinely run.

Source: Kerr-McGee

TABLE IV

ACTIVITIES AND DECAY TYPES OF THE MAJOR RADIONUCLIDES IN ADU INPUT MATERIAL
(per metric ton of uranium content)

Radionuclide	Activity (Ci)	Decay Type
^{238}U	0.33	(α,γ)
^{234}U	0.33	(α,γ)
^{235}U	0.008	(α,γ)
^{234}Th	0.33	(β,γ)
^{234}Pa	0.33	(β,γ)
^{230}Th	0.002	(α,γ)
^{226}Ra	0.002	(α,γ)
^{222}Rn (gas)	0.002	(α)

conversion process rejects much of the ^{234}Th and ^{234}Pa, but these decay rapidly to (some mass but negligible extra activity of) ^{234}U. In wastes the dominant long-term radionuclides are U isotopes, ^{230}Th, ^{226}Ra, and ^{222}Rn. Dominant nonradiological hazards derived from U concentrate (vs process chemicals) are heavy metals such as Mo and V.

b. The Dry Hydrofluorination Process and its Wastes. Two very different processes are used to make uranium hexafluoride (UF_6) from ADU. The dry or hydrofluorination process at the Allied Chemical Plant consists of five basic steps plus pretreatment (for some lots) and a recycling stream. We briefly describe the steps and note intermediate effluents and final wastes.

- Pretreatment: Concentrate lots containing significant sodium have the latter removed chemically. Liquid wastes containing alkali metal sulfates and small amounts of U, ^{226}Ra, ^{230}Th, and nonradiological toxic heavy metals are discharged to settling ponds. Pond decantate is discharged and sludge is sent to the pond sludge calciner.

- Step 1: Calcination. Sodium-free ADU is heated under partly oxidizing conditions to form U_3O_8 and some uranium trioxide, UO_3. This material is blended and sized. High temperatures and rapid reactions cause some fine particulates to be suspended in the gas flow. These particulates are composed of ADU, U_3O_8, and unresolved compounds of other radionuclides and nonradioactive impurities. Primary dust collecting bag houses capture most particulates for later recycling. A very small fraction of particulates escapes into the well-ventilated process space, to be trapped almost quantitatively by high-efficiency particulate air (HEPA) filters. HEPA filters ultimately become wastes to be buried. Worn out primary bag house filters are recycled for U content in new sections of conversion plants. In older sections, recycling facilities are not available.

- Step 2: Reduction. U_3O_8/UO_3 from calcination is reduced in a fluidized bed with molecular hydrogen (cracked from ammonia) to form UO_2, uranium dioxide. Particulates are formed as in Step 1 and lead to similar recycling flows and HEPA filter wastes.

- Step 3: Hydrofluorination. UO_2 reacts with vaporized hydrogen fluoride gas, HF, to form uranium tetrafluoride, UF_4, a stable solid. Particulates in the off-gases are again formed (UO_2, UF_4, other radio-

nuclides, other impurities) and again lead to recycling flows and HEPA filter wastes. HF is scrubbed from off-gases to form $CaF_2/CaCO_3$ sludge, stored on site (very low activity).

- Step 4: Fluorination. The UF_4 is reacted with fluorine gas to form UF_6 in the presence of CaF_2 bed material. Again, particulates are formed and trapped. Fluorination ash is also formed, containing the bed CaF_2 plus nonvolatile impurities (trace metal and non-U radionuclide fluorides, for example). The ash is leached with sodium carbonate, Na_2CO_3, to recover most of the U content for recycling. Leached ash, containing traces of radionuclides, is shipped off site for burial.

- Step 5: Distillation. UF_6 is distilled for purification in a nominally captive stream. Purified UF_6 is condensed in cold traps, which are later heated to transfer UF_6 to product cylinders. Very small UF_6 releases are scrubbed out in water scrubbers as hydrolyzed UO_2F_2 particulate. Some of the U is recovered for recycling; minor amounts of CaF_2 sludges are formed. "Still bottoms" containing traces of U and the other radionuclides are stored on site.

- Uranium Recycling Flow and Chemical Production Flows. Fluorinator ash leachate, UF_6 cylinder waste, and scrubber materials containing U are redissolved, ultimately to form ADU. Liquid waste streams are formed containing some U particulates (very little dissolved U), most of the uranium daughter nuclides, and some fluorine compounds. The U-recovery wastes are sent off site to a LLW burial facility. CaF_2 sludges formed in chemical treatment of liquid waste streams to trap fluorine are ponded on site, and these contain traces of radionuclides and other feed impurities.

All steps produce contaminated failed equipment and wipes from routine cleaning.

The net waste production is not reported in full, but partial estimates exist or can be derived and these cover the bulk of the radioactivity. "True" low-level wastes shipped off site for LLW burial have been noted above. Annually, these amount to about 1500 metric tons with a maximal content of 47 Ci, from a production of 14 000 metric tons of uranium (MTU). There are also ponded CaF_2-based sludges, which will be considered as wastes upon decommissioning of the facility. These sludges have a reported volume of 0.045 m^3 (weighing 100 kg) per MTU of input. At present there are about 30 000 MT of reacted CaF_2 and 76 000 MT of total sludge (wet, with

unreacted $CaCO_3$). Estimated radioactivities of the total wastes, per MTU input, are 5×10^{-5} Ci each of ^{238}U, ^{234}U, and ^{226}Ra, 6×10^{-5} Ci each of ^{234}Th and ^{234}Pa (rapidly decaying), and 4×10^{-4} Ci of ^{230}Th (Allied Chemical Corporation, 1970). The ^{230}Th is enriched over U in the wastes, and represents about the maximum activity of ^{230}Th if it all appeared in waste while only 0.1% of total U appeared. Approximately four hundred 55-gallon drums of contaminated trash are sent annually to a commercial LLW burial site.

c. The Wet Process and its Wastes. The wet conversion process at the Kerr-McGee plant consists of seven basic steps.

- Step 1: Feed preparation and digestion. The ADU input to Kerr-McGee has more alkaline earth diuranate than ammonium diuranate; it is only nominally ADU. It is reacted with nitric acid to make the hexavalent nitrate salt, uranyl nitrate, $UO_2(NO_3)_2$. Very small amounts of particulates are liberated ultimately to the process space and most are trapped on final HEPA filters, which in turn become solid wastes. Neutralization of liquid wastes to ammonium nitrate solids produces some ponded sludges with coprecipitated radionuclides and other feed impurities.

- Step 2: Solvent extraction. Tributyl phosphate in a hexane solvent is used for countercurrent chemical extraction of uranium (as UO_2^{2+} cation), leaving the feed impurities, including most of the non-U daughter radionuclides. The uranyl nitrate is then re-extracted into an aqueous phase. The impurities in solution are combined with other liquid wastes and are largely precipitated by addition of ammonia. The resulting slurry is ponded on site (raffinate); soluble remnants are released in liquid effluent.

- Step 3: Concentration. Two stages of water evaporation with some cleanup of organic matter bring the $UO_2(NO_3)_2$ to a high concentration. Minor droplet emissions occur. Scrubbing contributes a very small final amount of liquid waste, which is treated with other such wastes.

- Step 4: Denitrating calcination. This operation generally resembles calcination in the dry hydrofluorination process, except that the input is a concentrated aqueous solution. The product is mostly trioxide UO_3 rather than U_3O_8. Off-gases contain substantial amounts of nitrogen oxides, which are recovered for nitric acid.

- Steps 5-7: Reduction, hydrofluorination, fluorination. These steps are quite similar to Steps 2, 3, and 4 in the dry process, both in technical design (but fluorination is done without a CaF_2 bed) and in contributions to waste production.

Note that final distillation of UF_6 product for purification is not used because solvent extraction has earlier removed sufficient impurities. One significant difference between this process and the dry process is that daughter radionuclides are found mostly in ponded sludges of nonnegligibly soluble inorganic salts from the treatment of intermediate liquid waste streams. Some sludges, because of ammonium nitrate content, have an additional problematic trait of being hygroscopic and thus of not drying completely. The waste yield per MTU input is about 60 kg of CaF_2 sludge plus 1000 ℓ of raffinate sludge containing mostly solids, plus lesser NH_4NO_3 sludges and cleanup wastes. Some CaF_2 sludge has been buried on site; final disposition of wastes has not been decided to date. Radioactivity in wastes per MTU is that from about 0.015% of the U plus virtually all the ^{230}Th (4×10^{-4} Ci) and a similar amount of ^{226}Ra. Nonradioactive impurities in feed material appear, primarily, in the raffinate sludges.

3. Environmental Controls on Wastes

There are only two basic types of wastes from conversion, dry solids and ponded sludges, and both are of moderately low activities and modest volumes. Fluorination ash from the Allied Chemical Plant is shipped off site after packaging to DOT specifications. The sludges are ponded on site and no final disposition has been decided upon.

Shallow land burial (SLB) of the dry wastes is best characterized for the commercial (vs on-site) burial grounds. We discuss the environmental controls of SLB in Sec. V. The potential for migration of the conversion-waste radionuclides into the biosphere is inherently rather low because their chemical forms are nonvolatile and only a portion is soluble [about one-third of the trapped particulates (Allied Chemical Corporation, 1970), for example]. At least in the case of off-site burial, there is generally no segregation of conversion wastes from other types of waste that might contain mobilizing agents such as chelating compounds. This is one potential concern.

The ponded sludges are subject to few environmental controls beyond the siting of ponds inside the fence. The

high degree of insolubility of sludges (for CaF_2; less for ammonium-salt sludges from the wet process) assures that undisturbed ponds will not give rise to leachates with high concentrations of radionuclides. Leaching of sludges can also give rise to a nonradiological hazard, that posed by trace metals and fluorides. Even calcium fluoride has a reasonable rate of dissolution if water flows are large. Existing EPA regulations must be applied in the future to control these chemical hazards at decommissioned plants.

4. Conclusions

With completely defined SLB practices both on site and off site, the principal concern arising from conversion wastes is probably the ^{230}Th and ^{226}Ra content in sludge, yielding an activity of about 500 pCi per gram, hence posing a modest hazard of radon-daughter release. [The airborne *effluents'* content of ^{230}Th and of vegetation-damaging HF is of more concern, but this is not a *waste* problem (Perkins, 1982a)]. The uncertainty about deposition of the sludges adds the modest concern that radionuclides might be mobilized by chelators or other agents after control of burial sites is lost (at or away from former conversion plants). It is not possible to state whether the solid-waste disposal will meet the proposed regulations for SLB because the regulations are not final and the technical controls over radionuclide mobilization are incomplete.

E. Uranium Enrichment

1. Introduction

The concentration of ^{235}U in natural uranium is about 0.7%. Because light-water moderated nuclear reactors require fuel with 2 to 4% ^{235}U (higher for many military reactors), uranium enrichment is a necessary process in reactor fuel production (WASH-1248, 1974).

Current technology for uranium enrichment is based on the gaseous diffusion principle. At a given temperature ^{235}UF$_6$ will diffuse through a porous barrier at a slightly higher rate than ^{238}UF$_6$. By allowing UF$_6$ to diffuse through a series of barriers many times, a product highly enriched in ^{235}U can be obtained.

There are three gaseous diffusion plants operating in the United States, which are owned by the U. S. Government and operated by contractors. The enrichment plants at Oak Ridge, Tennessee, and Paducah, Kentucky, are operated by Union Carbide Corporation and the Portsmouth, Ohio plant is operated by Goodyear Atomic Corporation. These facilities, often operating in series, produce UF$_6$ enriched to various degrees for both commercial and government reactors. Because parts of the gaseous diffusion process are classified and because DOE facilities are not licensed by NRC, there is very little specific information about quantities and characteristics of radioactive wastes produced by enrichment plants.

2. Process

a. Gaseous Diffusion Process. The average velocity of gas molecules at a given temperature is proportional to their masses. The gaseous diffusion process is based on the principle that lighter gas molecules will diffuse through a porous barrier at a slightly higher rate than heavier ones. Using volatile uranium hexafluoride as the gas, the maximum theoretical increase of ^{235}U over ^{238}U is a factor of 1.0043 for a single stage separation (ORO-684, 1972). This small separative capability is multiplied by using many stages in series, called a cascade, producing UF$_6$ in which ^{235}U is greatly enriched (WASH-1248, 1974).

Uranium hexafluoride is received at the enrichment plant's toll enrichment facility in steel storage cylinders, where it is weighed, sampled, assayed, and its parity determined. The UF$_6$ cylinders are then transferred to the feed vaporization facility. Uranium hexafluoride is a solid at room temperature, sublimating to a gas at 56.4°C (ERDA-1549, 1976). The UF$_6$ is vaporized in a steam-heated vaporizer or autoclave and fed to the diffusion cascade.

Because UF$_6$ is extremely reactive with water, corrosive to most metals, and incompatible with organic materials (such as lubricating oil) the cascade and other process equipment is constructed primarily of nickel-plated steel, monel, and aluminum (ERDA-1549, 1976). In the cascade, the UF$_6$ gas is passed through porous barrier tubes enclosed in a "converter." About one-half the gas diffuses through the barrier into the converter and is fed to the next higher stage (ERDA-1549, 1976). The remaining, undiffused gas is recycled to the next lower stage. Axial-flow compressors are used to move the process gas through the converters. Stage coolers are required to remove the heat of compression and to keep process stream temperatures at an optimum level. Stages are grouped together to form "cells," which can be taken off line and bypassed to facilitate maintenance. Contaminants in the process stream are gases whose molecular

weights are less than that of UF_6. These "lights," primarily N_2 and O_2, with traces of HF, coolant, and reaction products from the breakdown of coolant and UF_6, move to the top of the cascade where they concentrate in the purge cascade. The lights are then filtered and vented to the atmosphere.

The enriched UF_6 (about one-fourth of the uranium input) is withdrawn from the system into desublimation cells. The product is deposited in steel cylinders that have been precooled by refrigeration equipment. Fugitive UF_6 is collected in liquid nitrogen traps. Chemical traps containing activated alumina are used as a backup system to collect UF_6 and HF (ERDA-1543, 1976; Schneider and Kabele, 1979).

In addition to enriched UF_6, the gaseous diffusion process produces a stream of process gas depleted in the fissionable isotope called the "tails." The depleted tails are withdrawn from the bottom stage of the production cascade in a series of compression-cooling cycles that liquify the gas, which drains into storage cylinders. The tails cylinders are cooled and stored on site for future use. About three-fourths of the plant's uranium input is withdrawn as depleted tails.

b. Stage Coolant System. Each stage in the diffusion cascade is equipped with a stage cooler to remove the heat of compression and to maintain temperatures in the process stream at an optimum level. This system uses an evaporative coolant, $CClF_2$-$CClF_2$ (R-114) (ERDA-1549, 1976; Schneider and Kabele, 1979). The R-114 coolant is maintained at a pressure greater than that of the process stream so that system leaks will result in leakage of R-114 into the process stream rather than the reverse. The R-114 is chemically compatible with UF_6. Any R-114 in the process stream migrates to the purge cascade and is vented to the atmosphere.

The R-114 enters the stage cooler as a liquid and boils as it removes heat from the process stream. The coolant is then circulated through a series of water-cooled condensers, where it cools and liquifies, then returns to the stage cooler. Cooling water for the condensers is furnished by the plant's Recirculating Cooling Water (RCW) system. The hot water from the condensers is routed to cooling towers, then recirculated back through the system. Because of the corrosive acids that are formed by mixing R-114 and water, the R-114 pressure is kept greater than the water pressure. It is considered preferable, in case of leakage, to allow R-114 into the RCW system than to have corrosive acids in the converter (ERDA-1549, 1976).

c. Recirculating Cooling Water System. The primary function of the Recirculating Cooling Water (RCW) system is to condense the R-114 evaporative coolant in the diffusion cascade stage coolers. Secondary uses are for cooling process booster pump stations, evacuation pump station cooling, and lubricating oil coolers. A series of cooling towers is provided to reduce on-tower water temperatures from about 54°C to about 32°C for recirculation back through the system (ERDA-1549, 1976).

Major water losses from the cooling towers result from evaporation and blowdown. Evaporation of cooling water results in the buildup of dissolved solids (calcium, sulfates, etc.) in the system. To reduce this problem, a quantity of water, called blowdown, is released to a holding pond. The blowdown is chemically treated, then released to the environment via local streams. In addition, corrosion inhibitors are added to the RCW system, contributing chromium, phosphates, zinc, and chlorine to the blowdown (Schneider and Kabele, 1979). Make-up water for the RCW system is supplied by a nearby river.

d. Uranium Recovery. The uranium recovery process reclaims uranium from decontamination operations, laboratory wastes, alumina traps, and other process operations for recycle. Combustible solid materials are incinerated, calcined, dissolved, flocculated, and filtered. The resulting solution is then processed by solvent extraction with tributyl phosphate and *Varsol*, producing uranyl nitrate hexahydrate (UNH). The UNH is calcined to produce U_3O_8 and converted to UF_6. The UF_6 gas is filtered, passed through magnesium fluoride traps to absorb volatile impurities, and cold trapped into storage cylinders for eventual refeed to the diffusion cascade.

3. Wastes Produced

The primary wastes produced by the gaseous diffusion process are uranium and uranium daughters (Schneider and Kabele, 1979) and nonradioactive effluents (ERDA-1549, 1976). Some of the UF_6 received by enrichment plants may come from reprocessing of government-owned reactor fuel. These "reactor returns" introduce small amounts of ^{237}Np, ^{239}Pu, ^{230}Th, and ^{99}Tc into the system (Schneider and Kabele, 1979; ERDA-1543, 1976). UF_6 from conversion facilities may also contain such impurities as vanadium and molybdenum fluorides.

24

Gaseous wastes are produced by vaporization of UF_6 in the toll enrichment and feed vaporization facilities, in the diffusion cascade, the tails withdrawal facility, and from decontamination and uranium recovery operations. Gaseous effluents released to the atmosphere include HF, various fluorides, ammonia, nitrogen oxides, hydrocarbons, acetone, R-114 coolant, sulfur oxides, ozone, trichlorethylene, xylene, and particulates (Schneider and Kabele, 1979).

Liquid wastes are from decontamination and uranium recovery operations, laundry detergents, sewage treatment, and cooling tower blowdown. These wastes are collected in unlined holding ponds to allow the particulates to precipitate before release to nearby bodies of water (ERDA-1543, 1976; ERDA-1549, 1976; Schneider and Kabele, 1979).

Solid wastes are uranium-contaminated incinerator ash, process equipment, process sludge, clothing, filter trap materials, scrap metal, and holding pond sludge. These wastes are generally buried on site.

There is a remarkable lack of reliable data regarding volumes and activities of radioactive wastes produced by uranium enrichment plants. The EIS for Expansion of U.S. Uranium Enrichment Capacity (ERDA-1543, 1976) had to forego a health and safety assessment for toxic and radioactive materials because of insufficient data. A recent report describing reference nuclear fuel cycle facilities (Schneider and Kabele, 1979) states that a significant amount of data is still needed regarding the characterization of wastes and effluents, and environmental controls. The only information we have found regarding the quantities and activities of wastes buried at enrichment plants is shown in Table V. Note that the accumulated waste quantity at the Portsmouth Plant at the end of FY 1977 is reported to have been 2655 kg of uranium. The Portsmouth Site EIS reported 4500 kg of uranium buried on site (ERDA-1555, 1977). A factor of 2 difference may be within expected error, but it does exemplify the lack of consistent information about wastes at enrichment plants.

We have found little mention in the literature of ingrowths of uranium daughters in enrichment plant wastes. These radionuclides would be produced fairly slowly by uranium decay and could cause problems in the future.

4. Environmental Controls

Environmental control of wastes at enrichment plants includes filtering of gaseous wastes, chemical treatment of liquid wastes, and burial of solid wastes. Gaseous wastes containing both radioactive and nonradioactive contaminants are derived from purge systems and decontamination and uranium recovery operations. These gases are water-scrubbed and passed through HEPA filters before release to the atmosphere (ERDA-1543, 1976; Schneider and Kabele, 1979). Because gaseous waste streams are so poorly characterized, it is not possible to assess the adequacy of the treatment processes.

Liquid wastes result primarily from decontamination and uranium recovery operations and cooling tower blowdown. All liquid wastes are discharged to holding ponds. Liquid effluents from plant laundries pass through a sewage treatment facility before discharge to the holding pond. Cooling tower blowdown is pH-adjusted and treated with sulfurous acid to reduce hexavalent chromium to the trivalent state. The chromate is then precipitated with slaked lime and the remaining water is released to local streams

TABLE V

ESTIMATED QUANTITIES AND ACTIVITIES OF WASTES AT
GASEOUS DIFFUSION PLANTS[a,b]

Facility	Waste Type	Annual Rate		Accumulated	
		Mass (kg)	Activity (kCi)	Mass (kg)	Activity (kCi)
Oak Ridge	Uranium			48 323	<<1
Paducah	Uranium	150 000	0.06	2 759 700	1.05
Portsmouth	Uranium	170	---	2655	<<1

[a]Data from Duguid (1977).
[b]At the end of FY 1977.

(ERDA-1549, 1976). The holding ponds are periodically dredged to control sludge buildup. Sludges are generally packaged in steel drums and buried on site (ERDA1549, 1976). The holding ponds are unlined, and there seems to be little published data regarding the specific compositions and activities of these sludges, though they are known to contain ^{237}Np, ^{239}Pu, ^{230}Th, and ^{99}Tc in addition to uranium (Schneider and Kabele, 1979).

Contaminated oil, cleaning solutions, solvents, etc. are disked into open fields. Uranium-contaminated sludge from the sewage treatment plants is disposed of in the same manner (ERDA-1549, 1976).

Solid wastes are primarily filter and trap sludges, contaminated equipment and machine parts, and holding pond sludges. Combustible materials are incinerated and the ash is buried on site (ERDA-1543, 1976; Schneider and Kabele, 1979). Recoverable uranium is removed from contaminated equipment and treated for re-feed into the diffusion cascade (ERDA-1549, 1976). Solid wastes with uranium contamination too low to allow economic recovery or with fixed surface contamination are buried or stored above ground. Wastes containing >10 nCi/g of transuranics are stored retrievably (Duguid, 1977). Production cascade tails in steel cylinders are stored in above-ground structures or on storage pads.

All DOE facilities monitor themselves on a regular basis. We surveyed the environmental monitoring reports for uranium enrichment plants. No apparent migration of radionuclides has been reported in ground or surface water, or in soils.

5. Conclusions

Radioactive wastes produced at uranium enrichment facilities are gaseous, liquid, and solid. Gaseous wastes are generated by vaporization and diffusion of UF$_6$ and by decontamination and uranium recovery operations. Gases are water scrubbed and filtered before being released to the atmosphere.

Liquid wastes result primarily from decontamination and uranium recovery operations and from cooling tower blowdown. All liquid wastes are chemically treated and collected in unlined holding ponds to allow contaminants to settle before release to local surface waters.

Solid wastes are contaminated equipment, filter trap materials, incinerator ash, process sludges, and sludges dredged from liquid waste holding ponds. Solids are buried or stored above ground on site.

Contaminated oil, solvents, and cleaning solutions are disked into open fields. Contaminated sludge from sewage treatment operations is disposed of in the same manner.

We have been unable to find sufficient data regarding waste volumes and activities at gaseous diffusion plants. Wastes and effluents generated by the uranium enrichment process are poorly characterized. For these reasons we feel that there are insufficient data to assess the adequacy of environmental controls at this time. Because the volumes of waste involved are relatively small and no migration of radionuclides from the on-site burial grounds has been reported, we feel that other areas discussed in this report (especially shallow land burial) are of more immediate concern. We do, however, feel that data collection on gasous diffusion plant wastes should continue.

F. Fuel Fabrication

1. Introduction

Fabrication entails the chemical processing of uranium hexafluoride (UF$_6$) into uranium dioxide (UO$_2$) powder and then making final fuel assemblies from the raw UO$_2$ powder. Five fabrication plants for *commercial* fuel (processing low-enriched UF$_6$, that is ^{235}U content up to approximately 5% of total U) are operating in the United States today. Three of them are single-site facilities that process UF$_6$ to final fuel assemblies "under one roof." They are Exxon at Richland, Washington; General Electric at Wilmington, North Carolina; and Westinghouse at Columbia, South Carolina. The other two fabrication plants have two distinct sites each, one for UF$_6 \rightarrow$UO$_2$ and the other for UO$_2 \rightarrow$fuel. Babcock and Wilcox perform the operations at Apollo, Pennsylvania and Lynchburg, Virginia, respectively. Combustion Engineering has its facilities at Hematite, Missouri and Windsor, Connecticut, respectively. (DOE facilities at Hanford, Washington and Savannah River, South Carolina also fabricate fuel assemblies in smaller, research quantities. Highly enriched naval reactor fuel is a special case outside this discussion.) The total throughput of the five commercial fabrication plants in 1978 was about 2650 metric tons of U metal, of which about 40% was exported to other nations. This tonnage is one-fourth that of conversion plants, because about three-fourths of the U is rejected during enrichment as "tails" depleted in ^{235}U.

Fabrication produces somewhat larger quantities and total activity levels of radioactive wastes than does conversion, but is of less concern because most daughter radionuclides (especially ^{230}Th) have been removed in conversion. The input material is virtually pure U compounds of fairly low activity and a few daughter radionuclides; there are no fission products in the (commercial) feed material. The fractional loss of U during fabrication is quite small, on the order of 0.1% or less with more modern waste treatment to feed U back into the recycling stream. The predominant wastes are simple solids—trapped particulates, trace-contaminated calcium fluoride sludges, contaminated failed equipment. These solid wastes, of modest activity, are disposed on and off site. As with conversion, fabrication is discussed for completeness and for shared *generic* concern about shallow-land burial; immediate hazards, both radiological and nonradiological, are of secondary rank compared to the front and back of the nuclear fuel cycle.

2. The Fabrication Process and its Waste Production

A review of process technology will lead one to appreciate the nature of the wastes produced and the environmental, health, and safety concern that the wastes engender. Generic engineering descriptions that are detailed and quantitative and that treat environmental and waste management concerns in due course are given in three recent publications (ORNL-TM-4902, 1975; Schneider and Kabele, 1979; Guilbeault and Reckman, 1979). These publications report mostly aggregated measurements or computed flows, partly because proprietary information on actual plants is protected and unavailable, thus these reports are useful primarily for assessing future expansion of fabrication capacity. A companion volume to this one (Perkins, 1982b) is based upon specific existing fabrication facilities and reflects current practices; it is based upon environmental and waste management concerns.

a. Input Material. The input material for fabrication is purified and enriched uranium hexafluoride, UF_6, delivered as a vapor-pressurized solid in tanks. On the basis of *metallic* content, this material is 96.8% ^{238}U by weight, 3.2% ^{235}U (varies a bit, depending on the reactors for which the product fuel is destined), 0.02% ^{234}U, and active traces of ^{234}Th and ^{234}Pa. The latter two nuclides have the same *activity* as their ^{238}U parent, because of the secular equilibrium that establishes itself in a few months:

$$^{238}U \xrightarrow{\alpha,\gamma} {}^{234}Th \xrightarrow{\beta,\gamma} {}^{234}Pa.$$
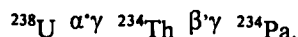
The radioactivity per metric ton of U metal is given in Table VI. Note that ^{234}U, a minor contributor to fuel value, has the highest activity of all, being 4.5-fold enriched over natural abundance as is ^{235}U.

b. The Wet ADU Process and its Wastes.

- Step 1: Delivery of UF_6. The tanks of UF_6 are heated to continuously deliver UF_6 vapor. Minor leakage will release UF_6 vapor that immediately hydrolyzes to uranyl fluoride, UO_2F_2, in moist air or in the wet scrubbers that protect the air quality of the process space and work space. The fraction of this small fraction that is re-entrained in the final air exhaust of the plant is caught by HEPA filters. These filters ultimately become low-level waste to be buried. The scrubber liquids with other minor UO_2F_2 content are filtered; wet filter beds form another ultimate low-level waste of very small activity.

- Step 2: Hydrolysis. UF_6 vapor enters a water-filled reactor, where it hydrolyzes quantitatively to UO_2F_2 in solution.

$$UF_6 + 2H_2O \rightarrow UO_2F_2 + 4HF \uparrow$$
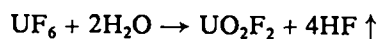
Very small quantities of airborne particulates are formed here, and contribute to ultimate wet filter wastes and HEPA filter wastes. Most HF goes off as a gas and is scrubbed out with calcium hydroxide.

TABLE VI

ACTIVITIES AND DECAY TYPES OF THE MAJOR RADIONUCLIDES IN THE UF_6 INPUT MATERIAL

(per metric ton of uranium content)

Radionuclide	Activity (Ci)	Decay Type
^{238}U	0.32	α,γ
^{235}U	0.068	α,γ
^{234}U	1.62	α,γ
^{234}Th	0.32	β,γ
^{234}Pa	0.32	β,γ

- Step 3: Precipitation. Ammonia solution is added to precipitate the uranium as ammonium diuranate (ADU). Excess NH_3 neutralizes the acidity and drives the precipitation.

$$6NH_3 + 2UO_2F_2 + 3H_2O \rightarrow (NH_4)_2U_2O_7 + 4NH_4^+F^-$$
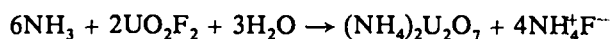
Again, minute leakages contribute to U-containing particulates that now contain the ADU species as well as UO_2F_2.

- Step 4: Dewatering. The dispersed ADU precipitate is solidified and dewatered by centrifugation (General Electric, Wilmington) or ultrafiltration (Babcock and Wilcox, Apollo). ADU solids pass on to Step 5, while liquid "wastes" (ultimate wastes and recoverable material) are treated to extents that vary by plant. Sequences of settling, centrifugation, filtration, ion exchange, and final precipitation are (or will be) employed to various degrees. The calcium hydroxide used to neutralize scrubbed HF gas and to precipitate F^- in NH_4F forms the poorly soluble calcium fluoride sludge, CaF_2. This sludge contains substantial traces of coprecipitated uranium (mostly as ADU), about 200 to 1000 ppm U currently, with future plant capacity designed to much lower contents. The ^{234}Th and ^{234}Pa daughters in initial material are ultimately irrelevant; new daughter activities appropriate to parent-U content grow into wastes and process-stream U-compounds alike. In addition to the dominant CaF_2 waste, there are also some HEPA and wet filter wastes from capture of airborne particulates.

- Step 5: Calcination. ADU is heated in a reducing atmosphere containing H_2 or NH_3, forming UO_2 quantitatively. Particulates and associated modest filter wastes are formed, as in other steps.

- Mechanical assembly. The calcined UO_2 is pulverized, blended for size control, made into slugs, and pressed into pellets. These are dirty operations generating HEPA filter wastes. The pellets are sintered in a mild reducing atmosphere, ground to size (a strong source of waterborne U particulates, largely recovered for recycling), dried, and loaded into fuel-rod claddings, which are then welded closed and put into large arrays or fuel-rod assemblies to be shipped to users. Major wastes from these mechanical steps are the particulates (some airborne, most waterborne, some caught in cleaning wipes), floor drainage, etc. Some UO_2 particulates are caught in machine oil at

the pelletizer. This oil is sealed in barrels; some is stored on site, some is shipped off site as waste.

- Recycling streams. Clean UO_2 (as from grinding operations) is recycled in significant quatities (15% of throughput) by dissolution/neutralization/precipitation/calcination in a separate flow stream. The wastes generated are as in the mainline Steps 2 through 5. "Scrap" U-containing material (mixed with substantial amounts of unrelated material) is recycled at a small rate (ca 2% of throughput) by dissolution and solvent extraction for purification, after which it enters the "clean" recycle flow. Additional types of waste are generated, such as filters that catch the scrap's largely non-U contaminants, and a liquid waste stream.

All the steps produce contaminated failed equipment and routine cleaning wipes.

Process wastes are partly treated; combustible U-contaminated wastes (pre-filters, parts of HEPA filters, wipes) are incinerated at some facilities but not at others, with the resulting ash partly diverted to U-recycling, if U-content is high enough.

The net radioactive waste production, as reported, varies widely by facility. The dry wastes (excluding CaF_2 sludges) that are packaged and shipped for off-site burial amount to 0.6 to 7.6 m^3/MTU input. The average is 2.0 m^3/MTU. The U-content varies from 0.5 to 9.2 kg/MTU input. The activity varies likewise from (1.6 to 30) $\times 10^{-4}$ Ci of ^{238}U/MTU, from (0.8 to 15) $\times 10^{-3}$ Ci of total U/MTU, and probably (1.5 to 25) $\times 10^{-4}$ Ci of ^{234}Pa and lesser (decayed) activity of ^{234}Th.

The CaF_2 sludges are produced in quantities of about 1 MT (dry weight) CaF_2/MTU, by simple stoichiometry $UF_6 \leftrightarrow 3CaF_2$. The activity is less than 3.3×10^{-4} Ci of ^{238}U/MTU input, less than 1.6×10^{-3} Ci in total U/MTU, and comparable activity of ^{234}Pa. The CaF_2 sludge also has a fair degree of nuisance value as a chemical waste, because of the toxicity of fluoride and its limited but nontrivial solubility.

c. The Dry Direct Conversion (DDC) Process and its Wastes. This process produces wastes that are analogous to those in the wet ADU process; smaller quantities of waste are produced in the UF_6-to-UO_2 portion than in the comparable portion of the ADU process. Fluoride from UF_6 comes off quantitatively as HF gas, in both steam-decomposition $UF_6 \rightarrow UO_2F_2$ and

28

UO_2F_2 calcination steps, and is trapped to form $CaF_2/CaCO_3$ sludges. For both wet and dry processes, however, the major waste production lies in the mechanical fabrication steps, which are not materially affected by choice of the UF_6-to-UO_2 technique. Thus, we will only *outline* the DDC process.

UF_6 is delivered by heat and then hydrolyzed by steam in a fluidized bed (rather than in bulk solution). The resulting UO_2F_2 is reduced to UO_2 in two stages in a fluidized bed, using molecular hydrogen, H_2, as a reductant. Liquid effluent flows are much reduced and so are the wet filter wastes (which are only minor even for the ADU process).

3. Environmental Controls

Fabrication produces two basic types of waste, wet and dry, both of moderate radioactivity levels and of significant, but not yet overburdening, volumes; however, there is some concern about filling available burial sites with reactor-plus-fabrication wastes (Guilbeault and Reckman, 1979). The dry wastes are packaged, shipped, and buried off site, while CaF_2 sludges remain ponded on site. The ponds are occasionally landfilled.

Off-site shallow-land burial (SLB) of the dry wastes provides small potential for radionuclide migration back into the biosphere because fabrication wastes are non-volatile. They possess little solubility unless the continued practice of not segregating wastes at commercial SLB sites will later lead to introduction of mobilizing agents such as chelating compounds. This is a moderate potential concern because uranium-compound quantities are relatively small.

The ponded CaF_2 sludges are subject to few environmental controls beyond the siting of ponds "inside the fences" (which eventually is contravened by the decommissioning of the facility). The substantial insolubility of CaF_2 ensures that radionuclide-containing leachates will not be a significant problem unless ground water flows become large. In the latter case, the chemical hazard posed by fluorides when CaF_2 dissolves very slowly must be addressed by applying existing EPA regulations.

4. Conclusions

With the better defined SLB regulations coming on line (10 CFR 61), there should be little concern about dry fabrication wastes; unlike conversion wastes, the ^{226}Ra-daughter content is not significant. Nonetheless, we can not yet state whether the dry-waste disposal will

meet the proposed regulations for SLB because the regulations are not final and the technical controls of radionuclide mobilization are incomplete. The wet CaF_2 sludges pose a modest concern that the contaminating radionuclides might be mobilized by chelators or other agents after the facility is decommissioned; concern about the chemical hazard of fluoride leachates is at least as large. Technical controls of fluoride are more firm than for radionuclides, but must be mandated for decommissioned facilities, making use of ponding records.

G. Nuclear Power Reactors

1. Introduction

Light-water reactors (LWR) will be the major source of nuclear power in the US at least until the year 2000. There are currently in use two different types of LWRs. These are the boiling water reactor (BWR) and the pressurized water reactor (PWR). The wastes from these two reactor types differ both in quantity and radioactivity. This difference will be illustrated later.

Radioactive wastes from LWRs are caused by two different mechanisms. The first is the fissioning of fuel, which produces energy and fission products. The second is the activation of materials exposed to a high neutron flux. A third mechanism, neutron capture, produces a very small but important waste stream: transuranic waste (for example, ^{239}Pu).

Coolant water circulating through the reactor picks up small amounts of the fission products and transuranic material that have escaped from the fuel elements through failures in the fuel cladding. Also, the coolant water picks up activated materials in the form of corrosion products and minerals. The periodic replacement of reactor parts (shim rods, guides, instrumentation) is another source of activated wastes. Even though these parts are classified as low-level waste, some of them have radiation fields in the range of 100 000 R/hr. Fortunately, most of the activated materials have a short half-life (<5 yr).

2. Process

Reactor coolant water is sent through ion exchange and filter systems to remove the radioactive materials. Water from leaks through pump seals, valve stems, and cleanup activities is sent to evaporators to be reduced in

volume. Contaminated clothing, tools, and parts are separated into compactible and noncompactible fractions.

Ion exchange resins, filter sludges, and evaporator bottoms are dewatered and mixed with a sorbing material (such as vermiculite) or a solidification agent (for example, cement, urea formaldehyde) before being packaged for shipment in drums or tanks. The dry solids are packaged for shipment in plywood boxes.

3. Wastes Produced

Approximately 36 000 m³ of low-level wastes were generated by power reactors in 1978 with a total activity of 400 000 curies. This represents about 43% of the volume and 46% of the radioactivity of the low-level waste generated in the United States. About half are process wastes (spent resins, filter sludges, and evaporator bottoms), slightly less than half are dry compressible wastes and contaminated equipment, and 5% are irradiated reactor components (which contain 89% of the radioactivity). Typical quantities and activities are listed in Table VII. Typical radionuclides are ^{51}Cr, ^{54}Mn, ^{59}Fe, ^{58}Co, ^{60}Co, ^{65}Zn, ^{134}Cs, ^{136}Cs, ^{140}Ba, ^{141}Ce, and some transuranics.

TABLE VII

VOLUMES AND ACTIVITIES OF REACTOR
WASTES GENERATED PER YEAR[a]

Waste Form	m³	Ci
Spent resins, filter sludges, and evaporator bottoms	17 116	41 316
Dry compressible waste and contaminated equipment	16 653	2723
Irradiated components	1794	360 515
Total	35 563	404 554

[a]Sixty-six installation.
From: B. D. Guilbeault and B. J. Reckman, "Preliminary State-by-State Assessment of Low-Level Radioactive Wastes Shipped to Commercial Burial Grounds," NUS Corp., prepared for EG&G, Idaho (August 1979).

The amount and radioactivity of the waste generated by a nuclear power reactor is dependent upon whether the reactor operates with boiling water (BWR) or pressurized water (PWR), and further, on whether a condensate polishing system is used, and on the type of condensate polishing system. A boiling water reactor with a deep-bed condensate polishing system produces the maximum volume of waste—1000 m³ (untreated) with an activity of 4880 curies per year per Gigawatt-yr generated. In comparison, a pressurized water reactor produces about 490 m³ of waste contaminated by 420 curies of radioactivity per year per Gigawatt-yr (Table VIII). These wastes are designated as low level. (The high-level wastes are contained in the spent fuel rods, which are being stored at present.) Transuranic radionuclides are present in both PWR and BWR wastes at concentrations ranging from 0.02 nCi/g to 8.5 nCi/g (Phillips et al., 1979).

Implementation of Federal government regulations requiring the solidification of all waste before transportation to burial sites could result in increased volumes ranging from 610 to 1420 m³/yr from a 1-Gigawatt reactor (Phillips et al., 1979). Broad application of volume reduction techniques could increase the TRU content of the wastes to more than 10 nCi/g. This would then be TRU waste and could not be disposed of in shallow land burial facilities.

The waste types are deep-bed resins, concentrated liquids, filter sludge, cartridge filters, and trash (compactible and noncompactible). Table IX shows the total volume and activity of each waste type.

4. Environmental Controls

a. Treatment. NRC required the solidification of all wastes from facilities licensed after November 1975 when they issued the Regulatory Standard Review Plan and Branch Technical Position for Section 11.4 of the Safety Analysis Report. "Solidification" means that all waste should be in a solid, immobile form before shipment from the facility generating the waste. There are no NRC requirements for the installation of solidification systems in power plants licensed before November 1975, but all US commercial disposal sites require that wastes in liquid form be solidified before arrival at the site. The three operating disposal sites do accept wet solids. Cement or urea formaldehyde are currently used as solidification agents in US power plants; bitumen is used in European power plants.

TABLE VIII

UNTREATED WASTE VOLUMES AND ACTIVITIES FROM LWRs

| Waste Type | Boiling Water Reactors | | | | Pressurized Water Reactors | | | |
| | Deep–Bed CPS[a] | | Precoat CPS | | Without CPS | | With CPS | |
	Volume[b]	Activity[c]	Volume[b]	Activity[c]	Volume[b]	Activity[c]	Volume[b]	Activity[c]
Deep–bed resin	130	1900	6.5	1.4	27	610	9.1	200
Concentrated liquids	400	580	17	16	110	200	140	24
Filter sludge	150	2000	220	500	---	---	0.4	12
Cartridge filters	---	---	---	---	11	120	11	120
Trash								
Total	320	402	320	402	330	63	330	63
Compactible	220	5.2	220	5.2	220	4.9	220	4.9
Noncompactible	100	397	100	397	110	58	110	58
1 GWe plant	1000	4880	560	920	480	1000	490	420

[a]Condensate polishing system.
[b](m^3/GWe-yr).
[c](Ci/GWe-yr).
From: J. Phillips et al., "A Waste Inventory Report for Reactor and Fuel Fabrication Facility Wastes," NUS Corp. (Prepared for DOE Off. Waste Man.) March 1979.

The only volume reduction system commonly used in nuclear facilities is the trash compactor. The Tennessee Valley Authority (TVA), however, is studying the installation of volume reduction and solidification systems at their subject power plants. The systems will combine evaporation, calcination, and/or incineration of radioactive liquid and/or chemical wastes and combustible solids (Martin and Riales, 1981).

b. Packaging. Containers of treated (solidified and/or compacted) waste are placed in casks for transport to shallow land burial facilities. The most widely used container for compactible trash and process wastes (resins, concentrated liquids, and filter sludges) is the standard DOT Spec 17-H 55-gallon carbon steel drum, but larger metal containers may also be used (Phillips et al., 1979). For noncompactible trash, plywood boxes ranging from 0.02 to 3.6 m^3 in size are used.

The waste is classified according to four categories defined by 10 CFR 71. These four classifications are Low Specific Activity (LSA), Type A, Type B, and Large Quantities. Classifications are determined by the total activity of radionuclides in each of seven transport groups. A transport group is assigned to almost 300 radionuclides. The following is quoted from Phillips, 1979.

Proposed changes to 10 CFR 71 will eliminate the seven transport groups and establish curies limits on each radionuclide based on its own toxicity. One of the effects of this change will be to increase the quantity of less toxic members that can be shipped in a given container. For example, the 3 curie limit for Co-60 in Type A packages will increase to 7 curies. Other changes are a revised definition of LSA material and a new classification called "low level solids" (LLS). The changes to the LSA definition will essentially eliminate bulk liquid shipments as LSA material. For both the LSA and the LLS categories the specific activity

31

TABLE IX

SOLIDIFIED WASTE VOLUMES AND ACTIVITIES WITH CURRENT PRACTICES
(m^3/GWe and Ci/GWe installed)

| | Boiling Water Reactors | | | | Pressurized Water Reactors | | | |
| | Deep Bed | | Precoat | | With CPS[a] | | Without CPS | |
Waste Type	Volume (m^3)	Activity (Ci)	m^3	Ci	m^3	Ci	m^3	Ci
Deep-bed resin	150	1900	8.8	1.4	9	610	34	200
Concentrated liquids	620	580	31	16	250	200	230	24
Filter sludge	200	2000	305	500	7.1	---	---	12
Cartridge filters	---	---	---	---	11	120	11	120
Trash (all)	330	402	330	402	330	63	330	63
Total	1300	4880	675	920	607	1000	605	420

[a]Condensate polishing system.

From: J. Phillips et al., "A Waste Inventory Report for Reactor and Fuel Fabrication Facility Wastes," NUS Corp. (Prepared for DOE Office of Waste Management) March 1979.

limits are tied to the individual isotopic curie limits. Furthermore, the LLS definition considers the leachability characteristics of the solidification agent used to immobilize the radioactive waste. Under these revised regulations most of the LLS material from power reactors would be shipped as Type A material.

Some of the waste containers have to be placed in a shielded cask for shipment to the shallow land burial facility. The type of cask used depends on the waste category (LSA, A, B, or Large Quantity). The primary purpose of the cask is to ensure that the dose rates at specified distances from the transport vehicle do not exceed regulatory limits defined in 49 CFR 173.393. Type A casks must meet the requirements of DOT Specification 7A as found in 49 CFR 178.350 as well as requirements of 49 CFR 173.24 and 49 CFR 173.393. It must be capable of maintaining its shielding integrity and preventing dispersal in normal traffic accidents defined in 49 CFR 173.398 (b). Type B and Large Quantity casks are subject to more stringent requirements. Low-level wastes from power reactors are seldom shipped in Type B or Large Quantity casks.

c. Transport. NRC's 10 CFR 71 requires that a facility be licensed to deliver licensed materials in excess of Type A quantities to a carrier for transport. Various titles in the Code of Federal Regulations direct packaging, source, or special nuclear material, marking and labeling of the packages, loading and storage of packages, placarding of the transportation vehicle, monitoring requirements, and accident reporting. The wastes are shipped to one of three operating commercial shallow land burial facilities at Barnwell, South Carolina, Beatty, Nevada, and Richland, Washington. TVA has submitted requests to NRC for 5-yr license amendements to operate on-site waste storage facilities at the Browns Ferry and Sequoyah power plants (Martin and Riales, 1981).

5. Conclusions

Data on reactor waste volumes, activities, and characteristics are very sparse. Data used here are derived from a recent study by the NUS Corporation for the DOE Office of Waste Management (Guilbeault and Reckman, 1979). The study used data gathered in a survey of 29 nuclear power plants, including 12 BWRs

and 18 PWRs. Average volumes and activities were calculated from these data. In some cases, averages may have been calculated from as few as four numbers ranging from 0.0 to 2098. Standard deviation on the data would appear to be large. Even so, this is the best study to date.

Implementation of incineration, bituminization, and other volume reduction or solidification techniques will change the character of the waste. Although incineration reduces the volume, it also increases the radioactivity proportionally and the resulting ash is highly dispersable. We expect that the ash will have to be immobilized in concrete, resulting in a net reduction in volume of only about one-half.

All of the radioactive waste from power reactors is sent to commercial burial grounds for disposal (except for the irradiated fuel elements). Because existing disposal capacity is expected to be exhausted by 1990 (and before if an existing facility is closed), the increasing need for disposal capacity resulting from growing waste volumes along with the decreasing availability is probably one of the most serious problems facing the nuclear fuel cycle. Additional information on the disposal of low-level waste is presented in Sec. V.

Overall, the radioactive wastes from power reactors are carefully handled and well regulated. Environmental control technology is available and is being used. We have no major concerns in this area. A minor concern to be aware of is the difficulty control technology will have in containing tritium and krypton should it be required that these materials be captured and buried as a low-level waste.

H. Spent Fuel Storage

1. Introduction

All commercial power reactors in the United States are light water reactors (LWRs). The fuel is metal-clad uranium dioxide (UO_2) in which readily fissionable ^{235}U is enriched to 3 or 4%. The remainder of the uranium is relatively nonfissionable ^{238}U. The UO_2 pellets are encased in stainless steel or zirconium alloy (Zircaloy) cylinders (fuel rods), which are made into bundles (fuel assemblies) in a square array. As we discussed in Sec. G above, LWRs are divided into two classes—boiling water reactors (BWRs) and pressurized water reactors (PWRs). The fuel assemblies for these two reactors differ in design, size, and amount of fuel contained. Table X compares PWR and BWR fuel assemblies.

When the fuel in the reactor can no longer sustain a nuclear chain reaction at economic power levels, it is considered "spent" and replaced by fresh fuel assemblies. One-third to one-fourth of the reactor's fuel load is replaced each year. At discharge from the reactor, spent fuel contains about 4 g of fissile Pu and 8 g of ^{235}U/kg U and about 98% of the original ^{238}U. In addition, the fuel contains other fission products, activation products, and transuranic (TRU) elements. Radioactive decay produces intense radiation and a substantial amount of heat. Fission products will decay to about 0.1% of the original activity in about 300 yr. Transuranic elements have much longer half-lives. It takes about 250 000 yr for ^{239}Pu to decay to 0.1% of its original activity. Because of the differences in decay rates, the need for shielding and cooling decreases more rapidly than the need for isolation.

In anticipation of prompt reprocessing, most reactors were designed to accommodate only about 1-1/3 reactor loadings of spent fuel in reactor storage pools. Since the President's decision to dispose of spent fuel in geologic repositories, the number of fuel assemblies is increasing. At the end of 1978 there were about 4700 metric tons of uranium (MTU) stored as spent fuel and NRC forecasts (based on 230 GWe total annual power output) indicate that this will increase to about 81 700 MTU by the year 2000 (NUREG-0575, 1979). Forecasts by the Nuclear Assurance Corporation (based on 380 GWe) predict a total of 119 802 MTU as spent fuel in storage by the year 2000 (Woodhall, 1977). Faced with a shortage of space by the end of 1978, 65 of the then 69 operating reactors had obtained or were seeking licenses to expand their fuel storage capacity. The available methods will allow a 100 to 200% increase in fuel storage capacity at reactor sites (DOE/EIS-0015, 1980). In addition, two Independent Spent Fuel Storage (ISFS) facilities are in operation. General Electric's Morris plant (Morris, Illinois), a reprocessing facility licensed to store spent fuel, has a capacity of about 750 MTU. Morris is now storing 310 MTU (Eger and Zima, 1979) and has contracts for more. Nuclear Fuel Service's West Valley plant (West Valley, New York) is a former reprocessing plant, having a storage capacity of 260 MTU. NFS, now storing about 170 MTU, has withdrawn from the reprocessing business and is no longer receiving spent fuel. Licensing proceedings for Allied General's Barnwell plant (Barnwell, South Carolina) are currently in suspension. Should licensing for this plant go to completion, an additional 400 MTU of fuel storage capacity would become available.

33

TABLE X

DESCRIPTION OF LWR FUEL ASSEMBLIES[a]

	PWR	BWR
Overall assembly length, m	4.059	4.470
Cross section, cm	21.4 × 21.4	13.9 × 13.9
Fuel element length, m	3.851	4.064
Active fuel height, m	3.658	3.759
Fuel tube OD, cm	0.950	1.252
Fuel tube array	17 × 17	8 × 8
Assembly total weight, kg	658	283.9
Uranium/Assembly, kg	461.4	188.7
MO_2/Assembly, kg	523.4	214.1
Zircaloy/Assembly, kg	108.4	59.6
Hardware/Assembly, kg	26.2	10.2
Total metal/Assembly, kg	134.6	69.8
Radioactivity/Assembly, Ci[b]		
Fission and activation products	1.8×10^5	5.6×10^4
Actinides	4.6×10^4	1.4×10^4
Heat Generation/Assembly, W[b]		
Fission and activation products	6.5×10^2	2.0×10^2
Actinides	8.9×10^1	2.5×10^1

[a]From DOE/EIS-0046-D (1979).
[b]Activity and heat generation rates are for 6.5 yr after discharge.

2. Process

Spent fuel is stored in water basins to dissipate heat generated by the fuel and to protect the environment from radioactivity until it can be reprocessed or transferred to a geologic repository. We mentioned earlier that the heat-generating capacity of spent fuel decreases more rapidly than the radioactivity. Table XI shows the radioactivity and thermal power of PWR fuel at selected time periods after discharge from the reactor. Note that the thermal power decreases by almost 3 orders of magnitude after 5 yr of storage and that the radioactivity of ^{238}Pu, ^{240}Pu, and ^{241}Am holds steady or even increases.

Fuel storage operations are relatively simple and the technology to handle the fuel until final disposition is well developed. The fuel assemblies are packaged in DOT-approved transport casks at the reactor. Upon receipt of the fuel at the storage basin, the casks are washed to remove road dirt. During transportation, gases (^{3}H, ^{85}Kr, etc.) will have built up inside the container. These gases are vented to an off-gas treatment system. The casks are then lowered by crane into the unloading pool where the fuel assemblies are transferred to storage baskets, under water. The storage baskets are then transferred, by another crane, through an underwater transfer aisle to the storage basin. The empty casks are returned to the handling area for decontamination. All wash water and decontaminating solutions are routed to a waste water treatment system.

The temperature of the storage basin water is maintained at between 20 and 50°C (Johnson, 1977) to inhibit evaporation of the basin water and to reduce the growth of algae (DOE/EIS-0015, 1980). This can be accomplished with primary and secondary cooling systems. Basin water is circulated through a heat exchanger, which transfers the excess heat to a secondary cooling water system (DOE/EIS-0015, 1980). The cladding temperature of freshly discharged fuel is about 10°C above the bulk water temperature (Johnson, 1977).

The water in fuel storage basins is oxygen-saturated deionized water. About 2000 ppm of boron as boric acid are generally added to PWR storage basins to maintain an environment similar to that in the reactor, and hydrozaline is often added to keep iodine in solution (Johnson, 1977).

TABLE XI

RADIOACTIVITY AND THERMAL POWER IN SPENT FUEL[a,b]
(per MT uranium charged to reactor)

Radionuclide Content (Curies)	Years After Discharge		
	0	2	5
Important Activation Products			
$^{14}C^c$	6.6×10^{-1}	6.6×10^{-1}	6.6×10^{-1}
^{55}Fe	2.0×10^3	1.2×10^3	5.2×10^2
^{60}Co	6.3×10^3	4.8×10^3	3.3×10^3
^{63}Ni	5.5×10^2	5.5×10^2	5.3×10^2
^{95}Zr	2.8×10^4	1.2×10^1	1.0×10^{-4}
Total Activation Products	1.4×10^5	6.7×10^3	4.3×10^3
Important Fission Products			
^3H	5.1×10^2	4.6×10^2	3.9×10^2
^{85}Kr	1.1×10^4	1.0×10^4	8.3×10^3
^{90}Sr	7.8×10^4	7.5×10^4	6.9×10^4
^{106}Ru	5.3×10^5	1.3×10^5	1.7×10^4
^{129}I	3.7×10^{-2}	3.7×10^{-2}	3.7×10^{-2}
^{137}Cs	1.1×10^5	1.0×10^5	9.6×10^4
Total Fission Products	1.4×10^6	1.2×10^6	4.8×10^5
Important Transuranium Products			
^{238}Pu	2.7×10^3	2.8×10^3	2.8×10^3
^{239}Pu	3.2×10^2	3.2×10^2	3.2×10^2
^{240}Pu	4.7×10^2	4.7×10^2	4.7×10^2
^{241}Pu	1.0×10^5	9.4×10^4	8.1×10^4
^{241}Am	8.4×10^1	4.0×10^2	8.0×10^2
^{244}Cm	2.2×10^3	2.1×10^3	1.8×10^3
Total Transuranium Products	3.8×10^7	1.0×10^6	8.7×10^4
Thermal Power, Watts	1.0×10^6	5.9×10^3	2.1×10^3

[a]Data from DOE/EIS-0015 (1979).
[b]Calculated with the ORIGEN code for PWR fuel irradiated to 33 000 MWD/MTU at a specific power of 30 MW/MTU.
[c]Based on 2.5 ppm nitrogen (by weight) in UO_2.

Radionuclide concentrations are maintained at between 10^{-3} and 10^{-4} $\mu Ci/ml$ through the use of filters and ion exchange systems (Johnson, 1977; Eger and Zima, 1979). Table XII shows the concentrations of some radionuclides in fuel storage basins. During fuel discharge operations, a combination of dissolved and particulate species in the reactor primary coolant and species released from fuel assembly surfaces will mix with storage basin water, causing activity levels in the water to rise by as much as 0.5 $\mu Ci/ml$ (Johnson, 1977).

These higher activity levels are returned to normal in a relatively short period of time by means of the basin water cleanup system (Eger and Zima, 1979; DOE/EIS-0015, 1980). Short-lived species (for example, I) appear in reactor storage basins, but are rare in ISFS basins; tritium is also substantially higher in reactor basins (Johnson, 1977).

Basin water may also be contaminated by particulates. Activation products in the basin water are primarily from corrosion products from reactor primary circuits

TABLE XII

RADIONUCLIDE CONCENTRATIONS IN FUEL STORAGE BASINS[a,b]
(μCi/mℓ)

Radionuclide[c]	Current Generation Basins with High-Integrity Fuel	Early Generation Basins	Older DOE Basins
^{3}H		1×10^{-3}	
^{54}Mn		1×10^{-6}	
^{58}Co	5×10^{-6} to 3×10^{-5}	2×10^{-5} to 1×10^{-4}	
^{60}Co	5×10^{-6} to 1×10^{-4}	1×10^{-5} to 1×10^{-4}	5×10^{-5}
^{90}Sr		2×10^{-5}	2×10^{-2}
^{131}I	1×10^{-7}	1×10^{-6}	
^{134}Cs	1×10^{-5} to 1×10^{-4}	3×10^{-4}	3×10^{-3}
^{137}Cs	3×10^{-5} to 1×10^{-4}	5×10^{-4} to 1×10^{-3}	3×10^{-2}
^{140}Ba		1×10^{-5}	
^{144}CePr			3×10^{-3}
Pu			2×10^{-5}
Total, μCi/mℓ	1×10^{-7} to 1×10^{-3}	1×10^{-3} to 1×10^{-2}	1×10^{-2} to 5×10^{-1}
Dose Rate, mrem/hr	<1	<5	5.50

[a]Data from Johnson (1977).
[b]At equilibrium conditions; higher values are generally present immediately following reactor fuel discharge.
[c]At some pools, the following isotopes are also significant: ^{141}Ca, ^{140}La, ^{63}Ni, ^{89}Sr, ^{95}Zr/^{95}Nb.

that deposit as "crud" layers on fuel assembly surfaces. These corrosion product layers range from 25 to 50 μm thick on BWR fuel assemblies. On PWR assemblies, they are much thinner or absent (Johnson, 1977). Crud layers are insoluble at basin water temperatures, but do release particulates to basin water. Some of these particulates are removed by the basin water cleanup system. At some facilities (for example, Morris), vacuuming the basin is used to control the buildup of particulates (Johnson, 1977; Eger and Zima, 1979; DOE/EIS-0015, 1980). Table XIII compares some characteristics of some representative spent fuel storage basins.

3. Wastes Produced

The primary waste is, of course, the spent fuel itself. Nominal exposure rates (burnups) of irradiated fuel are about 33 000 megawatt days/metric tons of uranium (MWD/MTU) for PWR fuel and 27 000 MWD/MTU for BWR fuel (DOE/EIS-0015, 1980). Burnups may be lower because of reactor maintenance and peak power output schedules. Table XIV shows typical burnups for spent fuel in storage. The fission products in spent fuel occur in the fuel matrix (DOE/EIS-0015, 1980). The activation products are corrosion products that have

plated out on the hardware components of the fuel assembly. These corrosion products are irradiated in the high neutron flux in the reactor and activated to a radioactive state. Neutron capture in the fuel matrix produces the long-lived transuranics.

The wastes generated by spent fuel storage operations are solids, liquids, and gases. Solid wastes constitute about 95% of the total waste volume produced by fuel storage operations (DOE/EIS-0015, 1980). These wastes include ventilation filters, rags, protective clothing, plastic, wood, rubber, paper, failed equipment, and tools.

About 98% of the total activity in fuel storage wastes is found in the liquid waste fraction (DOE/EIS-0015, 1980). Liquid (more properly termed "wet") wastes are filter sludges, ion exchange resins, and detergent solutions from the operation of water treatment and decontamination systems.

We have been unable to find any data regarding the actual volumes of fuel storage wastes. Wastes generated by fuel storage operations at reactor sites are mixed with those from other reactor operations and are not identified separately. Most publications that estimate volumes, for example, WASH-1248 (1974), NUREG-0002 (1976), do so in the context of fuel reprocessing and these estimates may not be wholly representative of

TABLE XIII

CHARACTERISTICS OF FUEL STORAGE BASINS[a]

	ISFS Facilities		Reactors	
	Morris	NFS	Dresden (BWR)	Three-Mile Island (PWR)
Water Volume (m³)	2555	3142	3800	984
Temperature (°C)	25 - 35	27 - 32	27 - 32	---
Water Chemistry				
pH	4.4 - 9 (mean 5.8)	6 - 8	6.5 - 8.5	5.2 - 5.5
Cl^- (ppm)	0.02 (norm)	---	0.1	0.01
Conductivity (μmho/cm)	1	20	0.5 - 1	8.2 - 8.5
Boron (ppm)	1	---	---	2120 - 2140
Lithium (ppm)	---	---	---	low
Heavy metals	---	---	<0.1	NA
Radioactivity Concentration				
Total (μCi/mℓ)	5×10^{-2} (max)	4×10^{-4} (norm)	8×10^{-4} (norm)	5×10^{-2} [b]
Cs (^{134}Cs,^{137}Cs)	2.8×10^{-4}	---	---	4×10^{-3}
Cobalt	0.8×10^{-4}	---	---	7×10^{-2}

[a]Data from Johnson (1977).
[b]Maximum during refueling.

TABLE XIV

MAXIMUM BURNUP OF STORED SPENT FUEL[a]

Cladding	Reactor	Burnup (MWD/MTU)	Discharge Date
Zircaloy-2	BWR	25 000	1974
Zircaloy-4	PWR	33 160	1976
Stainless Steel	BWR	22 000	1975
Stainless Steel	PWR	33 200	1973

[a]From Johnson (1977).

current waste generation rates. Based on estimates for reprocessing facilities and waste volume projections for reactors and ISFS facilities, we suggest that the current waste production rate for fuel storage operations is probably in the range of 5000 to 6000 m³/yr. No estimates of radioactivity can be made.

4. Environmental Controls

The management of radioactive wastes at spent fuel storage facilities depends upon an array of support systems designed to dissipate heat, control water quality, and treat gaseous and liquid wastes.

The temperature of the basin water is controlled by primary and secondary cooling water systems. Basin water is pumped through a heat exchanger, which transfers the excess heat to a secondary cooling water system (DOE/EIS-0015, 1980). Some facilities, such as Morris, rely upon a single cooling water system. The water is passed through the heat exchanger several times. Performance has been adequate (Eger and Zima, 1979).

The quality of basin water is controlled by the basin water cleanup system. Basin water is circulated through a series of filters and ion exchange systems to remove radioactive species. Filter sludges and ion exchange resins are dewatered and solidified. Performance has been quite adequate, maintaining radionuclide concentra-

tions in the water at between 10^{-3} and 10^{-4} μCi/mℓ (Tables XII and XIII). Storage basins are sometimes vacuumed to reduce the buildup of particulates.

Gaseous wastes are controlled by the off-gas and building ventilation systems. Radioactive gases released during cask venting are routed directly to the off-gas treatment system, where they pass through scrubbers, HEPA filters, and an iodine absorber to remove most of the iodine and particulates before release to the atmosphere via the stack. Table XV shows the amounts and activities of potentially gaseous and volatile elements in spent PWR fuel, and Table XVI shows the estimated fractional release of some radionuclides from off-gas systems. Ventilating air from the rest of the basin area is filtered and released directly to the atmosphere. Table XVII shows the estimated atmospheric release of radionuclides resulting from fuel handling and basin storage operations. Estimated atmospheric releases from the Morris plant for the last half of 1974 through the first half of 1978 were 135 μCi α and 394 μCi β (Eger and Zima, 1979).

The largest volume of waste water is produced by cask washing and decontamination operations. Liquid wastes are collected, treated by filtration and ion exchange, and dewatered by evaporation.

At reactor sites, filter sludges and ion exchange resins are solidified with cement or urea formaldehyde, or

TABLE XV

AMOUNTS AND ACTIVITY OF POTENTIALLY GASEOUS AND VOLATILE ELEMENTS IN SPENT PWR FUEL[a,b]
(per fuel assembly)

Element	Amount (g-moles)		Activity (Ci)	
	10 yr	100 yr	10 yr	100 yr
³H	7.32×10^{-3}	4.5×10^{-5}	2.15×10^2	1.36
He	3.35×10^{-1}	3.86×10^{-1}	0	0
¹⁴C	1.25×10^{-2}	1.23×10^{-2}	7.79×10^{-1}	7.7×10^{-1}
Br	1.24×10^{-1}	1.24×10^{-1}	0	0
⁸⁵Kr	1.94	1.88	2.29×10^3	6.79
¹²⁹I	8.42×10^{-1}	8.42×10^{-1}	1.46×10^{-2}	1.46×10^{-2}
Xe	18.34	18.34	0	0
¹³⁴Cs	---	---	2.5×10^3	0
¹³⁵Cs	---	---	1.59×10^{-1}	1.59×10^{-1}
¹³⁷Cs	8.15	5.33	3.82×10^4	4.77×10^3

[a]Data from Jenks (1979).
[b]33 000 MWD/MTU at 0.461 MTU/assembly.

TABLE XVI

ATMOSPHERIC RELEASE FRACTIONS FROM OFF-GAS SYSTEMS AT STORAGE BASIN FACILITIES RECEIVING SPENT FUEL[a]

Nuclide	Fraction of Fuel Leaking	Fraction of Activity Released to Cask Cavity	Fraction to Off-Gas System	Fraction to Atmosphere	Overall Release Fraction to Atmosphere
^{3}H	1×10^{-4}	1×10^{-2}	1	1	1×10^{-6}
^{14}C	1×10^{-4}	3×10^{-2}	1	1	3×10^{-6}
^{85}Kr	1×10^{-4}	3×10^{-1}	1	1	3×10^{-5}
^{129}I	1×10^{-4}	1×10^{-1}	1	1×10^{-3}[b]	1×10^{-8}
Particulates[c]	1×10^{-4}	1×10^{-4}	0.1	1×10^{-4}[d]	1×10^{-13}

[a]DOE/EIS-0015 (1980).
[b]Charcoal filters in off-gas system.
[c]Assumed to be other fission products and actinides.
[d]Air passed through a prefilter and a HEPA filter.

TABLE XVII

ATMOSPHERIC RELEASE FRACTION FROM STORAGE BASINS[a]

Nuclide	Fraction of Fuel Leaking	Fraction of Nuclide Activity Released to Water Basin	Fraction Released to Room Air	Fraction to Atmosphere	Overall Atmospheric Release Fraction
Spent Fuel Handling[b]					
^{3}H	0.002	0.02	1	1	2×10^{-5}
^{14}C	0.002	0.03	1	1	6×10^{-5}
^{85}Kr	0.002	0.3	1	1	6×10^{-4}
^{129}I	0.002	0.1	0.01	1	2×10^{-6}
Other Fission Products	0.002	0.0001	Negligible	Negligible	Negligible
Actinide	0.002	0.0001	Negligible	Negligible	Negligible
Spent Fuel Storage					
^{3}H	0.001	0.01	1	1	1×10^{-5}
^{14}C	0.001	0.02	1	1	3×10^{-5}
^{85}Kr	0.001	0.2	1	1	3×10^{-4}
^{129}I	0.001	0.1	0.01	1	1×10^{-6}
Other Fission Products	0.001	0.0001	Negligible	Negligible	Negligible
Actinide	0.001	0.0001	Negligible	Negligible	Negligible

[a]From DOE/EIS-0015 (1980).
[b]Failure is assumed to occur during transfer of fuel from the shipping cask to storage or during transfer of fuel from storage to a shipping cask.

sorbed onto an adsorbent material (such as vermiculite). The product is then packaged and shipped for commercial burial (Phillips et al., 1979). At the NFS West Valley plant, liquid wastes are precipitated as ferric hydroxide sludge, packaged in drums, and buried on site (Kibbey and Godbee, 1980). At the Morris plant, liquid wastes are discharged to a 2600 m^3 underground vault. Excess liquids are routed through an evaporator and the concentrate returned to the vault (Eger and Zima, 1979). Currently, all sludges at Morris are stored in the vault, but the installation of a solidification facility is planned (Eger and Zima, 1979). The NFS plant has a vault/evaporator system similar to that at Morris (Nuclear Fuel Services, Inc., 1973).

Solid wastes, ventilation filters, protective clothing, trash, tools, failed equipement, etc. are produced in nearly every aspect of fuel storage operations. Solid wastes are compacted when possible, packaged, and shipped for commercial burial (Phillips et al., 1979; DOE/EIS-0015, 1980).

5. Conclusions

Spent light water reactor fuel is stored in water basins at reactor sites and Independent Spent Fuel Storage facilities. The water basin concept is designed to dissipate heat generated by the spent fuel and to protect the environment from radiation. Water temperature is controlled by water cooling systems and radioactivity in the basins is controlled by filtration and ion exchange.

Ninety-five per cent of the waste volume produced by fuel storage operations consists of dry solids (ventilation filters, trash, protective clothing, tools, and failed equipment). Solid wastes are compacted, when possible, packaged, and shipped for commercial burial. Gaseous wastes, produced by cask venting operations and a small amount of radionuclides escaping from storage pools, are treated by an off-gas treatment system to remove most of the iodine and particulates before being released to the atmosphere via the stack.

Liquid wastes, mostly from cask washing and decontamination activities, are treated by filtration, ion exchange, and evaporation, producing sludges. At reactor sites, these sludges are solidified, packaged, and shipped for commercial burial. At the NFS plant, sludges are packaged and buried on site. At the Morris plant, all sludges are stored in an underground vault.

Data regarding waste volumes are scarce, but the annual waste generation rate from fuel storage operations is probably in the range of 5000 to 6000 m^3/yr. No data are available regarding total waste activity.

Spent LWR fuel has been stored successfully for some time. All fuel storage facilities are licensed and under NRC supervision, and adequate environmental controls seem to be provided. Commercial burial of wastes generated by fuel storage operations will be discussed in Sec. V.

I. Conclusions - Nuclear Fuel Cycle

The nuclear fuel cycle is the largest source of radioactive wastes. Because detailed characterization of waste is necessary to determine the adequacy of environmental controls, we considered each individual step in the fuel cycle to identify the characteristic wastes generated, the available waste treatment options (and which ones are being used), and the ultimate mode of waste disposal. Different types of facilities use different methods to treat any given waste type. For example, combustible wastes are incinerated at conversion, enrichment, and fuel fabrication plants (though primarily for uranium recovery) but not at reactors. Ion exchange resins and filter sludges from reactor cooling water and fuel storage basin water cleanup are packaged and shipped to commercial burial facilities. The same wastes generated at Independent Spent Fuel Storage facilities are stored on site.

Another interesting feature of the nuclear fuel cycle is the difference in availability of information regarding waste volumes and activities, depending on the licensing status of the facility. Wastes from licensed facilities, subject to NRC regulations and supervision, are more fully documented. Unlicensed facilities are not subject to NRC review and information is rare to almost completely absent. Even more striking is the lack of data for facilities that store wastes on site rather than use commercial burial. These stored wastes are generally considered nonexistent in a *de facto* sense and will probably remain so until the facility is decommissioned. The volumes and activities of stored wastes at these sites are rarely reported in any meaningful way because, technically, there are no wastes, only effluents.

The major wastes produced by uranium mining operations are overburden from open pit mines, waste rock from underground mines, and waste water from mine dewatering activities. Mining operations are the source of wind- and waterborne contamination. Little data are available regarding mine spoils. Although spoils piles are at present poorly regulated (especially at inactive mines), we feel that an effective reclamation program can be developed under RCRA. Land reclamation pertinent to uranium mining is to be specifically addressed by EPA in the fall of 1980.

Of greater concern is the large quantity of contaminated water pumped from uranium mines. Mine dewatering activities cause changes in the aquifer, transport contaminants (both radioactive and toxic) to the surface, and deplete a natural resource (a very important resource in the western US). The extent of these activities is largely undocumented. As mines become deeper, the volume of pumped water will continue to increase. Although adequate controls (lined ponds, chemical treatment) are feasible, a lack of appropriate state regulations coupled with complex federal/state regulatory interactions have permitted gaps that could be environmentally significant. Although uranium mine wastes are a large problem, mining activities are not under DOE's jurisdiction. EPA is preparing a report to Congress that identifies potential mine waste hazards and should recommend appropriate actions. We feel that developments should only be monitored at the present time.

Potentially significant environmental effects may result from radiological and toxic trace elements in uranium mill tailings piles. Radon is considered to be the single greatest contributor to radiological risk from mill tailings. Another concern is the seepage of contaminated mill process water or water from natural precipitation through the tailings and into ground water supplies.

As a result of the Uranium Mill Tailings Radiation Control Act of 1978, DOE remedial action programs are addressing the problems of stabilizing inactive mill tailings piles. Environmental controls at newly licensed mills will be regulated by NRC's new Uranium Mill Licensing Requirements. The long-term effectiveness of these regulations has not been proven. The Environmental Safety and Engineering Division must maintain an interest in the outcome of these actions.

Our major concern, however, is the potential health risk and environmental degradation posed by existing uranium milling operations licensed before NRC's new guidelines became effective and for which remedial action may not be "practicable." Environmental controls at these sites seem to be at a minimum and documentation is poor. If remedial actions are necessary, cost and feasibility should be determined.

Insufficient data have been collected to date regarding the middle part of the fuel cycle: UF_6 conversion, isotopic enrichment, and fuel fabrication. Waste volumes and activities are generally accepted as being "very low," so there seems to be little concern for most radioactive elements in these wastes. We do not have enough data to say whether this view is justified. Solid wastes from conversion and fabrication plants are shipped off site for commercial burial or stored on site at enrichment plants. Liquid wastes from all three sources are discharged to holding ponds to allow contaminated materials to precipitate before being relased to nearby surface waters.

We are somewhat concerned about the holding pond sludges. In some instances, ponds are periodically dredged and the sludges buried on site as solid waste. We do not know the extent and effectiveness of these dredging operations. All ponds are located on site and should, therefore, pose no immediate problems. Monitoring data concerning the effectiveness of holding ponds as an environmental control are scarce. At present, the ultimate disposition of the sludges after site decommissioning is not clear. Something will have to be done eventually. Chelating and leaching may mobilize both radioactive and toxic elements, releasing them to the environment. We are collecting more information regarding environmental controls at these facilities.

Commercial power reactor and spent fuel storage operations have been conducted under licensed supervision for many years. Wastes generated by these sources seem to be carefully handled and well regulated. We have only a few minor points to make regarding reactor wastes. Reactor operations produce large volumes of waste, all of which are buried commercially. Existing disposal facilities are expected to be filled by about 1990. The increasing need for disposal capacity will soon become a serious problem unless more attention is paid to volume reduction. Volume reduction is a recurrent theme and will be addressed again in Sections IV and V.

At present, reactors are licensed to release ^3H and ^{85}Kr as gaseous effluents. If these elements should be declared wastes, they will have to be captured and buried as LLW. We doubt that current technology is prepared to effectively contain these gaseous radionuclides. The effectiveness of LLW disposal methods will be discussed in Sec. V.

IV. INSTITUTIONAL WASTES

A. Introduction

Institutional low-level radioactive wastes are generated by more than 16 000 licensed hospitals, medical schools, universities, and private industries throughout the United States, and account for 25% of the total low-level waste disposed of annually in commercial burial grounds (Guilbeault and Reckman, 1979; Oertel, 1980). Typical wastes include depleted radiation sources, accelerator

targets, chemical reagents, animal tissues, absorbent papers, laboratory glassware, vials, and excreta. Though waste volumes are high, the contaminating radionuclides are generally short-lived and the overall activity content is low.

Not all institutional wastes are buried. The facilities producing these wastes are licensed, but regulation seems to be somewhat relaxed, pemitting a wide variety of alternative disposal methods, including municipal land-fills and sewers, and on-site burial. In view of the large volumes of waste produced and the range of waste forms and management practices, we find it rather surprising that institutional wastes have not received more attention in the past. Most of the available information is contained in a survey of large medical and academic institutions conducted by the University of Maryland (Anderson et al., 1978; Beck et al., 1979) and a preliminary report by NUS Corporation (Guilbeault and Reckman, 1979). For this reason, we feel it is important to include institutional wastes in this overview assessment.

B. Wastes Produced by Institutions

Twenty-five per cent of all low-level waste buried commercially in the United States is generated by the institutional population. Table XVIII shows, by source, the volumes of waste currently buried. Guilbeault and Reckman (1979) report that the activity of institutional wastes buried in 1978 was only ~0.3% of the total from all sources.

Table XIX shows the volumes and activity shipped for burial by the large medical and academic institutions surveyed by the University of Maryland. Although the data base includes only part of the total institutional population, it suggests growth trends. The volume of waste shipped for burial by the study population increased 17% between 1975 and 1977, but the activity apparently increased only 4% during the same period.

To determine the relationship between use and waste production, the Unversity of Maryland survey (Beck et al., 1979) divided institutional wastes into three "waste streams." These are the medical, bioresearch, and non-bioresearch waste streams. The wastes produced by the medical waste stream result from administering radioactivity to human patients for diagnostic or therapeutic purposes. A lesser amount of medical waste is produced by routine clinical assay techniques such as radioimmunoassay or other analyses used to quantify amounts of hormones, proteins, or other biochemical species in

blood or urine. Wastes include syringes, vials, pipettes, reagents, unused radiopharmaceuticals, patient excreta, and exhalations during patient respiration. Probably most of the radioactivity released from hospitals is patient excreta released to the sewer, largely in an uncontrolled manner (Beck et al., 1979). Seventy-five per cent of the radionuclides contaminating medical wastes have half-lives shorter than one week, predominantly ^{99m}Tc (Beck et al., 1979; Kibbey and Godbee, 1980). Table XX shows the most common radionuclides in medical wastes having half-lives longer than one week.

The bioresearch waste stream results from bio-chemical, biophysical, and physiological research, using radiolabeled tracer techniques, including *in vivo* animal research. Bioresearch waste forms include dry solids, liquid scintillation vials, adsorbed liquids, and biological wastes (primarily animal carcasses and tissues). The most significant contribution to bioresearch waste is liquid scintillation (L-S) vials. Data suggest that many L-S vials are disposed of as dry solids, but it is impossible to determine the fraction of dry solid wastes that are L-S vials (Beck et al., 1979).

Liquid scintillation fluids constitute an estimated 50% of the adsorbed liquids in bioresearch wastes, not including the quantity of L-S fluids disposed of in the vials (Anderson et al., 1978). The most common disposal method for L-S fluids is shipment for burial, though much of it is apparently released to the sewer. Because the activity content of L-S fluids is low, the primary hazard is chemical rather than radiological (Granlund, 1978). The major solvent used in L-S fluids is toluene, but xylene and 1,4 dioxane are also used. Triton X-100, ethoxyethanol, methanol, and ethanol are often added to labeled solutions to increase solubility (Beck et al., 1979). Other organic liquids and compounds are often added as labeled tissue solubilizers or as oxidizing agents. These include perchloric acid, Hyamine hydroxide, and ethanolamine phenethylamine (Beck et al., 1979). Scin-tillators in the fluids may be PPO, PBD, or butyl PBD, naphthalene, POPOP, DMPOPOP benzene, bis MSB benzene, or PBBO. Furthermore, the fluid contains the radiolabeled biochemical being measured, usually in blood, serum, or tissue fluids (Beck et al., 1979).

The wide range of radiolabeled compounds in bioresearch is very complex. The most common tracer is tritium (^{3}H). Labeled tritiated compounds include nucleic acids, amino acids, fatty acids, hormones, steroids, drugs, toxins, and carcinogens (Beck et al., 1979). However, in view of the total amount of labeled compounds used in bioresearch, the hazard presented by the solvent is the more significant (Beck et al., 1979).

42

TABLE XVIII

ESTIMATED ACCUMULATED LOW-LEVEL-WASTE VOLUMES BURIED AND PRESENT AMOUNTS SHIPPED TO BURIAL ANNUALLY[a]

Source	Total Accumulation[b]	Present Annual Burial Rate
Government	1.4×10^6	$\sim 2.8 \times 10^4$
Commercial[c]	4.5×10^5	$> 5.6 \times 10^4$
Fuel Cycle		3.4×10^4
Reactor Operation		$\sim 2.8 \times 10^4$
Fuel Fabrication		$\sim 5.7 \times 10^3$
Other Steps		Small
Non-Fuel Cycle		$\sim 2.2 \times 10^4$
Institutional		$\sim 1.9 \times 10^4$
Medical		$\sim 1.8 \times 10^4$
Academic		$\geq 7.5 \times 10^2$
Industrial and Other Research (including pharmaceutical)		$\sim 3 \times 10^3$
Total	1.9×10^6	$\sim 8.4 \times 10^4$

[a]From Kibbey and Godbee (1980).
[b]Accumulation times: government $\simeq$ 4 decades, commercial $\simeq$ 2 decades.
[c]Includes all categories listed below; mill tailings are excluded.

TABLE XIX

ESTIMATED VOLUMES AND ACTIVITY OF WASTE SHIPPED FOR BURIAL BY SURVEY INSTITUTIONS[a]

Year	Volume (m³)	Activity (Ci)[b]
1972	4627	---
1973	5148	---
1974	5265	---
1975	6448	1628
1976	6977	---
1977	7771	1688 [c,d]

[a]Data from Andersen et al. (1978) and Beck et al. (1977).
[b]Includes sealed sources.
[c]22% of the activity was in the form of sealed sources.
[d]56% of the activity was particle generator waste.

TABLE XX

PRIMARY RADIONUCLIDES IN MEDICAL WASTES HAVING HALF-LIVES LONGER THAN ONE WEEK[a]

Radionuclide	Half-Life	Per Cent of Total
^{131}I	8.05 d	76
^{125}I	60.20 d	8
^{75}Se	120.40 d	5
^{51}Cr	27.80 d	4
^{32}P	14.28 d	4
^{169}Yb	37.80 d	1
^{3}H	12.30 yr	1

[a]Data from Beck et al. (1979).

In contrast to medical wastes, nearly all the radio-nuclides in bioresearch wastes have half-lives longer than one week. Table XXI shows the most common radio-nuclides in bioresearch wastes. In a University of Maryland survey, bioresearch institutions contributed 79% of the total waste volume shipped for burial (Beck et al., 1979).

The nonbioresearch waste stream is produced mostly by colleges and universities. The wastes result from investigations in physics, inorganic chemistry, earth science, and materials science, but excluding life science. Approximately half the activity shipped for burial as sealed sources is produced by nonbioresearch institutions (Beck et al., 1979). Sealed sources are radioactive materials permanently sealed, encapsulated, or affixed in nondispersible form, and include radiation sources, accelerator targets, and neutron generator targets. Other nonbioresearch wastes include dry solids, gaseous wastes liquid scintillation vials, adsorbed liquids, and laboratory glassware (Beck et al., 1979).

The waste forms produced by the institutional population are varied. The most common institutional waste forms are shown in Table XXII.

An interesting feature of institutional wastes identified by the University of Maryland survey lies in the area of accelerator wastes (wastes from neutron generators, cyclotrons, Van de Graff generators, etc.). For the 1977 study population, accelerator operators shipped ~940 Ci of activity in 11.8 m^3 (Beck et al., 1979). Thus, 56% of the activity was contained in a negligible fraction of the waste volume (see Table XIX). Most of the activity in accelerator wastes is in the form of spent tritium targets from neutron generators.

TABLE XXI

PRIMARY RADIONUCLIDES IN
BIORESEARCH WASTES[a]

Radionuclide	Half–Life	Per Cent of Total
^{3}H	12.30 yr	75
^{125}I	60.20 d	11
^{32}P	14.28 d	5
^{14}C	5730 yr	3
^{35}S	87.90 d	3
^{51}Cr	27.80 d	2

[a]Data from Beck et al. (1979).

C. Environmental Controls

In 1978, institutional wastes contributed 25% of the volume and 0.3% of the activity of all low-level radioactive wastes buried commercially. Table XXIII shows the volumes and activity of institutional wastes buried in 1978, by waste form; Table XXIV shows the typical radionuclides in institutional wastes. A University of Maryland survey (Andersen et al., 1978; Beck et al., 1979) identified, for the study population, a 60% increase in the volume of liquid scintillation vials and a 44% decrease in the volume of adsorbed liquids shipped for burial between 1975 and 1977. This increase in L-S vials and decrease in adsorbed liquids suggests that increasing volumes of L-S fluids are being buried in the vials or released to sewers, rather than being adsorbed into a media such as vermiculite. Beck et al. (1979) state that the increase in L-S fluids (both adsorbed and non-adsorbed) between 1975 and 1977 was probably about 11%.

Packaging procedures for wastes vary from one institution to another and from one kind of institution to another. The University of Maryland survey reported that, for the study population, about 77% of the total waste volume was shipped in 55-gallon steel drums. Packaging alternatives are shown in Table XXV.

Not all institutional radioactive wastes are shipped for commercial burial. Table XXVI shows the various alternative disposal methods used by institutions. Volume reduction techniques are sometimes used before disposal by one of the alternative methods. These include burning solid wastes, either in controlled incineration or in uncontrolled open pit burning and evaporation or distillation of liquid wastes (Beck et al., 1979).

All institutions use one or more of these alternative disposal methods for at least part of their waste and no data are available regarding the volume of waste disposed of by these methods. The Barnwell, South Carolina site, formerly receiving 80% of institutional waste volume (Beck et al., 1979) no longer accepts L-S fluids. These wastes must be transported to other sites at proportionally higher costs. The lack of sufficient regional disposal sites and increasingly stringent packaging requirements also increase disposal costs. As these costs increase, so will the number of institutions that resort to alternative disposal. Commercial burial may become economically unfeasible for many smaller institutions.

We feel that more attention needs to be paid to waste treatment. Only 12% of the large institutions surveyed in 1977 used any volume reduction technique for dry solid

TABLE XXII

INSTITUTIONAL WASTE FORMS[a]

Waste Form	Description
L-S Wastes	Waste liquid scintillation vials and fluids, including empty vials
Organic Liquids	Waste organic liquids other than L-S fluids
Aqueous Liquids	Solutions of water soluble radionuclides, including laboratory glassware washings
Biological Wastes	Primarily animal carcasses and tissues, including animal bedding excreta and labeled culture media
Patient Excreta	Excreta or materials contaminated with excreta from patients undergoing radiodiagnostic or radiotherapeutic treatment
Gaseous Wastes	Almost exclusively ^{133}Xe used in human or animal ventilation studies
Dry Solids	All dry solid waste materials containing real or suspected radioactive contamination

[a]From Beck et al. (1979).

TABLE XXIII

INSTITUTIONAL WASTES BURIED IN 1978[a,b]

Waste Form	Volume (m³)	Activity (Ci)
Dry Solids	8761	1026
L-S Vials	9223	1081
Adsorbed or Solidified Liquids	1461	171
Biological Wastes	1803	209
Total	21 248	2487

[a]Data from Guilbeault and Reckman (1979).
[b]Based on 2390 institutions.

TABLE XXIV

TYPICAL RADIONUCLIDES IN INSTITUTIONAL WASTES[a]

Nuclide	Half-Life	Per Cent of Total
^3H	12.30 y	29.1
^{14}C	5730 y	8.9
^{32}P	14.28 d	5.3
^{35}S	87.90 d	2.0
^{51}Cr	27.80 d	1.6
^{67}Ga	77.90 d	0.1
^{99m}Tc	6.05 h	30.9
^{125}I	60.20 d	3.5
^{131}I	8.05 d	5.9
Others		12.7

[a]Data from Guilbeault and Reckman (1979).

TABLE XXV

WASTE VOLUMES PACKAGED FOR SHIPMENT BY THE
1977 SURVEY INSTITUTIONS[a]
(by per cent volume)

Waste Type	55-Gallon Steel Drums	30-Gallon Steel Drums	Double[b] Drums	Cardboard or Fibreboard Boxes/Drums	Other Metal Cans or Drums	Wooden Crates
Dry Solids	72.1	8.1	0.1	18.1	1.4	0.3
L-S Vials	82.6	16.8	0.1	>0.1	0.4	---
Adsorbed or Solidified Liquid	59.3	8.1	25.6	1.8	5.20	---
Biological Wastes	65.1	16.9	4.0	12.8	1.2	---

[a]Data from Beck et al. (1979).
[b]30-gal steel drum within a 55-gal steel drum.

TABLE XXVI

ALTERNATIVE DISPOSAL METHODS USED FOR
INSTITUTIONAL WASTES[a]

Disposal Method	Description
Sewer Disposal	Release of liquid or semi-solid waste to municipal sewer system (either controlled or uncontrolled)
Common Refuse	Disposal of solid waste in municipal solid refuse system
Commercial Burial	Shipment of waste to commercial burial site
On-Site Burial	Burial on land owned by the institution
Transfer	Transfer of the waste to another or parent institution for disposal (usually sealed sources)
Venting to Atmosphere	Release of gaseous wastes (primarily ^{133}Xe) to the atmosphere

[a]From Beck et al. (1979).

46

wastes, and only 25% were burning at least part of their biological wastes (Beck et al., 1979). The volume of institutional waste, especially bioresearch, that is buried commercially is increasing at about the same rate as low-level waste from all sources (Beck et al., 1979). In view of the low activity-to-volume ratio of institutional wastes, if this trend continues, current burial sites will soon be filled with large volumes of low-activity waste. Beck et al (1979) suggested that, in 1977, large institutions shipped for burial 431 000 ℓ of liquid scintillation fluids in the vials and an additional 132 000 ℓ as adsorbed liquids. Andersen et al. (1978) suggested that 50% of absorbed liquids are organic liquids. This results in a total of 563 000 ℓ of L-S fluids and organic liquids shipped by large institutions in 1977. Volume reduction techniques such as evaporation/distillation and incineration could help to mitigate the chemical and biological hazards presented by these wastes.

Another area of concern is waste segregation. Institutional wastes are often not segregated from other higher activity wastes in burial grounds. Many additives in liquid scintillation fluids are designed to keep radionuclides in solution. If these fluids should come in contact with higher activity wastes, radionuclides could be solubilized, facilitating migration away from the burial grounds.

The Nuclear Regulatory Commission (NRC) has recently proposed to deregulate some biomedical wastes. Animal carcasses and L-S fluids containing less than 0.05 mCi/g ^{3}H or ^{14}C will be considered nonradioactive. The proposed rule will also raise the limit for sewer-disposal liquids from 1 to 7 curies. Other new limits are 1 curie for undefined waste, 1 curie for ^{14}C, and 5 curies for ^{3}H (Federal Register, October 8, 1980). The ultimate effects of such a ruling are not clear.

D. Conclusions

Institutional wastes are generated by many thousands of licensed medical, educational, and research facilities. In most cases the volumes are high, but the activity is quite low. More than 50% of the activity in institutional wastes is contained in a negligible fraction of the waste volume and, although the volume of these wastes seems to be increasing at about the same rate as other low-level wastes, data suggest that the activity-to-volume ratio is decreasing.

Medical wastes result from administering radioactivity to patients for diagnostic or therapeutic purposes and clinical assay. These wastes are primarily patient excreta, which are released in an uncontrolled manner to municipal sewer systems. Most of the radionuclides in medical wastes have half-lives of less than one week.

Bioresearch wastes are produced by biochemical, biophysical, and physiological studies, including *in vivo* animal research, using radiolabeled tracer techniques. Seventy-nine per cent of all institutional wastes are produced by bioresearch. The largest fraction of these wastes is liquid scintillation vials, liquid scintillation fluids, and other organic liquids. Other wastes include dry solids and animal carcasses and tissues.

Nonbioresearch wastes result from research in the physical sciences and from particle accelerators and research reactors. Most of the activity as sealed sources is produced by nonbioresearch facilities. Other wastes are extremely varied.

In addition to commercial burial, institutional wastes are disposed of by any of several alternative methods, including release to municipal sewer and refuse systems, on-site burial, and venting of gaseous wastes to the atmosphere. Volume reduction techniques, such as incineration, compaction, and evaporation or distillation are used by some institutions. In view of the large volumes and low activity of institutional wastes, we feel that waste treatment should receive more attention. Because of the chemical and biological hazards of liquid scintillation and other organic fluids, it is surprising that evaporation/distillation and incineration of these fluids is not more common. Incineration and/or compaction of solid wastes could result in a sizeable decrease in waste volumes without increasing activity-to-volume ratios to unacceptable levels.

More care should be taken at burial sites to segregate adsorbed liquids and wastes suspected of containing liquids from wastes generated by other sources (for example, reactor wastes). These fluids could solubilize radionuclides and increase their migration potential.

As the cost of commercial burial increases, the number of institutions disposing of wastes by other methods (sewer, trash, etc.) can be expected to increase. Wastes currently disposed of by these alternative methods are generally contaminated with very short-lived radionuclides, but it is not unreasonable to suggest that, if commercial burial becomes economically unfeasible, radionuclides having longer half-lives may be disposed of by inappropriate means. A proposed NRC rule to deregulate some biomedical wastes may help to alleviate some of the institutional waste disposal problems.

The primary hazards presented by institutional wastes are actually chemical and biological rather than radio-

logical, which, compared to many fuel cycle wastes, are relatively low. Because DOE seems to have no direct jurisdiction over institutional wastes and the hazards are currently relatively low, we suggest that these activities should only be monitored at the present time.

V. DISPOSAL OF RADIOACTIVE WASTES

A. Introduction

As we discussed in Sec. III of this report, the most significant source of radioactive waste is the nuclear fuel cycle. Nonfuel cycle sources, however, such as research, nonpower oriented industry, medicine, and education also produce substantial volumes of waste. Some of these were described in Sec. IV. Levels of activity in wastes vary drastically and several different methods are required to deal with them.

High-level waste (HLW) is defined in 10 CFR 50 as aqueous wastes from the first cycle solvent extraction and concentrated wastes from subsequent extraction cycles during fuel reprocessing. No mention is made of activity levels. Perhaps a more rational definition is that proposed by the International Atomic Energy Agency in 1978. By this definition, HLW is any waste containing more than 1 Ci of alpha or 100 Ci of beta-gamma radiation per metric ton. A clear activity-based definition of HLW could help keep some high-activity wastes out of shallow land burial sites. High level wastes are currently stored at DOE facilities. These wastes will ultimately be interred in a Federal repository.

Transuranic (TRU) wastes are produced primarily by spent fuel reprocessing, and by research- and weapons-related activities at DOE facilities. They are defined as wastes containing more than 10 nCi of TRU activity per gram of material. Because TRU wastes result from spent fuel reprocessing, they are limited to government sources. The major problem with these wastes is the long-lived alpha emitters (for example ^{238}Pu, ^{241}Am) they contain.

Low-level wastes (LLW) are those that are not spent fuel or HLW and contain less than 10 nCi of TRU activity per gram of material (10 CFR 50). This definition is misleading because some wastes, though low in TRU content, may contain very high beta-gamma activity.

The current method of LLW disposal is shallow land burial (SLB), which uses burial pits, trenches, and shafts excavated in the near surface environment. There are five DOE and six major commercial SLB facilities, and many smaller sites owned by universities, fuel cycle facilities (only three are currently active), and other institutions. Shallow land burial facilities may be classified according to the hydrogeologic conditions at the site (arid, well-drained humid, and poorly drained humid).

Spent fuel, HLW, and TRU wastes are presently in storage, awaiting the opening of a federal repository. The major emphasis in technology development is directed toward geologic disposal, and it is expected that a repository will be operational by the mid-1990s. An alternative near-term option to geologic disposal is seabed disposal. The Seabed Disposal Program is aimed at determining the feasibility and developing the technology to dispose of spent fuel, HLW, and TRU wastes in sediments on the ocean floor.

B. Shallow Land Burial

1. Introduction

Waste materials are generated by every human activity involving the use of natural resources. Those waste materials associated with the nuclear industry have come to be known as "radioactive wastes." The "nuclear industry" includes the Department of Energy, private industry, hospitals and research institutions, and facilities associated with the nuclear fuel cycle. The term "radioactive waste" implies nothing specific about the actual radioactive content of the waste material; rather it suggests the source of the waste material. Nearly all natural materials are radioactive to some degree, and wastes from many nonnuclear industries contain higher concentrations of radioactivity than do some "radioactive" wastes (McBride et al., 1978).

The present categorization of radioactive wastes is a definitional nightmare. High level wastes include spent fuel and liquid wastes from the first step of solvent extraction during fuel reprocessing. Transuranic wastes are those wastes, other than high level, that contain more than 10 nCi/g of transuranic activity. Large volume wastes, such as mine and mill tailings, or slurries from conversion, are dealt with at the point of generation. Nearly all other radioactive wastes are loosely referred to as low level wastes.

The term low level is a misnomer. It includes, at one extreme, wastes that are essentially noncontaminated material, such as paper goods or clothing used in a radioactively contaminated area but not brought into direct contact with radioactivity. Frequently the expense of monitoring such material to certify it as

noncontaminated is greater than the disposal costs. At the other extreme are ion exchange resins or activated components from reactors, which will have radionuclide concentrations measured in curies/m^3, and require biological shielding during handling and transport. These low level wastes have historically been disposed to shallow land burial sites at DOE or commercially operated facilities. A large volume of liquid radioactive wastes was disposed of by discharge to seepage pits or absorption beds at AEC (now DOE) facilities. In general, this practice has ceased and these wastes are either solidified and disposed of by shallow burial or stored in tanks for some alternative disposal method.

2. Waste Characteristics

Detailed descriptions of the waste derived from both DOE and commercial sources are contained in the relevant sections of this report. However, a brief review of these wastes is appropriate before describing the disposal technology.

a. Solid and Liquid Residues from the Fuel Cycle. Liquid wastes are commonly solidified on site through ion exchange, evaporation, or some other treatment process, and the resulting solids disposed of off site. The solid wastes include routine trash, contaminated or obsolete equipment, and periodically replaced reactor components. These wastes are contaminated with nuclides in the uranium and thorium decay chains, fission and activation products, and low concentrations of transuranic nuclides. They are classified generally as non-TRU wastes, although transuranic concentrations may approach the 10 nCi/g level.

b. Industrial Wastes. Various kinds of radioactive wastes are generated by both nuclear and nonnuclear industries. These may include radioactive sources or tracers used in test procedures, manufacturing residues, etc. Few data are available on their exact character.

c. Institutional/Medical Wastes. These include material from hospitals, private research institutions, and universities, including used radiation sources, analytical wastes, such as liquid scintillation vials, contaminated biological material, and routine trash. With the exception of radium, ^{14}C, and ^{3}H (tritium), the radionuclides involved are relatively short lived.

d. DOE Non-TRU Wastes. These include a broad range of wastes from weapons and energy development activities. They are contaminated with nearly every element in the periodic chart, but the primary radionuclides (in terms of curie quantities) are similar to those in fuel cycle wastes. These wastes may contain transuranic elements, but at concentrations below the 10-nCi/g limit.

e. TRU Wastes. If commercial fuel reprocessing were to become a reality again, these plants would produce TRU waste, which, at present, is generated only by DOE facilities or their contractors. These wastes result primarily from weapons work, although nuclear heat source and reactor fuel research also generate some TRU waste. These TRU wastes are currently stored in a retrievable fashion, awaiting eventual disposal in some other manner.

Before about 1970, wastes now defined as TRU were placed in shallow land burial facilities at both DOE and commercial sites.

A complete understanding of the characteristics of the low level waste now disposed of by shallow burial would include the curie concentrations of all the significant radionuclides in the waste. Such data are not available for all waste material, and there are no standardized record keeping systems to produce the data for all wastes. The Department of Energy maintains a computerized waste inventory system, SWIMS, which identifies volumes and radionuclide content of major waste forms. Data for commercial sites are derived, at best, from information supplied by the waste generator. Additionally, spot surveys have been conducted of commercial waste generators, both fuel cycle and nonfuel cycle (Phillips, 1979). From these studies, data can be extracted that summarize the volumes and ranges of radionuclide content of wastes from various sources, as shown in Table XXVII.

3. Description of Current Burial Sites

At present there are five major DOE burial sites and six commercial sites (only three of which are currently operating). Many additional smaller sites are operated by universities, fuel cycle facilities, or other institutions. However, these last sites handle a relatively small fraction of the total waste disposed annually.

TABLE XXVII

VOLUMES AND ACTIVITIES OF LOW-LEVEL WASTE

Source	Annual Volume (m³)	Concentration Ranges (Ci/m³)
DOE	6.5×10^{4a}	0.14^b - 300^c
Commercial		
Reactors	3.5×10^{4d}	0.16^e - 200^f
Institutional	2.1×10^{4d}	0.12^g
Industrial	2.0×10^{4d}	23^h

[a]From DOE SWIMS data for 1979.
[b]Average low-level alpha.
[c]Average induced activity.
[d]From Phillips, 1979.
[e]Routine trash.
[f]Irradiated components.
[g]Overall average, includes α, β, and γ.
[h]Overall average - no data on radionuclide composition.

The major DOE and commercial sites are located in a wide range of climatic and hydrogeologic environments. These will be described under three separate headings—arid, well-drained humid, and poorly drained humid (Fig. 5). Although the waste forms and radionuclides will vary, the primary distinction between DOE and commercial sites lies in the source of the waste rather than its character. A current exception to that is the presence of large volumes of facility decommissioning wastes in DOE burial grounds, and relatively little at commercial sites. That situation will change in the future as existing fuel cycle facilities (including reactors) become obsolete and qualify for decommissioning.

Specific examples will be cited where possible to illustrate classes of problems; however, no attempt is made to identify every example of that type of problem. The examples cited were selected based on how well they illustrated the condition, rather than to provide a judgement of the adequacy or inadequacy of that specific practice.
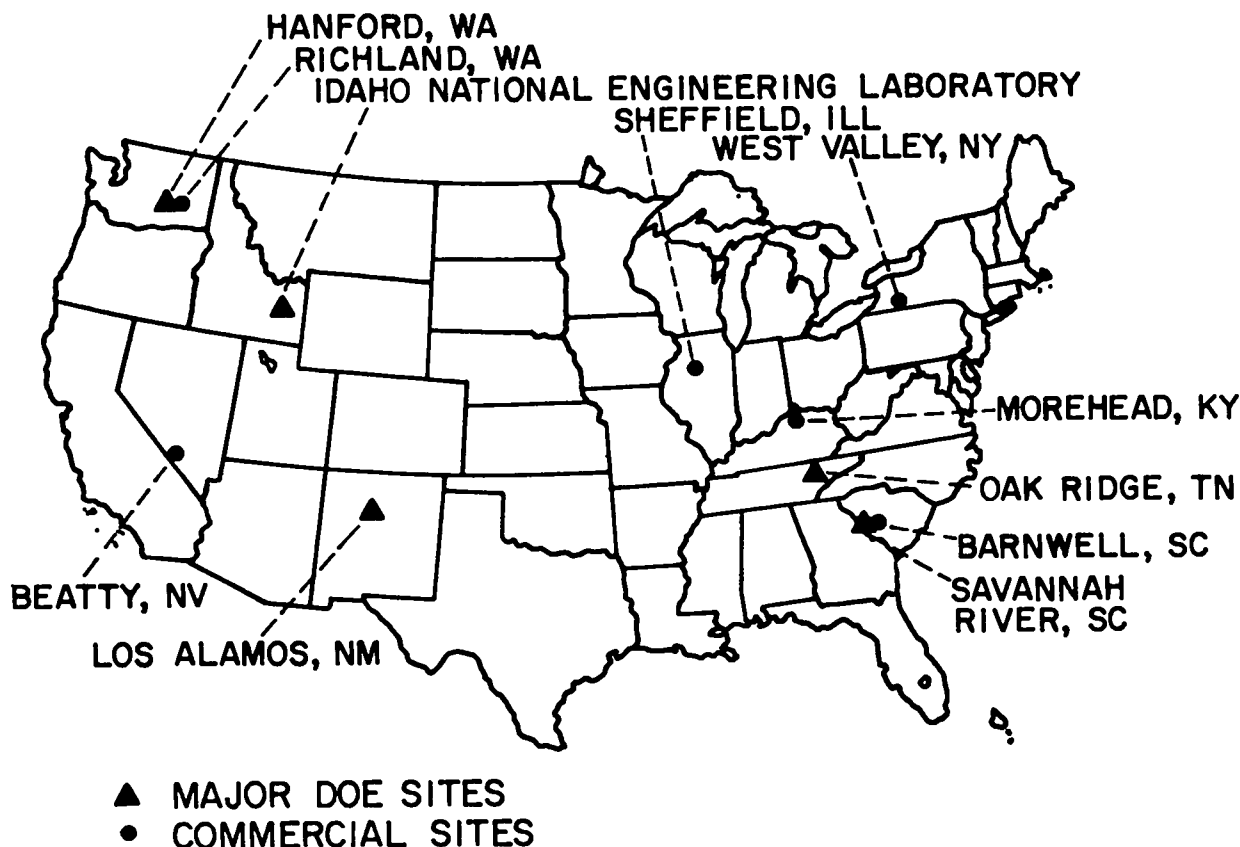

Fig. 5. Location of disposal sites.

a. Arid. Several major burial sites (Los Alamos, Hanford, Richland, INEL, Beatty) are located in arid portions of the west. Precipitation is less than 500 mm/yr, with potential evaporation well in excess of that amount. Saturated conditions rarely if ever occur within the burial trenches. Ground water levels beneath the sites are in excess of 60 m below the surface, and ground water recharge from the site areas is nonexistent or very small. Radionuclide transport by soil water is dominated by partially saturated flow substantially less than saturation, with gaseous or vapor transport playing a strong role. The movements are, therefore, quantitatively small, but difficult to monitor precisely.

Current release of radionuclides from these sites is dominated by physical intrusion of the biosphere into the waste, by both plants and animals. High concentrations of tritium in clover on one site at Los Alamos (Hakonson et al., 1973), and concentrations of ^{137}Cs in rabbitbrush at Hanford (Klepper et al., 1979) are typical examples.

The use of simulation modeling to explain or predict site performance is limited, in part, by the highly irregular character of environmental conditions at these sites. Soil erosion, for example, is dominated by high-intensity rain storms that may last an hour or less, with a return frequency of 5 to 10 yr. Wind erosion, controlled in part by soil moisture and vegetative conditions, is difficult to model quantitatively.

b. Humid, Poorly Drained. Humid climates are typified by precipitation that equals or exceeds the potential evaporation. Soil moisture levels are high most of the year and saturated zones commonly exist fairly near the surface.

A common philosophy controlling the location of disposal sites was the requirement for relatively impermeable materials as a disposal medium. This is characteristic of the sites at Sheffield, West Valley, Maxey Flats, and Oak Ridge. The soils and subsoils are derived from glacial till or shale, and there is substantial surface runoff. Saturated zones occur as perched water bodies near the surface, but are generally not of sufficient extent or permeability to be considered aquifers. The upper 2 to 5 m of the ground is weathered, commonly resulting in secondary permeability such as microfractures in shale (ERDA-7643, 1976).

The transport of water and radionuclides is dominated by saturated or near-saturated flow through fractures or other anomalous high-permeability zones (such as the sand lenses in glacial till at West Valley). Typically, the permeability of the burial medium is less than that of the trench cap, leading to water accumulation in trenches. Surface overflow may result (such as at West Valley), or high hydraulic heads may produce lateral flow in fractures (such as at Oak Ridge).

Two concerns dominate at these sites. First, it has been historically difficult or impossible to construct trench caps of sufficiently low permeability and longevity that water will not enter the waste material and accumulate under saturated conditions. The permeability of the natural material is too low to permit adequate drainage away from the surface. Second, localized nonhomogeneities in the burial medium allow transport (often lateral) at unpredictable locations. The presence of migration is thus difficult to monitor and, when identified, difficult to interpret. This ambiguity has led, rather directly, to the closing of sites such as Maxey Flats, where the specific cause of environmental releases has been impossible to document.

c. Humid, Well-Drained. Two essentially adjacent sites exist in this category; Barnwell and Savannah River, South Carolina. The surface soils are well drained, consisting of sand and sandy silt or clay. There are water tables at depths of 15 to 20 m, but saturated zones within the trenches are uncommon to nonexistent. At both these sites the shallowest water table drains laterally to surface streams, and is underlain by aquifers that are not recharged in the vicinity of the site. Hydrologic studies show that flow is upward from the underlying aquifers to the overlying saturated zones.

Transport of water and radionuclides is dominated by partially saturated flow, but site drainage can be designed to reduce water infiltration of trench caps. Permeability of the burial medium is sufficient to allow vertical drainage away from the burial trenches and generally precludes water accumulation in the trenches.

Water flow is relatively uniform over the site and can be monitored through conventional soil or ground waer sampling methods. The hydrology of the sites is sufficiently simple to permit simulation modeling. Savannah River, for example, relies on calculated ground water flow times to provide for the decay of tritium release from the trenches before its discharge into off-site surface waters.

4. Basis for Evaluation

The objective of environmental controls is to limit, to within some standard, the extent to which radionuclides or other contaminants in waste material enter the biological environment or become entrained in environ-

mental pathways that will eventually release them to the biosphere. In describing and evaluating shallow burial, no specific distinction need be made between DOE and commercial facilities. Further, no attempt has been made to detail each specific operation or to evaluate its appropriateness to the local environmental conditions. Rather, general environmental conditions will be described and the class of problems arising in each will be detailed.

In assessing shallow burial as a control technology, a distinction must be made between the environmental release itself (the radionuclide concentration entering the biosphere) and the process leading to that release. As will be shown, our understanding of the process is far more important than the actual magnitude of the release, even when the release is within existing standards. The point at which a control must be applied (waste form, trench boundary, site boundary) will depend a great deal on our understanding of the transport process and the potential for environmental degradation. Thus, the discussion that follows will identify, when possible, the potential for environmental contamination, the extent to which the control process is understood, and the success of the control as currently applied.

5. Adequacy of Control Technology

In evaluating the adequacy of SLB to provide environmental controls on radioactive waste, several questions come to mind.

● How do we measure the adequacy of SLB?
● What is the character of the control provided by SLB?
● What is the performance record of shallow burial sites?
● What kind of improvements are needed in SLB?

a. Measurement of Adequacy. The most forthright means of determining the adequacy of SLB is to compare releases from existing sites with current standards. However, to date no standards have been promulgated that specifically apply to environmental releases from waste disposal facilities. Quantitative evaluations are most commonly based on maximum permissible concentrations (MPC) of radionuclides in air and water (NUREG-0216, 1977). Depending on the specifics of the evaluation, known discharges from current sites are within drinking water standards or airborne concentration limits (Duguid, 1976).

However, the use of MPC is not appropriate to evaluation of these releases. These values were derived for occupational exposure conditions, or for application to continuous effluent release. Rather, pathway analyses leading to dose-to-man calculations are most acceptable. Standards are being proposed that limit chronic and acute exposures to workers and the general public. However, these standards will limit "calculated" exposures. That is, the releases from a given site must be established before the site is developed, and the impact of those releases on some future population determined. The procedures that will guide these evaluations are obscure. Unlike an industrial facility, the range of remedial actions that can be applied to correcting an inadequate burial ground are relatively limited. Thus, there must be substantial assurance, before the fact, that the burial ground will perform as expected.

Evaluations have been conducted of both DOE and commercial burial grounds, and well illustrate the difficulties of measuring their adequacy.

The Department of Energy (formerly ERDA) contracted for the development of monitoring methodologies for the five major burial grounds (Dames and Moore, 1976). Presumably, this methodology would provide a means of measuring the adequacy of the sites, based on the specific environmental transport pathways operative at each site. However, a general insufficiency of site data precluded the development of such a monitoring program. Rather, the contractor determined to evaluate the existing data and identify the need for further data collection. Tables XXVIII and XXIX are reproduced from that report.

TABLE XXVIII

RELATIVE DATA "ADEQUACY" RATING (R_w) FOR EACH SELECTED BURIAL GROUND

Laboratory Site	Burial Ground	R_w[a]
ORNL	Burial Ground #4	0.31
SRP	Old Burial Ground	0.48
INEL	Main Burial Ground	0.41
HW	200—West Grounds	0.44
Los Alamos	Burial Ground G	0.36

[a]Higher values of R_w indicate greater adequacy.

52

RELATIVE PRIORITY (Pd) FOR FURTHER DATA COLLECTION FOR EACH SELECTED BURIAL GROUND

Laboratory Site	Burial Ground	Pd[a]
ORNL	Burial Ground #4	0.53
SRP	Old Burial Ground	0.21
INEL	Main Burial Ground	0.14
HW	200—West Grounds	0.10
Los Alamos	Burial Ground G	0.10

[a]Higher values of Pd indicate higher priority.

The ranking system employed provides a measure of the adequacy of existing data for development of a monitoring program and a measure of the relative need for collection of additional data. The latter ranking is an implicit statement of the potential for environmental contamination at each site. Such a ranking distinguishes between an understanding of the release process and definition of the magnitude of the environmental impact.

A somewhat comparable evaluation of existing commercial sites has been undertaken by the US Geological Survey (USGS). A program to develop hydrologic data at five of the sites (Barnwell, Sheffield, West Valley, Maxey Flats, and Beatty) was begun in 1976. The study was intended to develop data on site performance, contributing to the establishment of hydrogeologic siting criteria. The results of those studies are not currently available, as the field investigations lasted substantially longer than was initially proposed.

A summary of performance data on commercial sites was prepared by the NRC (NUREG-0216, 1977). Descriptions are presented of the observed releases at these sites and the processes leading to them. Data on West Valley, Maxey Flats, and Barnwell were presented, illustrating that radionuclide concentrations in waters on or near the site were substantially below MPC values. The report concluded that

These problems have not resulted in risks to public health and safety and corrective actions are being taken.

Two of these three sites have been closed because of questions regarding site performance. Obviously, one measure of the adequacy of shallow burial is its general acceptability as a disposal method. The fact that three of the original six commercial sites are closed speaks for itself. The reasons for these perceived inadequacies will be explored below.

b. The Character of Environmental Control Provided by SLB. An SLB facility is inherently a passive system; there should be no active control processes other than those provided by the interactions of waste form, site engineering, and site conditions. Thus, the "control" consists of properly designing that system in advance of its use. Environmental monitoring provides us with a method for comparing site performance with our expectations. The expectations of site performance are commonly phrased in terms of how effective the site will be at *containing* the radioactivity. However, recognition of the operational impossibility of *complete containment* leads to an evaluation of the adequacy in terms of the rate of release of radionuclides from the site.

Radioactive waste, because of the immutable character of radioactive decay, has focused attention on *how long* the wastes will remain hazardous. For example, early work in the use of seepage pits for disposal of liquid wastes recognized this as a process of *storage* of the radionuclides in a soil column for sufficient time to permit decay to some acceptable level before off-site discharge occurred (Division of Reactor Development and Public Health Service, 1955). Continued emphasis on *containment* of solid wastes in near-surface pits led, naturally enough, to a realization that site control would be required. While burial grounds were considered "dedicated," the site control period was originally envisioned as "decades in length."

In recent years increasing attention is being focused on the issue of site control. The proposed NRC criteria for shallow land burial (10 CFR 61) stated that the long-term adequacy of the site was of primary concern in site selection and design. This adequacy must be evaluated in terms of environmental release processes acting essentially independent of man, and in terms of impacts on potential future "intruders" or "reclaimers" at the site. The intruder is one who excavates to some significant depth at the site to intentionally intersect the waste or for some unrelated purpose. The reclaimer is one who uses only the surface of the site, perhaps for agricultural activity, but may derive his water supply from ground or surface waters originating on site. It is apparent that the impact on these future individuals is strongly dependent on the various assumptions used to calculate the exposure, on the initial radionuclide concentrations at the

site, and of course on the length of time between initial disposal and the exposure event.

There are several possible approaches to ensuring the long term adequacy of the site. Physical control and continued maintenance are planned for all sites following site closure. However, it is difficult to establish precisely how long site control is required or for how long it can be assured. The former issue, the required length of the control period, would seem to be relatively straightforward. Radionuclides decay at some fixed rate, and it is a simple matter to calculate the length of time for some initial concentration to reach some lower concentration. However, our knowledge of the initial concentrations of radionuclides in waste is extremely limited. Data on curie quantities, such as those presented in Sec. III, are frequently based on calculations from external radiation measurements, or on what "should" be present in the waste stream. Further, there is no general acceptance of a *de minimus* radionuclide concentration in any foreseeable time frame. The concept of a "dedicated" site begins to break down when we are forced to specify a time period over which site control must be maintained. Clearly, we cannot rely on institutional controls forever.

A second and somewhat similar approach is to acknowledge that site control may be lost after some reasonable time period—100 to 200 yr is commonly proposed (Adam and Rogers, 1978). Initial concentrations of radionuclides could then be controlled in order to bring "calculated" exposures to the intruder or reclaimer within acceptable limits. This approach is, of course, limited by many of the same restrictions as the first. It provides additional "control" insofar as it requires knowledge of the input concentrations.

Yet a third approach is to provide some measure of protection for these future release events through waste and burial site engineering. Measures such as deeper burial of the more hazardous materials, encapsulation in monolithic blocks, or other intruder barriers have been proposed. Of course, man's historical efforts at preventing intrusion into his treasure vaults has been notoriously ineffective (Wacks, 1979). Whether such experience can be extrapolated to preventing intrusion into our waste dumps is open to question. It certainly seems unreasonable to apply exhaustive efforts to prevent someone, sometime, from digging, for some reason, into our waste. Without a more precise definition of the problem, a clear solution is fugitive at best.

One means of providing increased assurance against future intrusion, particularly by the unaware excavator or reclaimer, is to perform a potential land use analysis as part of site selection. The site selection could then be biased toward locations that discourage particular types of land use. For example, agricultural activities are very unlikely at many western locations simply because of the absence of water. Conversely, selection of a site that was in the path of expected surburban expansion might enhance the likelihood of future intrusion. Such considerations have been proposed (10 CFR 61), but much work is needed to develop the capability.

No clear-cut answer has emerged from these various possible solutions to the site control issue. A wide range of burial limits has been derived from the many radionuclides in low level waste (Adam and Rogers, 1978). The only one currently enforced is the 10 nCi/g limit on burial of transuranic contaminated wastes. This limit, derived rather simply from considerations of hazards posed by radium in natural uranium deposits, has been widely challenged, but remains in effect perhaps more because of its simplicity than of validity.

c. Performance Record. The adequacy of SLB as a disposal method must be addressed with regard to both short-term and long-term goals. For the short term we must determine whether radionuclide releases have been within expected limits. For the long term we must consider the capability of the site to store waste radionuclides until site control is no longer required.

(1) Expected Releases. There is no clear measure of expected site performance. Ideally, one would wish a pre-use prediction of the rates of radionuclide release along all possible pathways, and a later comparison of experience with those predictions. This is not possible.

Site evaluations were couched in more general terms. Criteria for sites required low permeability soil material, water tables deeper than the buried waste, absence of major potable water supplies downgradient from the site, etc. The actual retention of contaminants in the site was not calculated, but it was expected that it would be nearly quantitative. To the extent that releases from the sites have been small and of no particular public health significance, those expectations have been met. To the extent that the particular releases were anticipated and quantitatively predicted, site performance has not been as expected.

It is possible to evaluate performance in a qualitative sense. Did the sites perform as well as we had hoped? Can we *explain*, using current knowledge, the observed releases at existing sites? Can we use this knowledge to design and select improved sites in the future?

The answer to the first two questions is a qualified no. To the last question, a qualified probably.

Sites located in arid environments, as can be seen from Table XXX, have generally performed in an acceptable manner (Wheeler, 1976). But then, the near absence of water at these sites has effectively eliminated one of the dominant release pathways. Small releases have occurred because of vapor migration, plant uptake, or plant intrusion. Although our scientific understanding of these processes may be limited, the releases serve more as focuses of attention than as focuses of concern.

Sites in humid environments, particularly at the poorly drained sites, have commonly *not* performed as hoped, and great difficulty has been encountered in explaining the observed releases (see Table XXIX). The actual significance of the releases, in terms of public health, is somewhat argumentative. As mentioned earlier, no clear standards exist. Qualitatively, however, strong evidence exists that our *understanding* of the sites is not adequate because *three* of the four commercial sites in the east are no longer operating. In each case, the sites were closed ostensibly because of environmental concerns. Although there was general agreement that releases were not a threat to public health, it was not possible to present, in public arena, convincing arguments regarding site performance.

Although DOE sites in the east are not subject to the same form of public scrutiny, at least one site (ORNL) displays serious failings that raise questions regarding the adequacy. Remedial measures are being investigated to reduce the radionuclide discharges from ORNL (Duguid, 1976).

TABLE XXX

PERFORMANCE RECORD OF BURIAL SITES

Site & Classification	Burial Medium	Radionuclide Releases		Probable Cause of Release	Extent of Release
		Tritium	Other		
Arid					
Beatty, Nevada	Alluvial sand and gravel	No	No		
Hanford, Washington	Clay, sand, and gravel	No	Yes	Plant Uptake	Near Trench
INEL, Idaho	Alluvial sand and gravel	No	No	---	---
Los Alamos, New Mexico	Volcanic tuff	Yes	No	---	Near Trench
Richland, Washington	Clay, sand, and gravel	No	No	---	---
Humid, Poorly Drained					
Maxey Flats, Kentucky	Shale	Yes	Yes	Water Accumulation in Closed Trenches and Effluents from Liquid Treatment	Offsite
ORNL, Tennessee	Shale	Yes	Yes	Shallow Water Table	Offsite
Sheffield, Illinois	Glacial drift; sand, silt, and gravel	Yes	Yes	Shallow Water Table	Onsite
West Valley, New York	Glacial drift; silty clay/interbedded sand and gravel	Yes	Yes	Water Accumulation in Closed Trenches and Effluents from Liquid Treatment	Onsite
Humid, Well-Drained					
Barnwell, South Carolina	Sand and clay sand	No	No	---	---
SRP, South Carolina	Sand and clay sand	Yes	No	Near Surface Ground Water	Onsite

(2) Causes of Environmental Release. As has been mentioned, the general strategy applied to the use of shallow burial is that the burial trench will provide for sufficient containment that radionuclides will not be released to the environment at unacceptable rates. As pointed out in Table XXX, a frequent cause of radionuclide release has been entrainment in shallow water tables. These saturated zones have frequently developed within the burial trench, because of inadequate capping, or changes in the surface topography at the site. Although it was implicitly recognized that water could accumulate in trenches, it would appear that inadequate attention was given to preventing that process. It should be noted that this water-related release process has only been significant at poorly drained humid sites, where low permeability of the soil surrounding burial trenches caused ponding within the waste. The distinction between natural conditions that unavoidably led to the releases, and inadequate management that allowed the condition to occur, is a difficult one to make. Further, it is difficult to install a comprehensive subsurface monitoring system in conditions characterized by low permeability and heterogeneous materials.

There would seem to be several fundamental fallacies underlying the use of shallow burial. Discharge of liquid wastes to seepage pits was predicated on knowledge of the rate and direction of movement from the pits, and a rather good knowledge of the characteristics of the waste itself. If these wastes were not discharged to low permeability formations, there would be little chance for significant absorption. Low-level wastes disposed to shallow burial were assumed to be relatively nonleachable, but their specific characteristics were not (and are not) known. As detailed in the introduction to this section, there is only incomplete knowledge of the radionuclide content, to say nothing of the complex chemistry. Detailed studies of the trench water at Maxey Flats has revealed radioactive and nonradioactive chemistries that almost defy characterization (Columbo and Weiss, 1978). Further, the burial sites were, as mentioned, located in relatively impermeable material. There seems to have been an implicit assumption that the general strategy applied to liquid waste disposal via seepage pits would then apply to these solid wastes. Additionally, the "solid" nature of the waste, and the low permeability of the burial medium was taken as an advantage. It seems to have worked to our detriment, simply confounding our understanding of the environmental releases from these sites.

A second fallacy, which still exists, derives from the very term "low level." Initially, it was assumed that these wastes were in fact low-activity wastes, of no particular consequence. One of the most recent statements of the waste management problem, the IRG report (IRG, 1979), defines low level waste as

Low Level Wastes (LLW) contain less than ten nanocuries of transuranic containments (sic) *per gram of material, or they may be free of transuranic contaminants, require little or no shielding, have low, but potentially hazardous, concentration or quantities of radionuclides.*

Examination of Table XXVII shows that this is an inadequate definition and worse, an inadequate understanding of the scope of the low-level waste problem. As an extreme example, ion exchange resins from the cleanup at Three Mile Island may run as high as 53 000 Ci/m^3 (Carter, 1980). Yet this waste is presently considered to be low-level waste because it is, by definition, neither high-level nor TRU waste. Clearly, the broad application of near-surface burial to such a wide range of waste materials is inappropriate at best.

Finally, there is a conflict between two stated goals of waste disposal. It is desired to provide long-term protection against waste exposure by human or erosive intrusion. One means of providing this protection is by permitting gradual dissemination of the contaminants through the surrounding geologic material. However, that approach defies the objective of providing maximum containment of the contaminants in the waste trenches, to minimize environmental releases. Further, the continual propagation of "complete containment" burial sites can lead to an insufferable problem by exhausting the land area available for disposal.

(3) Long-Term Adequacy. There is a third arena in which the adequacy of shallow land burial should be examined. The Resource Conservation and Recovery Act (RCRA) is legislation aimed at reducing the amount of resources discarded in hazardous waste materials. The bill specifically excludes radioactive waste from its purvue. The effect of the legislation, in the long term, will be to substantially reduce the volumes of hazardous waste disposed of to land burial, evaporation ponds, etc.

There is no comparable legislation for radioactive waste. Current radioactive waste may appear to have little resource value. As decommissioning of major power plants gets underway, large quantities of metal

and building material will be headed for land burial—material that might be a resource if decontaminated. However, independent of the resource value of the waste material, there is another resource that is consumed by land burial—the land itself.

The availability of "good" sites for disposal is shrinking rapidly because of competition from other sources for those sites. Sites suitable for successful radioactive waste disposal are also good for nonradioactive waste disposal, whether hazardous, municipal, or otherwise. Volumes of waste are growing steadily as the population to expands, using energy and resources at an ever increasing rate. Several estimates have been made of the expected future volumes of radioactive waste. A report by the Interagency Review Group (IRG, 1979) presents projected volumes under two different sets of assumptions related to power growth, waste treatment options, etc.

The data illustrate that there is an expected doubling in the land required about every 10 yr for both cases. This presents no serious cause for concern on a time scale of 20 to 30 yr. However, the doubling rate points out that in 100 yr, at that rate of increase, we will require a thousand times as much land as we now have committed to land burial, and in 200 yr the increase is over a million.

Although there is probably little validity in projecting a decade or two of growth over a period of 200 yr, it does point up the need to reduce the rate at which we commit valuable land to the sole purpose of burying our unwanted material.

This generation rate can be reduced through several approaches, including reduced generation volumes, volume reduction of generated waste, treatment to remove contaminants (a form of waste treatment) or various schemes related to *in situ* treatment to reduce or remove the hazardous properties.* As the hazardous nature of radioactivity cannot be reduced except by natural decay, our only option for treatment, before and after disposal, may be to remove, and thereby concentrate, the radioactive contaminants for some alternative disposal method.

d. Improvements to Shallow Land Burial. A reasonably clear picture is emerging regarding the technical difficulties with current burial practice (Deichman, 1979). Current LLW programs are focused on many aspects of those problems; it is premature to judge the

———————————————
*E. B. Fowler, Informal Progress Report to NRC, April 1979.

likely success in the coming years. The work is focusing on improved waste forms, on better site selection criteria, and on engineering improvements to the site to restrict radionuclide releases. Less reliance will be placed on the site environment for containment, thus reducing the need for a thorough understanding of the potential migration processes and pathways.

Major modifications of the regulatory framework governing low level waste disposal are in progress. Until now, all commercial burial sites in NRC agreement states were regulated by the state without any nationally standardized criteria. The NRC is now proposing such criteria (10 CFR 61). Additionally, the Environmental Protection Agency is proposing criteria relating to acceptable releases from shallow burial sites, which will be of value in determining the acceptability of their performance.

6. Summary

As initially applied, shallow land burial was expected to produce more control over radioactive waste than conventional landfill practices of the time. Migration and environmental release of contaminants was expected, at least implicitly, but little effort was devoted to precise quantitative prediction of the rates of release. Primarily, the site characteristics were expected to restrict the release to some acceptably low level.

By the late 1960s and early 1970s, sufficient data regarding site performance had been accumulated to show environmental releases that could not be readily explained by state-of-the-art knowledge. Although releases were below those indicative of any significant public health hazard, these releases were not considered "acceptably low." Our knowledge of the environmental transport processes was not sufficient to allow a clear definition of the release and transport occurring at these sites.

Along with an increased awareness of active environmental releases, increased attention has been focused on long-term site control issues and protection of future users of the site. Site use restrictions and radionuclide burial concentration limits are being proposed, but are not yet adopted (with the exception of the 10 nCi/g limit on burial of TRU waste).

Improvements in shallow burial practice now under development pertain primarily to increased containment of the contaminants in the waste pit and less reliance on the site environment to provide decay storage.

7. Conclusions

The adequacy of shallow burial as a control technology for disposal of low-level waste must be measured in two separate ways—magnitude of releases compared with existing or proposed standards and long-term suitability as a "disposal" method. This evaluation suggests:

• No standards for environmental releases currently exist. Proposed standards are phrased in terms of expected dose to man, but leave open the means for calculating these doses. In any event, our knowledge of most existing sites is not adequate to quantitatively describe the release process.

• Resolution of questions regarding long-term adequacy requires knowledge of radionuclide concentrations in the waste, general acceptance of a prescribed time during which site control will be maintained, and general acceptance of the methods for calculating human exposure after site control is terminated. These requirements will be difficult to fulfill.

• Improvements to shallow burial now under development focus on containment of contaminants in the trench, thereby aggravating the concerns about eventual intrusion into the waste.

• Solid waste contaminated with radionuclides or other toxic materials will continue to be produced in the future, probably at ever-increasing rates. The area of land required for essentially permanent storage of these materials through application of shallow land burial will continue to grow out of all reasonable bounds. It is mandatory that we begin to develop means for dramatically reducing the *volume* of contaminated material to allow more suitable, but proportionately more expensive, disposal alternatives. *In situ* waste treatment systems should be compared with other conventional waste treatment options. Treatment options should emphasize contaminant concentration and resource recovery rather than reduced mobilization and small changes in volume.

C. Terminal Isolation

1. Introduction and Background

By the late 1950s, the Atomic Energy Commission (AEC) began to search for a permanent disposal method for ever-increasing quantities of high level waste. Table XXXI shows waste inventories as of October 1, 1978. Programs to develop the necessary technology for solidification and immobilization of liquid wastes were initiated. In 1957 the National Academy of Sciences (NAS) issued a report (NAS, 1957) recommending that waste canisters be isolated in deep geologic formations, salt being the most suitable medium. In the mid-1960s, ORNL conducted a series of experimental tests in several abandoned salt mines near Lyons, Kansas, to develop disposal technology in bedded salt, and, in 1971, the AEC requested authorization to construct a pilot-plant repository at Lyons. The State of Kansas objected on the ground that previous mining and exploration activities in the area compromised the geologic integrity of the site. The project was abandoned.

In 1974, SNL began a program to develop a Waste Isolation Pilot Plant (WIPP) in the Los Médaños area of southeastern New Mexico. The facility would receive defense transuranic waste (TRU) and provide experimental test capability for HLW.

In 1975 the newly formed Energy Research and Development Administration (ERDA) announced that a disposal facility could be available in time to meet the country's needs. The Office of Waste Isolation (OWI) was formed and charged with the responsibility of identifying potential repository sites in salt, and to investigate the potential of igneous and argillaceous rocks as disposal media. In 1977, two of OWI's efforts were reassigned as independent specific site evaluation projects. These are the Basalt Waste Isolation Program (BWIP) at Hanford and the Nevada Nuclear Waste Storage Investigations Project (NNWSI) at the Nevada Test Site (NTS). The following year OWI was transferred to the Office of Nuclear Waste Isolation (ONWI).

Current waste management practices consist of storage of spent commercial reactor fuel in water basins, reprocessing of spent government-owned reactor fuel, storage of high-level liquid wastes in above- and below-ground tanks, retrievable storage (either above or below ground) of transuranic wastes, and the shallow land burial of low-level wastes. Present concepts call for the disposal of high-level and transuranic wastes (and possibly spent fuel elements) in Federal geologic repositories.

2. National Waste Terminal Storage Program

a. General. The National Waste Terminal Storage (NWTS) program provides the research and development activities to identify repository sites and to develop

TABLE XXXI

INVENTORIES OF DEFENSE AND COMMERCIAL[a]
NUCLEAR WASTE VOLUMES

	Volume (10^3 m^3)			
	Defense		Commercial	
	Current[b]	Projected to Year 2000	Current[c]	Projected to Year 2000
High-Level Waste				
Solid	182	250 [d]	2	37
Liquid	101	70 [d]	2.3	2.3
Transuranic Waste	311	500-3350 [e]	0	6

[a]Data from Waste Management Summary Document, DOE/NE-008 (1980).
[b]As of October 1, 1978.
[c]Commercial solid HLW reflects spent fuel. Currently there are about 4000 metric tons of reactor spent fuel being stored in reactor basins. There could be as much as 95 000 metric tons of spent fuel discharged by the year 2000. If the spent fuel were reduced and stored as HLW, the volume would be reduced by a factor of about 10.
[d]Maximum volume case assumed.
[e]Range depends on extent of decontamination and decommissioning (D&D) activities.

the technology for repository design, licensing, and operation. The repository will be designed to accept spent fuel, HLW, and TRU waste. Efforts are directed primarily toward geologic disposal of spent fuel and the development of spent fuel packaging facilities. Potential repository sites will be characterized in several different geologic environments, from which one may be chosen as the first regional repository. The first facility is anticipated to be operational by the mid-1990s. Figure 6 shows the organization of the NWTS program.

b. Office of Nuclear Waste Isolation. That part of the NWTS program concerned with mined repositories in deep geologic formations is managed by ONWI. This organization is responsible for developing the generic technology for repositories in both salt and nonsalt geologic environments and for conducting characterization activities on non-DOE lands. Areas of interest containing salt formations are (1) the interior Gulf Coast salt domes of Louisiana, Texas, and Mississippi, (2) the Permian Basin in Texas and New Mexico, (3) the Paradox Basin in Utah, and (4) the Salina Basin in Michigan, Ohio, and New York. Other potential host rocks of interest include granitic crystalline rocks, argillites (shale), basalts, and tuffs.

Site selection strategy for a geologic repository is based on a three-stage identification process of regional, areal, and site-specific studies. At each stage all available data will be reviewed in an attempt to ensure that each area has been adequately investigated at that level and to identify deficiencies in the data base. Any site may be rejected at any point in the investigation procedure. New screening methods, based on multiple barrier (combined engineered and natural migration barriers) geohydrologic principles are being developed by ONWI. Candidate sites in salt formations are expected to be identified by 1985 and sites in nonsalt formations 3 or 4 yr later.

Generic technology development programs under the direction of ONWI include thermal/mechanical analyses and modeling, waste/rock interaction studies, penetration sealing, risk assessment, and geologic data base management. Spent fuel packaging studies are being conducted at the Engine Maintenance Assembly/Disassembly (EMAD) facility at NTS.

In situ thermal testing programs are being conducted in salt at Avery Island, Louisiana, and in a cooperative US/Swedish venture in granite at Stripa, Sweden. Radioactive *in situ* testing in salt is being planned.

Facility design and development is currently directed toward optimization and pre-Title I design studies for a repository in dome salt.

c. Basalt Waste Isolation Program. This program is responsible for determining the suitability of basalt at the
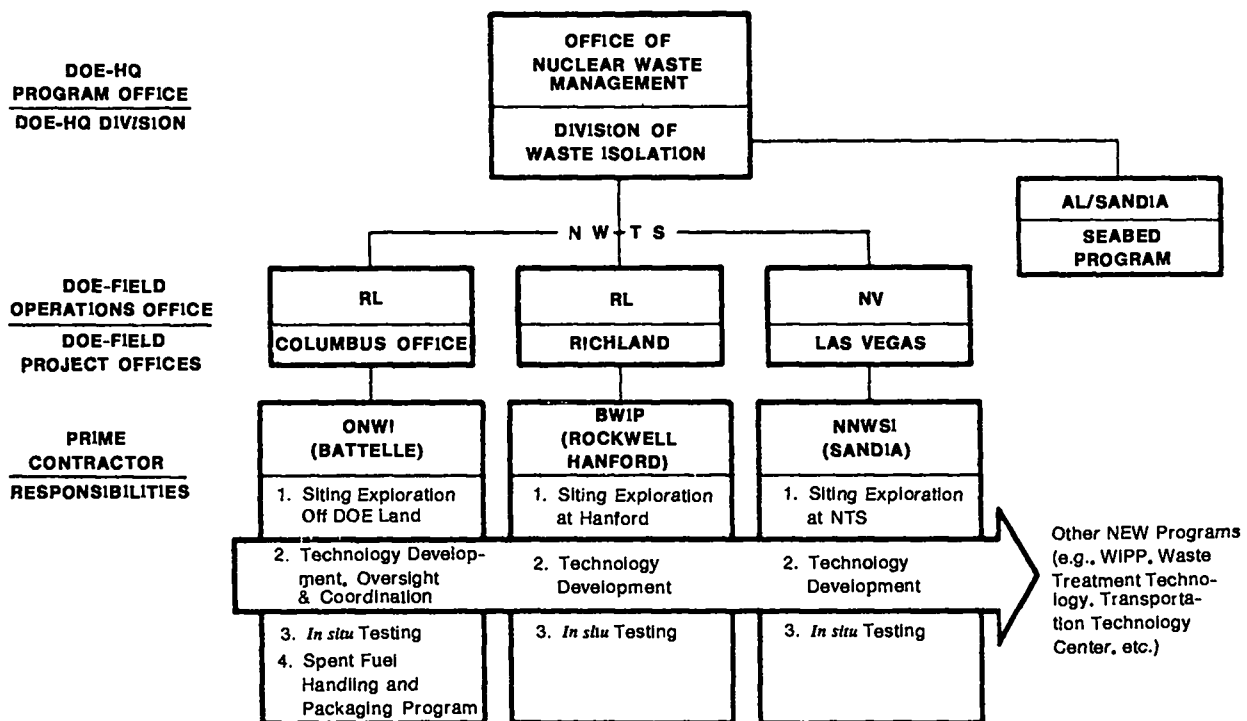
Fig. 6. NWTS Organization.

Hanford Reservation as a repository medium. Basalt-specific multiple barrier and penetration sealing studies are underway. Hot cell multiple barrier studies are planned for the near future. The Near Surface Test Facility (NSTF) at the Rockwell Hanford Operations is the site of rock mechanical and thermal testing. Nuclear waste loading is scheduled to begin in 1982, using spent fuel prepared at the EMAD facility at NTS. Testing with a canister of vitrified HLW is also being planned. The conceptual design for a repository in basalt is due to be completed in 1981.

d. Nevada Nuclear Waste Storage Investigations Project. The NNWSI is responsible for determining the suitability of the Nevada Test Site as a potential repository site. Several sites have been investigated. They are (1) Yucca Mountain (tuff), (2) Wahmonie Stock (granite), (3) Calico Hills (granite and shale), and (4) Syncline Ridge (argillite). Work on all the sites except Yucca Mountain has been discontinued because the structural integrity of the sites has become suspect. Exploratory drilling is now underway at Yucca Mountain, but a potential repository site will probably not be recommended before 1984. If the Yucca Mountain site is disqualified, investigation of the four southern counties

of Nevada will begin after consultation with state officials.

Seismic wave (both natural and induced) investigations and radionuclide transport studies in tuff, granite, and shale are being conducted in conjunction with ONWI programs.

The Climax Spent Fuel Facility at NTS is in operation. Five-year tests using encapsulated spent fuel are designed to support the NSTF at Hanford and to simulate repository operation in granite. The spent fuel was characterized and packaged at the EMAD facility.

In situ thermal and permeability testing at the 425-m-deep Piledriver facility (granite) was successfully completed in 1978. Thermal testing in shale was conducted in 1979 and scheduled to continue through 1980.

Engineering studies have been initiated for an at-depth test facility at a candidate site, if one is chosen at NTS.

e. Waste Isolation Pilot Plant. Characterization of the WIPP site in southeastern New Mexico was begun in 1974 by SNL. The facility was envisioned to receive defense transuranic waste and to provide experimental test capability for HLW. Following the recommendations of the Interagency Review Group (IRG, 1979), President Carter cancelled the WIPP and the site

60

consideration responsibility was transferred to ONWI. In January 1981, the WIPP was restored to the Defense Program and site development is expected to proceed as originally planned (Federal Register, 1981). A geologic characterization report, an environmental impact statement, and a draft safety analyis report for the site have been issued (Powers et al., 1978; DOE/EIS-0026, 1980; DOE/NE-0008, 1980). Although this site is no longer available for consideration as a commercial spent fuel or HLW repository, site validation studies and experimental programs at the WIPP should provide valuable information on repository performance in bedded salt.

f. Seabed Disposal Program. The purpose of the Seabed Disposal Program is to evaluate the feasibility and develop the technology to dispose of HLW and spent fuel in geologically stable formations on the ocean floor. The main objectives through 1983 are to collect data to determine the feasibility of a seabed repository and to develop the capability to assess foreign ocean disposal programs. Current efforts are directed toward (1) assessing potential sites in the north Pacific and north Atlantic, (2) determining geochemical and geophysical properties of deep sea sediments and the underlying basalt, (3) determining thermal and sorption characteristics of red clay sediments, (4) characterizing deep sea biota, (5) designing and testing emplacement devices, (6) developing predictive models for waste isolation, and (7) identifying the requisite engineering technology.

3. Mined Geologic Repositories

a. General. The overall objective of the National Waste Terminal Storage program is to provide a repository in which HLW, spent fuel, and TRU wastes will be permanently isolated from the biosphere. Recommendations by the NAS prompted a research and development program to investigate the feasibility of waste disposal in deep geologic formations. Studies were initially directed toward salt and have only recently been expanded to include regions underlain by nonevaporites as well. Because the existing data base for mined cavities (especially salt) is considerably larger than for other disposal alternatives (DOE/EIS-0046-D, 1979; DOE/NE-0008, 1980), most of the current emphasis is placed there. It is expected that the first regional repository will be selected by 1985 and operational by the mid-1990s, pending approval under the National Environmental Policy Act (NEPA). The programmatic approach integrates geologic exploration, *in situ* testing, repository design, development of licensing and operation criteria, and waste packaging technology development.

b. Technical Issues. The most important factors relevant to geologic disposal are depth and geometry of the emplacement zone, physical and chemical properties of the host rock, regional and local tectonic stability, hydrologic properties of the host rock and surrounding terrain, natural resource potential, and the multiple barrier concept. The capability of a potential site or medium to meet the requirements discussed below is the object of continuing research and development programs under NWTS direction.

(1) Depth of the Emplacement Zone. The overburden must be thick enough to prevent the repository from being breached by long-term surface erosion processes. Chemically and mechanically weathered material may be transported away from the site by water, wind, ice, and gravity. Thus, climatic conditions (both present and predicted future climates) and resistance to erosion of the overlying rock must be carefully considered in repository siting and design. Emplacement at a depth of about 600 m is generally considered to be adequate. The depth to the proposed emplacement zone at the WIPP site is 600 m and the number seems to be accepted, even though it may be arbitrary for other locations. Recently, depths ranging from 300 to 900 m have been proposed (DOE/EIS-0046-D, 1979; DOE/NE-0008, 1980). An adequate depth of emplacement will be a function of climate, rock erodability, and hydrology and geology of the host rock at different depths and, therefore, subject to local conditions.

(2) Geometry of the Emplacement Zone. The host rock must be of sufficient thickness and lateral extent to contain an economically reasonable volume of waste. The disposal medium must be large enough to ensure that radionuclides migrating away from the disposal site will not escape the preferred host rock before they have decayed to acceptable levels.

(3) Rock Properties. The properties of the host rock should be such that long-term containment of waste materials can be achieved. Desirable thermal characteristics would be low thermal expansion to avoid fracturing or weakening the rock and high thermal conductivity to minimize temperatures, rock deforma-

tion, and adverse chemical reactions (DOE/EIS-0046-D, 1979). Table XXXII shows the projected amount of heat released and activity levels from spent fuel canisters in storage.

Chemical reactions between the host rock and any foreign object placed in it are unavoidable. A number of typical reactions have been identified, which may be favorable or unfavorable. Reactions that form insoluble compounds or those containing water of crystallization (thus reducing the amount of free water) are favorable. Sorption of radionuclides onto rock materials is highly favorable. Unfavorable reactions include those between the waste canisters and corrosive fluids, such as brine, resulting in radionuclide release. In a thermal field, brine inclusion in salt may migrate toward the heat source.

The water content, porosity, and permeability of the rock should be low to minimize chemical reactions and radionuclide transport.

(4) Tectonic Stability. The regional and local tectonic regimes should be sufficiently stable that the repository will not be damaged by earthquakes or faulting. No site criteria now exist regarding the distance that a repository must be from potentially active faults. Induced faulting or fracturing could increase permeability and may adversely affect other rock properties,

such as heat dissipation and chemical reaction rates. Areas subject to geologically rapid elevation changes would be unfavorable. Regional uplift would increase erosion rates. Downwarping and associated sedimentation could increase temperatures and pressures. Both uplift and subsidence may modify the climate and hydrology at the site. The site must be located in an area free of igneous activity during the Pleistocene Epoch (the last 1.5 to 2.0 million years) (10 CFR 60).

(5) Hydraulic Properties. Because the primary mechanism for radionuclide transport is generally assumed to be by water, the fluid characteristics of the disposal medium are of utmost concern. All rocks contain some water and are permeable in varying degrees. The potential for fluid flow through the host rock depends on the water content, porosity, permeability, and hydraulic conductivity (the tendency for water to flow, driven by a pressure gradient). Porosity is the volume of intergranular pore space (plus secondary features, such as fractures). However, a high porosity does not necessarily imply fluid flow. Flow cannot occur if the pore spaces are not connected. Shale, a potential host rock, is composed of clay-size particles and typically has a high porosity, but its permeability is low because the pores are not connected. In addition, pore

TABLE XXXII

HEAT AND ACTIVITY IN SPENT FUEL CANISTERS[a,b]

| | Heat Generated (W) | | | | |
| | | Years After 1975 | | | |
	Year 2050	500	1000	100 000	1 000 000
Fission and Activation Products	1.6×10^6	4.5×10^4	1.9×10^4	1.1×10^4	5.4×10^2
Actinides and Daughters	7.1×10^7	3.4×10^7	3.6×10^6	1.4×10^5	
	Activity (Ci)				
Fission and Activation Products	5.0×10^{10}	2.7×10^7	8.2×10^6	5.5×10^6	1.1×10^6
Actinides and Daughters	9.9×10^9	1.0×10^9	6.0×10^8	1.4×10^7	

[a]Data from DOE/EIS-0046-D, April 1979.
[b]Based on an estimated 1.28×10^6 waste canister containing 3.80×10^5 metric tons of heavy metal.

spaces in shales are generally so small that capillary forces hold the water bound electrostatically (high matric potential). A rock may have both a high water content and a high permeability, but if the water is not subject to a pressure differential, flow may not occur. Thus, careful hydrologic investigations must be conducted at candidate sites to evaluate the local flow regime. Regional hydrology must be studied to determine its effects on local conditions. Changes in recharge or discharge, caused by changes in climate or regional water use, may modify the local flow regime (DOE/EIS-0046-D, 1979).

At excessive temperatures, liquid water may vaporize and migrate by diffusion. The temperature at which this occurs depends on the pressure (depth of burial) and the amount of material dissolved in the water.

(6) Natural Resources. The occurrence of natural resources (mineral, water, gas, geothermal potential, etc.) within 100 km of the site must be considered (10 CFR 60). Past, present, and future exploration or development activities could severely compromise site integrity or effect radionuclide release to the biosphere. Care must be exercised in resource evaluation to account for materials that may become valuable resources in the future.

(7) Multiple Barriers. The multiple barrier concept of waste isolation uses several engineered and natural migration barriers that, in combination, should ensure adequate protection. The system would typically include the waste form, packaging, and the disposal medium and surrounding area. The particular combination used at any one site will depend on the specific geologic environment and repository characteristics. An appropriate multiple barrier system will be identified by modeling and *in situ* testing.

c. Data Base. We do not as yet have a complete understanding of all the factors described above. Especially lacking is information regarding the response of potential disposal media to long-term temperature, stress, and irradiation (DOE/EIS-0046-D, 1979). These deficiencies can be mitigated by generic modeling, laboratory experiments, and field verification. Field verification is critical because *in situ* tests often yield results that differ significantly from model or laboratory predictions.

At the elevated temperatures and pressures of a repository situation, many different waste/rock and canister/rock chemical reactions will take place. The particular reactions that will be favored will depend heavily on the physical and hydrologic properties of the host rock and will be different for different media. Granite and basalt are generally assumed to be less chemically reactive than salt and shale, but both of these are relatively new areas of study and more data are being collected.

Physical responses of the host rock to repository conditions must be studied *in situ.* Unpredicted excessive thermal expansion may cause fracturing of the rock, increasing permeability and perhaps allowing fluids to enter or escape from the repository. In addition, seismically induced fracturing or faulting could modify the thermal regime in the repository, resulting in less than optimum heat dissipation, which, in turn, could create more unexpected problems.

Some degree of uncertainty will always be present, but adequate modeling and field verification should reduce it to a minimum.

Programs to investigate techniques to adequately seal boreholes, mine access shafts, and ventilation shafts seem to be underdeveloped. Boreholes and shafts serve as potential conduits for moisture into the repository and must be adequately sealed. Generic penetration sealing programs are underway but specific seal designs must be determined on a site-by-site basis.

A potential problem area that is not discussed in the EIS's and planning documents is that regarding the effects of biological activity in the repository. There seems to be little built-in concern for the case of bacteria entering the mine during excavation and waste-loading operations. Biological activity can produce gases (for example, H_2S, CO_2) which, after sealing, may be sufficient to increase pressures (Barnhart et al., 1980). The potential effects of biological activity in a repository have not attracted much attention. Perhaps this aspect of repository development programs is being considered, but it is not obvious.

d. Disposal in Salt. Salt deposits occur as either laterally extensive bedded deposits or as salt domes and are formed by precipitation of halide (NaCl) by evaporation of seawater. Bedded salt formations often contain interbedded shales and mudstones, carbonates (limestones and dolomites), and other sedimentary rocks. Impurities in salt are anhydrite ($CaSO_4$), calcium and magnesium chlorides, and sodium and magnesium sulfates (Bates, 1969). In deep sedimentary basins, such as along the Gulf Coast, the weight of the overlying rock column may force deep salt layers to move upward through weak points in the overlying strata. The salt flows upward as a solid, a process known as diapirism.

Dome salts are generally more homogeneous and pure than bedded salt (DOE/EIS-0046-D, 1979), but many are overlain by a caprock of anhydrite, gypsum, and impure limestone. Mineral resources associated with salt domes include oil and gas trapped in porous sandstones around their flanks, in caprock limestones, and in domed overlying strata. Major sulfur deposits occur in caprock limestones. Attempts to mine caprock gypsum have so far proved uneconomical (Bates, 1969). Finally, salt itself is a resource, but its widespread occurrence suggests that this should not be a factor in site selection.

Salt has a high thermal conductivity and readily dissipates heat. Low cohesive strength allows it to flow plastically under pressure (sometimes referred to as "creep"), giving rise to its "self-sealing" capacity. However, special techniques are required to keep tunnels open and maximum room size is less than for stronger media (Table XXXIII). Bedded salt deposits are generally quite extensive, but lateral continuity of favorable beds may be variable. If undisturbed, salt beds are often considered to be essentially impermeable, but inclusions of brine occur and interbedded layers of nonsalt rocks may contain water (Bates, 1969). Brine inclusions may migrate up thermal gradients toward waste canisters. Structurally, salt formations are often jointed and may contain extensive inclusions of anhydrite, limestone, dolomite, and other foreign rock types.

We have found no reliable information regarding techniques to locate inclusions of foreign material and brine in salt. Thus, the homogeneity of a repository in salt would probably not be determined with any real confidence until the mine is excavated.

TABLE XXXIII

MINING REQUIREMENTS FOR
GEOLOGIC REPOSITORY[a,b]
(MT × 10^6)

	Volume Excavated	Room Backfill	Total Backfill
Salt[c]	30	14	17
Granite	77	29	38
Shale	35	15	21
Basalt	90	32	46

[a]Data from DOE/EIS-0046-D (1979).
[b]Assuming repository area of 8 km².
[c]Values for salt may be lower, depending on the strength of the overlying rock.

Other areas regarding salt that need more development are ion exchange rates, response to irradiation, and chemical reactions between the media and waste materials (DOE/EIS-0046-D, 1979). These deficiencies are being addressed by ONWI research programs.

e. Disposal in Granite. Granite is a generic term applied to several specific igneous rocks having fairly similar characteristics. However, chemical variations can have a significant effect on the physiochemical properties relevant to waste disposal. Granite is formed by the slow crystallization of molten rock material at a depth of several kilometers. These granitic plutons are generally homogeneous with a dense, equigranular, medium- to coarse-grained texture. They are typically jointed in three perpendicular directions, the two vertical joint sets being the most prominent.

The moisture content of granite is usually a few per cent with most of it occurring as water of crystallization in hydrous minerals. Porosity is very low, except in fractures. Away from fractures, moisture movement is by intergranular diffusion. Both strength and rigidity are very high. Because of its high temperature of formation, granite is relatively resistant to thermal stress, but uncertainty exists with respect to some of the more hydrous minor minerals (DOE/EIS-0046-D, 1979). High temperatures may be capable of releasing bound water. The two most prevalent minerals, quartz and feldspar, are relatively resistant to weathering. However, feldspar may react with a weak acid to yield quartz and a clay mineral. Clays are highly regarded for their ion exchange capacity, but loss of strength may have an adverse effect. More information is being collected regarding total rock system/waste reactions (DOE/EIS-0046-D, 1979).

Mineralized veins (gold, silver, copper, etc.) rarely occur within granitic bodies (DOE/EIS-0046-D, 1979), but the surrounding rock is often altered by thermal and chemical effects associated with their intrusion (Park and MacDiarmid, 1970). Many common ore deposits are a direct result of granitic intrusion.

f. Disposal in Shale (Argillite). Argillite is derived from siltstone, claystone, or shale but exhibits a somewhat higher degree of induration than those rocks. In this report, shale and argillite are used synonymously. Shale is a clastic sedimentary rock formed by the compaction (and usually cementation) of clay-size particles, predominantly clay minerals but including a wide variety of other mineral fragments. These rocks are generally laminated and fissile, and interbedded with sandstones

and limestones. Because of the small particle size (<74 µm), porosity is high but permeability is low. Shales are often soft and are easily deformed under pressure, except when well cemented or metamorphosed. Special support systems (the technology is currently available) may be necessary to ensure retrievability.

The major constituent of argillites, clay, is well known for its high ion exchange capacity and therefore sorptive potential. Some clays will absorb water when wet, swelling substantially. This expansion may present problems in some shales during the operational and retrieval phases. Even if the shale is originally dry, leakage from an overlying aquifer may bring water into the repository. Completely sealing access and ventilation shafts may be difficult. Data base deficiencies, which include shale/waste reactions and the response of shale to thermal loading (DOE/EIS-0046-D, 1979), are being addressed by ONWI research programs.

g. Disposal in Basalt. Basalt is a very dense, fine-grained to glassy volcanic rock formed by the flow of Mg- and/or Fe-rich lava onto the earth's surface. The very high temperature of the lava results in a low-viscosity flow. Thus, indivdual flows are generally thin and widespread. However, repeated eruptions may build up a vast thickness of basalt, such as the Columbia River Plateau in the northwestern United States. In some areas basalt formations may be uniformly dense throughout their thickness, whereas in other areas broken "clinkery" layers and open lava tubes occur. Basalt flows are typically jointed, forming polygonal columns perpendicular to the cooling surface. Vugs, or cavities, sometimes occur within flows. Joints and vugs may contain weathering products or secondary minerals, such as zeolites.

Basalts exhibit very high strength, which is little affected by elevated temperatures, but expansion may occur (DOE/EIS-0046-D, 1979). Research programs conducted by the Basalt Waste Isolation Program are attempting to remedy data base deficiencies related to basalt/waste chemical reactions.

h. Disposal in Tuff. Tuff is a rock composed of indurated volcaniclastic material, ejected during explosive volcanic events. Eruptions often begin with the ejection of pumice, shards (fragments of silica glass), and volcanic ash, which are deposited downwind as air-fall deposits. These are followed by particle flows, energized by hot, expanding gases. As the ash flows come to rest and compact, glass shards in the core of the flow may

deform and stick to one another, a process known as welding. Ash flows that cool rapidly may remain glassy; those that cool slowly may crystallize to quartz, cristobalite, tridymite, and feldspars (the process of devitrification). Cooling typically results in the formation of joints and fractures. Escaping gases may deposit dissolved material in pore spaces, forming secondary vapor phase minerals. Interactions between the glass shards and ground water may result in the formation of secondary hydrous minerals, such as zeolites, or the dissolution and redeposition of silica in pore spaces (silicification). Weathering of feldspars may form mixed-layer clays and montmorillonite (an expanding clay). After initial deposition, flows may be reworked and redeposited by water or wind action. Most tuffs are silicic but a few are of basaltic composition. Because no basaltic tuffs are presently being investigated, "tuff" in this report means silicic tuff. When unaltered, they are typically greater than 80% shards and less than 20% crystals and rock fragments. Tuffs are common in the western United States, especially in the Southwest. Individual units may be several tens of kilometers in lateral extent and attain thicknesses of 3000 m (Johnstone and Wolfsberg, 1980).

Given the above description, it should be clear that the distinguishing characteristic of tuff is its extreme variability in physical and chemical properties. The following information is taken from the most recent status report of investigations at Yucca Mountain (Johnstone and Wolfsberg, 1980). The strength of tuff varies with degree of welding. Welded tuff can be up to three times stronger than nonwelded tuff, but laboratory experiments indicate that the strength of welded and partially welded tuffs at 200°C decreases by 30% from that at standard temperature. Unlike salt, granite, and basalt, the stiffness of tuff is anisotropic. Welded tuff is stiffest in a direction perpendicular to bedding, whereas nonwelded tuff exhibits maximum stiffness parallel to bedding. Water plays an important role in affecting thermal and mechanical properties. When dry, the compressive strength of welded tuff is 25% greater than when saturated. No information is available regarding the response of saturated tuff to irradiation. Recent studies suggest that the stability of mine openings during the operational phase may be affected by creep. More data in this area are being collected under the direction of the Nevada Nuclear Waste Storage Investigations Project.

The sorptive potential of tuff varies with lithology and the kind and abundance of minerals present. Nonwelded and poorly welded tuffs are more porous and less

resistant to weathering than are welded varieties. Some of the secondary minerals in nonwelded tuff, especially zeolites and clays, have high sorptive capacities. More effort needs to be directed toward understanding the chemical processes governing the sorption of radionuclides onto minerals in tuff.

Thermal conductivity in tuff depends on water content, porosity, mineralogy, and fracture spacing. Data are available for silica glass, quartz, cristobalite, feldspars, and zeolites, but none exist for mixed-layer clays and montmorillonite. The response of nonwelded tuff to thermal stress is complex, characterized by thermal contraction rather than expansion. This effect is presumably related to dehydration reactions in zeolites, hydrated glass, and clays. The behavior of tuff in elevated temperature fields may pose limitations on the thermal loading of a repository. This problem may be alleviated by spacing waste canisters in a wider array. More work is needed on near- and far-field thermal and thermomechanical modeling.

Characterization of tuff for terminal isolation purposes is a young program, and most of the current work is laboratory oriented. Extensive modeling and field verification data must be obtained before characterization of the tuff is complete.

i. Site Selection. The process of selecting a candidate site is directed by the Office of Nuclear Waste Isolation in cooperation with the (USGS). The primary objective is to identify specific sites with characteristics favorable to terminal isolation. As a result of the 1957 NAS report, salt has been studied extensively in this respect. Programs to investigate the potential of nonsalt geologies began in 1979 and are several years behind salt. Regions under investigation are shown in Fig. 7.

Site selection is a three-stage process, each stage becoming progressively more specific. Information obtained at each stage will be available to subsequent stages and should help to identify the uncertainties at each level. Stage I addresses the tectonic and hydrologic regime on a broad, nationwide scale, irrespective of any specific rock type, to identify favorable regions. This is accomplished by searching the literature, existing geologic exploration data, and satellite imagery data.

During Stage II, favorable regions identified in Stage I are broken down into study areas of about 10 000 km^2 (IRG, 1979; DOE/EIS-0046-D, 1979). Information on faulting, hydrology, seismicity, geologic structure, and stratigraphy is added to the data base. Geologic mapping, generic investigations of rock properties, and climatic studies are performed.

Fig. 7. Geologic repository study areas (after Waste Management Summary Document).

In-depth investigations of candidate sites (about 10 km²) are conducted during Stage III. Sites are identified in order of desirability. Activities include site-specific exploratory drilling and geophysical studies, the opening of test tunnels, *in situ* testing of rock properties, and hydrologic studies.

Investigations are underway to remedy important deficiencies in the existing data base, such as the response of the host rock to mechanical stress, elevated temperatures, and irradiation (DOE/EIS-0046-D, 1979).

j. In Situ Testing. In situ testing is a necessary part of the National Waste Terminal Storage Program. The geologic environment is a complex system in which many factors are interrelated in an extremely complicated manner. Some of these relationships are not well understood. Generic analyses involving computer codes, design assumptions, and laboratory experiments suggest that disposal of HLW in geologic formations is feasible, but these results must be validated under actual field conditions. More information is being obtained regarding the response of a host rock to various conditions of thermal, chemical, radiological, and physical stress. In addition, *in situ* testing is the only reliable way to provide data on optimum mining techniques, array spacing, borehole plugging, waste retrieval techniques, and canister and engineered barrier performance.

Various *in situ* testing programs are underway at Hanford, NTS, Avery Island, and Stripa, Sweden. A borehole plugging demonstration has begun at the WIPP site and the Colorado School of Mines is developing an experimental mine to explore mining techniques and site evaluation methodology in granitic rocks (DOE/NE-0008, 1980). The mine is expected to be used for thermal testing as well.

4. Waste Form Development

The multiple barrier concept of waste isolation provides for the combined effect of several engineered and natural containment barriers. One of the most important barriers is the waste form itself. On the recommendation of the IRG and for nonproliferation reasons, the President has decided to defer the reprocessing of commercially generated spent fuel (Report to the American Physical Society, 1978; IRG, 1979). HLW is liquid, an unacceptable waste form for terminal isolation. Interim storage of HLW and some TRU wastes in tanks of various designs has been disappointing. Leaks have occurred and public confidence has been strained.

Spent fuel is already in a ceramic form so the main concern is one of encapsulation. The primary option is to seal the fuel in canisters filled with an inert gas. Other techniques being developed include metal matrix fill, sand fill, and fill of other ceramic or glassy material (DOE/NE-0008, 1980). Investigations of various multiple packaging methods are underway. These programs are conducted by ONWI through the Engine Maintenance Assembly/Disassembly facility at NTS, with *in situ* testing at the Climax facility in granite and at the Near Surface Test Facility in basalt. Waste materials needs and evaluation are coordinated for the entire waste management program by the Waste Materials Steering Committee. The information below is referenced to the Commercial Waste EIS (DOE/EIS-0046-D, 1979), except where noted.

The production of a final waste form for HLW involves converting it to a solid. The major factors influencing the design of a preferred waste form are age and chemical composition of the waste, properties of the host rock, and the time period over which integrity of the waste and container must be maintained. Among the options for solidification of HLW are vitrification, calcine, supercalcine, glass ceramics, sintering, metal matrices, coated pellets, cermet, ion exchange media, and molecular stuffing.

a. Vitrification. This process simply converts the waste to a glass. Glass production is a fairly well developed technology, more advanced than the other alternatives. Glass is naturally unstable, however, and will spontaneously devitrify with time (NAS, 1978) resulting in slightly increased leach rates. Leach rates also increase with temperature. Recent studies suggest that some glasses containing alkali metals or alkaline earths may degrade when irradiated and exposed to even low moisture conditions (Hirsch, 1980). More data are being collected regarding the behavior of glass under repository conditions.

b. Calcine. Calcination produces a dry amorphous powder of waste oxides. Because of its high surface area, leach rates are high. Thermal conductivity is low and the powder may form some crystalline phases, releasing nitrates and water, which may decompose under irradiation and pressurize the container.

c. Supercalcine. This is a ceramic produced by spray calcining a solution of HLW, Ca, Sr, Al, and stabilized silica solution, which is then heat treated to form

crystalline phases. Leach rates are similar to glass. Supercalcine technology is still in the laboratory phase.

d. Glass Ceramics. These are a fine-grained crystalline material in a glass matrix, produced by controlled crystallization of waste glass. Glass ceramics are thermally stable and have leach rates similar to glass. The technology is new and requires several more years of work.

e. Sintered Products. Sintered products are produced by the sintering of calcine and glass-forming materials. They are fairly stable and exhibit leach rates similar to glass. The process is still in the laboratory stage and investigations are underway regarding their stability in hydrothermal and radiation fields.

f. Metal Matrices. These are various combinations of calcine, glassy beads, or ceramic pellets in a sintered metal matrix. Physical properties vary, depending on the metal used, but their thermal conductivity is high. Leach rates depend on the waste form used. Continued studies are directed toward the characteristics of each type and response to repository conditions.

g. Coated Pellets. The process involves coating pellets of calcine, supercalcine, or a sintered product with a leach-resistant material, then incorporating them into a metal matrix to increase thermal conductivity and strength. The procedure is still in the laboratory and the various combinations need more in-depth investigation.

h. Cermet. Cermet is a metal matrix containing small particles of fission product oxides. Cermet technology is in the laboratory stage of development and extensive work is continuing.

i. Ion Exchange Media. The product is a ceramic of hot-pressed ion exchange media. Leach rates are low and the thermal conductivity exceeds that of glass.

There is a problem with the product cracking under thermal stress. Production has reached the bench scale stage, but considerably more characterization is needed.

j. Molecular Stuffing. The molecular stuffing process involves depositing fission products in a "glass sponge," which is then sintered, yielding a product with highly concentrated waste at the center and waste-free glass on the outside. It is hoped that the extra glass layer will reduce leaching of the waste. Development is currently in the laboratory phase.

Although many alternative waste forms are under investigation, most of them are still in the early stages of development. It may be several years before a product with adequately favorable characteristics is forthcoming. A recent NAS study (NAS, 1978) concluded that glass may be suitable for demonstration purposes. However, work on other alternatives should continue.

5. Predictive Geology

One potential problem, which is not obvious, lies in the area of predictive geology. Geologic systems have not always performed as predicted (Dowding, 1975; Lambe, 1973; Meigh, 1976). A more appropriate term would be "probabilistic geology," because the prediction is really a probability estimate (usually expressed as events/yr). The probability estimate is an average rate of occurrence of an event (earthquake, volcanic eruption, etc.) over a given period of past time, derived from an empirical data base. Because the estimate is grounded in observations of past occurrences and experimental data, it can be extrapolated into the future only by assuming that the observed past conditions will not change, or at least change in a regular manner. As our knowledge of geohydrologic and climatologic systems increases, so does our ability to make this assumption with increasing confidence, but we cannot as yet do so with complete assurance. The assumption that past conditions can be extrapolated into the future depends heavily on the quality and extent of the data base.

Even if we feel confident that our estimate can be used in a predictive sense, how long will it remain valid? This question is potentially dangerous because there is no fixed answer. The validity period of a probability estimate depends on the quality of the data base and on the nature of the phenomenon. For cyclic phenomena, observation of a few cycles is generally sufficient to define the cycle, and the estimate should remain valid until the factors controlling the cycle change. For non-cyclic phenomena, such as earthquakes and volcanism, an extensive, high-quality data base is required before a statistical rate of occurrence can be calculated.

After a repository is decommissioned and the waste is no longer retrievable, the only thing we have with which to reassure ourselves of permanent isolation is our confidence in a set of probability estimates for geologic and hydrologic stability. Long-term performance expectations of waste forms, packages, and multiple barriers are all probability estimates, derived from data bases of, it is hoped, high quality. Geohydrologic stability depends

heavily on climatic stability, and many aspects of long-term climatological forecasting are hotly debated (Abeele, 1980). The probable effects of increasing CO_2 in the atmosphere (greenhouse effect) is the focal point of major controversy.

We feel that the nature and limitations of probability estimates are not fully appreciated. The estimates are often accepted as true predictions of the future rather than mathematical statements of past trends and observations. The hazard presented by probabilistic geology is the substitution of the word "prediction" for "estimate." The EDPs and EISs do not address this issue, and we are concerned that inappropriate application of probability estimates my lead to a false sense of security regarding the long-term performance of a geologic repository.

6. Seabed Disposal

Although the National Waste Terminal Storage program is directed toward geologic disposal on continents, disposal of wastes in sediments on the ocean floor is also being studied as an alternative near-term option. The Seabed Disposal Program, initiated in 1973 and directed by Sandia National Laboratory, is aimed at determining the feasibility and developing the technology to accomplish this task. A secondary objective of the program is to develop and maintain the capability to assess foreign ocean disposal programs. The program is divided into four phases:

1. Determine the technical and environmental feasibility of seabed disposal based on historical data—completed in 1976.

2. Determine the technical and environmental feasibility from recently collected oceanographic and effects data—to be completed in 1985-1987.

3. Assessment of engineering feasibility by 1980-1993.

4. Demonstration of capability after 1995.

The following information is drawn from the Commercial Waste EIS (DOE/EIS-0046-D, 1979).

The reference disposal system is the emplacement of appropriately treated waste or spent fuel in specially designed containers in red clay sediments in the middle of an oceanic tectonic plate, under the hub of a surface gyre (rotating circular ocean current). These areas are known to have been tectonically stable for up to 70 million years. They are covered by sediments to a depth of 50 to 100 m, biologically unproductive, and have weak and variable bottom currents. Known natural resources are limited to manganese nodules, rich in Mn, Co, Ni, and Cu. The usual mode of occurrence of nodules is in areas of higher biotic activity than the reference sites and because there are large reserves of low-grade ore on land, mining in the study areas seems unlikely.

The sediments are very fine grained (about 2 μm) clays. They are water saturated and have low permeabilities. The only known driving force for water movement through the sediments is compaction, which occurs at a rate of 1 to 3 mm/10^3 yr. Permeability data suggest that the pore water velocity is in the same range as the rate of sedimentation (about 1 mm/10^6 yr). Radionuclide migration away from a waste canister would be by diffusion and advection. The primary mechanisms for sorption and migration must be identified before adequate models can be developed.

The response of the sediments to thermal loading must be addressed. Heat generated by the waste may cause a volume change in the sediments by which the canister may sink or the heated sediments may rise and mix with the overlying water column. Detailed information about the water column overlying the reference sites is needed to determine its potential as an isolation barrier.

Emplacement methods under investigation include penetrometers, deep-sea drilling, and trenching.

a. Penetrometer. The waste canister is shaped like a projectile, which would be dropped from a ship, free fall to the bottom, and bury itself in the sediments. The sediments are capable of deforming plastically and should flow in behind the penetrometer, sealing the hole. A variation on this technique is to lower the canister part way with a winch before dropping it. More laboratory work and field testing are needed to determine the effects of impact on the canister and the sediments.

b. Drilling. This method calls for drilling a hole in the sediments with a ship-mounted drill rig. The technology is well developed and the hole can be drilled within a few meters of the target position. The time and cost required to drill the holes dictates that many canisters be placed in the same location.

c. Trenching. Trenching uses the same basic technology as is used to bury transoceanic cables or pipelines. The trencher is lowered to the bottom and dragged behind a ship, much in the manner of a plow.

69

The depth of emplacement is limited to only a few meters. In addition, there is currently no way to feed the waste canisters to the moving trencher.

One of the basic premises of repository design is that during an initial time period the waste will be retrievable. Retrieval of waste canisters from the ocean floor is possible with existing technology, but expensive.

Seabed disposal is a new technology and there is much to be learned. The program is now in Phase 2 and many of the problems are engineering difficulties, which should be rigorously addressed during Phase 3.

7. Conclusions

Generally, the envisioned program presents a well-coordinated project to solve the Nation's high-level and TRU waste repository needs in a logical manner. The idea of pursuing several approaches at the same time is excellent, as it avoids the "putting all the eggs in one basket" syndrome. The fact that any repository will be licensed by the NRC lends credibility to the entire project. Flexibility is inherent in the program, which is another excellent feature. The country is not 100% committed to terminally isolating spent fuel. This approach, terminally disposing of spent fuel, may well be in direct conflict with the stated Energy Independence Program so that the US may free itself of dependence on foreign oil. The amount of energy represented in the stored spent fuel may well be very significant in view of our foreign oil dependence.

The planned program, which envisions an initial period of investigating potential repository sites in various geologic media, as well as investigating alternative terminal isolation concepts, is well thought out. In addition, the initial stages of operation for the selected repository or concept will be carried out in a retrievable manner, which will allow one to "debug" the operating procedures before going into the terminal, irretrievable mode of operation.

The program, as presented in the Commercial Waste EDP, involves six major areas of concern regarding terminal isolation. These concerns are shown below.

1. Potential for containment and release of radionuclides at candidate geologic formations and sites.

2. Characterization of alternative isolation technology.

3. Health and safety requirements for the design and operation of a mined geologic repository.

4. State and public participation.

5. Intrusions by future generations into the repository.

6. Ecological and health effects of mining the repository.

Analyzing the program as stated in the EDP, we feel that these six major areas of concern have been adequately covered in depth and detail by DOE programs to assure the safety and public acceptance of a geologic repository. The state and public participation area is an important one in view of the past history in Kansas and New Mexico, and was addressed directly by President Carter in his radioactive waste program as announced on February 12, 1980 with the formation of the State Planning Council.

Area five, intrusions by future generations into the repository, may have a small problem with wording in the EDP. It is stated that *"...no known institutional mechanism could assure proper safeguards over the time period of hazard potential in the repository."* However, it goes on to state *"...that in the event that future generations determined to use the waste as a resource, a record of the kinds, quantities, exact locations and proper means of access should be developed."* This appears to be an institutional mechanism to identify in this record the kind, quantities, etc., and proper means of access. There appears to be a direct conflict in this area.

We suggest enhancing several areas of investigation pertaining to geologic repositories. These represent a unique technical analysis task, as the repository must perform in a manner consistent with its design goals over a long time period; approximately 1000 years for fission products and much greater if transuranics are present. The design and safety analyses etc. for a geologic repository are only as good as the assumptions used for possible future events. The areas of predictive geology and climatology are extremely important in identifying all potential events that may affect the repository in the future as input to any risk analysis or environmental impact. We feel that these are important to proper repository design and should be included as major areas of concern in addition to the six listed above.

Because of the extremely long time span for the proper performance of the repository and the use of predictive modeling in the design stages, we suggest that performance verification of the models used be an integral part of the retrievable mode. The retrievable mode of operation offers a time span, which may be hundreds of years, for verification of the models used, and also incorporation of any future technology that would significantly improve

70

the isolation characteristics or operation etc. of the repository.

It is too early in this program for us to fully evaluate the potential control technologies because the experimental results are not complete and the specific technologies have not yet been selected. We are somewhat concerned because we have not located any information about methods to recover from accidents or about post-operational monitoring. Perhaps it is too early in the program for these issues to be investigated in a meaningful manner. However, these issues must be addressed before actual operations will be allowed to start.

In regard to the question of whether a geologic repository is needed for the disposal of high-level and transuranic wastes, we consider the point to be moot. Although some can argue that the hazard of the waste in 1000 yr will be no greater than that of the original ore body, this argument does not seem appropriate. Both the Congress and the Administration have determined that geologic repositories are necessary. Further, this determination is in agreement with the majority of the international nuclear community. It has taken many years to obtain a consensus among the technical, governmental, and public communities, therefore, little would be gained by going back to the early arguments concerning HLW disposal.

D. Conclusions—Waste Disposal

Shallow land burial (SLB) was originally designed merely to provide more control over radioactive wastes than conventional landfill practices used at the time. Site characteristics were expected to restrict radionuclide releases but complete containment was not a primary goal. Releases did occur and, although they did not constitute a public health hazard, release and transport mechanisms were not well enough understood to maintain them below acceptable limits. More recently, increased attention has been directed toward long-term containment of waste radionuclides and protection for future users of the site. Site-use restrictions and radionuclide burial concentrations are being proposed but are not generally accepted.

The adequacy of SLB as an environmental control technology must be assessed on the basis of how releases compare with current and proposed standards and its long-term suitability as a disposal method. No standards for environmental releases currently exist. Proposed standards are phrased in terms of dose to man, but do not suggest the means for calculating these doses. In any

event, our knowledge of most existing sites is not sufficient to quantitatively describe release mechanisms. Ensuring long term adequacy of SLB as a disposal method depends on a knowledge of radionuclide concentrations in the waste, general acceptance of length of site control maintenance, and general acceptance of a method to calculate human exposure after site control is terminated. None of these three criteria seem to be met at the present time.

Improvements to SLB now under development seem to concentrate on two conflicting goals. First, SLB facilities are expected to provide long-term containment of waste radionuclides. By this criterion, SLB has clearly been inadequate because seven of the eleven major sites have allowed some radionuclide release. The second goal is to provide protection for future users of the site (intruders or reclaimers). The conflict is that improved containment merely increases the risk to the future user. The problem is further compounded by the lack of generally accepted standards against which to compare radionuclide releases.

If the current rate of waste production continues, the demand for more land for SLB will soon grow out of all reasonable bounds. Waste volume reduction efforts are currently insufficient to cope with increasing waste production rates. A significant reduction in waste volumes would result in an increase in radionuclide concentrations, for which current SLB practices are inappropriate. We must begin to develop methods to significantly reduce waste volumes and be prepared to accept higher radionuclide concentrations. Disposal alternatives must be developed to handle higher activity wastes. *In situ* waste treatment systems should be compared with other conventional waste treatment options. Treatment systems should emphasize radionuclide concentration and resource recovery, rather than reduced mobilization and small changes in volume. In its present form, shallow land burial is an inadequate waste disposal method.

Programs to develop a federal repository are aimed at identifying a suitable site and developing the technology to ensure long-term isolation of spent fuel, HLW, and TRU wastes. The major emphasis is on geologic disposal, but disposal in the seabed is being investigated as a near-term option. A geologic repository site is expected to be identified by 1985 and operational by the mid-1990s. Demonstration of capability to dispose of wastes in the seabed is expected after 1995.

Exploratory and development investigations are underway to study the suitability of salt (both bedded

and dome salt), granite, basalt, argillite, and tuff on land, and red clay sediments in stable ocean basins. Technology development includes hydrogeologic modeling, waste form development, waste packaging, penetration sealing, and *in situ* thermal and rock mechanical testing.

These programs are still relatively new, but seem to be well planned. It will be necessary to gather a considerable amount of information during the next few years. Because experimental results are not complete and specific technologies have not been selected, we feel that it is too early in these programs to evaluate the potential control technologies. We are, however, concerned that no information regarding recovery from accidents and postoperational monitoring has been located. Perhaps it is too soon to expect these issues to be addressed in a meaningful manner, but these questions must be resolved before actual operations will be allowed to begin.

In summary, shallow land burial has proved to be inadequate as a disposal method. It is too early in repository development programs to evaluate the potential control technologies.

VI. STANDARDS AND CRITERIA

A. Introduction

There are some areas of nuclear waste management that are almost without precedent. They differ substantially from traditional air and water pollution problems, especially in areas of future waste management technology. Demonstrating the adequacy of proposed future technology is difficult. Risk assessment is subject to falsely predicting a high consequence event, or failure to predict such an event. It is thought that the review process should reject projects, plans, or technologies, which may bear no risk, in order to avoid accepting proposals that may eventually prove hazardous (EPA-600/5-78-121, 1978).

Well-defined standards and criteria that could be applied to the task of assessing the overall long-term adequacy of a proposed technology would be helpful. The following discussion will explore the current definitions of radioactive waste, proper objectives for management of such wastes, and criteria for successfully meeting these objectives.

B. Definition of Radioactive Waste

Almost everything in the biosphere and all soils contain small amounts of naturally occurring and fallout radionuclides. If the only criterion is radionuclide content, then all solid wastes are radioactive. Obviously, there is no clear definition of radioactive waste. The Environmental Protection Agency attempted, without notable success, to define a minimum level of radioactivity *per se* for purposes of defining radioactive waste (EPA 1978a). Instead, their draft Criteria of Radioactive Wastes (EPA, 1978b) apparently starts from the premise that all wastes are indeed potentially radioactive materials, but only those radioactive materials requiring control for environmental and public health protection should be considered radioactive wastes "if they have no designated product or resource value and: a) are human produced ..., or as a result of regulatory activities are prohibited from uncontrolled discharge to the environment; or b) contain diffuse naturally occurring radioactive materials that, if disposed to the biosphere, would increase exposures to humans."

Apparently there are three considerations in this definition (EPA, 1978b). First, materials should be declared *waste* on the basis that they have no foreseen value. Obviously the designation of value has to be made by someone at some time. This issue must be resolved or the designation of value may become an escape clause. Second, they envision that radioactive wastes should include only those radioactive materials under some form of regulatory control. Such a designation automatically includes waste streams from any material having sufficient radioactivity to require a license. This intent is well met by source materials and by-product materials covered by the Atomic Energy Act, but leaves open some accelerator-produced nuclides and some naturally occurring radioactivity. EPA is looking to the Resource Recovery and Conservation Act of 1976 and the Clean Air Act of 1977 to possibly provide a mechanism to regulate such materials. These regulations may not be relevant. The third consideration is that if naturally occurring radioactivity in waste materials is accessible for human exposure only under normal natural conditions, radiological waste management would be of limited importance and hence the waste is *de facto* nonradioactive waste. The problem is definitively establishing whether a given waste stream material containing diffuse natural radioactivity would create an increased exposure condition over the same or similar materials under normal natural conditions.

Clearly the EPA approach to defining radioactive waste faces difficulties with matters of very long time frames and demonstration of compliance. In contrast to effluent monitoring, where there are identified release

points, well-characterized contaminated water or air being released, and recognized pathways to impacted populations, the matter of control and monitoring of compliance is exceedingly ill defined. Until very recently it has apparently been presumed that there should be *no* contaminated air or water releases from radioactive waste. Acceptable concentration and release rate limits to air and water are seemingly not derivable, and with regard to monitoring, it would appear that one is left to monitor everything (there are no pipes or stacks!), and the controlling monitoring limit is, by default, failure mode limit of "anything above background." Thus we have EPA declaring in their draft standards, "The fundamental goal for controlling any type of radioactive waste should be complete isolation over its hazardous lifetime" (EPA, 1978a). But how does one engineer for "complete isolation" over the hazardous lifetime (which could be millenia, by some accounts) of a waste, especially in the near-surface environment? Is shallow earth burial effectively ruled out? If so, how is the expense of a high-quality repository for just *any* type of radioactive waste justified?

The proposal to issue these Criteria for Radioactive Wastes has now been withdrawn by the EPA, perhaps in recognition of the many difficulties with this approach.

C. IRG Nuclear Waste Management Objectives

In March of 1978 the President established an Interagency Review Group on Nuclear Waste Management (IRG) to counter the impasse and conflicts in nuclear waste management. The IRG task was to formulate an administrative policy with respect to long-term nuclear waste management and to recommend programs to implement these policies. The IRG report (IRG, 1979) states several objectives for the federal agencies responsible for nuclear waste management. These objectives recognize that zero release cannot be assured, and that there must be pre-established release standards for waste management. The IRG objective includes (1) the primary objective of waste management, planning, and implementation is that existing and future nuclear waste should be isolated from the biosphere and pose no significant threat to public health and safety. However, (2) the selected technical options must meet all the relevant radiological protection criteria, as well as any other applicable regulatory requirements. Although zero release of radionuclides cannot be assured, any potential releases should be within pre-established standards and, beyond that, be reduced to the lowest level practicable.

Further, (3) the technology selected for waste disposal, as well as the reasons for its selection, must be well understood, clearly articulated, and widely accepted, and (4) the existence of residual technical uncertainties must be recognized and provided for in the program structure (IRG, 1979).

It is significant, in light of the issues associated with very long time frames mentioned above, that the IRG withdrew their initial formulation of the primary objective that contained reference to *risk*, and restricted themselves to addressing *release*. The conclusion was that, because of the time frames involved, it is not possible to reduce risks to pre-established standards. The goal should be to reduce releases (IRG, 1979). It will be shown in the following discussion that the requirement to ensure that *potential* releases meet certain standards places a considerable burden on the requirement that the waste disposal technology be well understood. In this context, being "well understood" entails being susceptible to long-range projection of performance over very long times.

D. Nuclear Waste Disposal Categories

It is apparent from the IRG description of the major classes of nuclear wastes that several rather different categories of management technology will be needed to meet the objectives of isolation from the biosphere with potential releases within pre-established standards. High-level wastes (HLW) include spent fuel and reprocessing wastes. Transuranic wastes (TRU), typically reprocessing or defense wastes, contain more than 10 nanocuries of transuranics per gram of material. The 10 nanocurie per gram concentration limit is a somewhat arbitrary standard currently under review (Healy and Rodgers, 1979a). Low-level wastes (LLW), which are wastes containing less than 10 nanocuries of transuranics per gram of material, require little or no shielding, but contain low but potentially hazardous concentrations or quantities of radionuclides. Uranium mine and mill tailings, because of their volume, are usually considered apart from LLW.

Waste management technologies traditionally associated with these categories are repository for HLW and TRU, shallow land burial for LLW, and controlled piles for tailings. However, Lawrence Livermore Laboratory (LLL) has proposed to NRC three generic disposal categories as a basis for associating appropriate standards and regulations with differing levels of control (King and Cohen, 1977).

1. Discharge directly to the biosphere with limited control and unpredictable release rates (for example, sanitary landfill), and with little or no expected intruder consequences.

2. Confinement in the biosphere for a period of time with controlled predictable low release rates (for example, a DOE or commercial shallow land burial facility, and with minimized intruder consequences because of limited concentrations at the time of burial).

3. Isolation from the biosphere in a repository to provide a high degree of assurance that the radionuclide content will not enter the biosphere (or will enter at a very low rate). Geologic isolation also provides a very low probability of intrusion.

These categories serve a useful purpose in the present assessment context because they suggest a range of criteria and standards needed to assess the adequacy of given control technologies in these categories. Included are (a) acceptable discharge (sanitary landfill) concentrations of radionuclides, and corresponding chemical/physical form if needed, (b) acceptable radionuclide concentrations, release rates, and acceptable resultant concentrations in air and water from controlled shallow land burial sites for radioactive wastes (confined waste sites) and (c) site design, performance, criteria, and waste form criteria for a technology that will meet the objective of isolation (a repository). The NRC has primary responsibility for providing these needed regulatory standards and criteria. They have recently attempted to fill the regulatory gap between existing standards and regulations and these needs in shallow land burial (confinement) technology through the publication of draft 10 CFR 61. This rule making promulgates licensing procedures, performance objectives, and technical criteria for disposal of LLW (Code of Federal Regulations, Title 10, Part 61 1979). Also, a draft environmental impact statement on 10 CFR Part 61 has recently been issued (NUREG-0782, 1981). To do so, however, is for them to face challenges in the arena of the IRG third objective, which requires that the selected technology (shallow land burial) be well understood (but it isn't), clearly articulated (no spokesman has emerged), and widely accepted (the recent rash of commercial burial ground closings and "nuclear initiatives" is not encouraging).

E. Current Applicable Standards and Criteria

1. General Remarks

The present draft EIS on Licensing Requirements for Land Disposal of Radioactive Waste, and the technical base for it represents a considerable revision and change in direction from the early (1979) draft and conception of 10 CFR 61. The rationale for this change is only briefly discussed in Chapter 7, page 7-2 of the main report. Whereas the early approach paid scant attention to the details of the waste streams going into near-surface land disposal (NSD), or to the technological options available or which might become available for waste form modifications, the new approach strains mightily to extract and process as much information about waste forms and burial technologies as is possible. The intent is apparently to use this approach to obtain the maximum utility (volumes and concentrations of radioactive waste) of NSD consistent with certain specified human exposure limitations. The NRC expresses this in terms of combining waste classification and regulatory requirements for waste form and packaging (page 7-2). The result is that NRC has set aside its regulatory concern for the whole of the low-level waste (LLW) problem in order to focus specifically on that portion of the wastes suitable for NSD. The shortcomings of this shift will be a recurring theme of this review.

The change leads to important changes in philosophy as well as in detail. Whereas the early concept (old 61.94) was to assure that releases and exposures are within the performance objectives *and* furthermore are as low as reasonably achievable (ALARA), the new concept (new 61.40) is to provide that "reasonable assurance exists" that exposures are within established limits. ALARA has been dropped. Further, the old approach made no attempt to build into the licensing requirements certain performance objectives based on projected long-term (hundreds or thousands of years) behavior of certain waste forms, and expectations of the effectiveness of certain disposal modalities to warn intruders to beware of hazardous materials. The new concept, in contrast, pushes precisely this approach to the limits of current knowledge of waste form behavior and impact assessment methodology. Thus the regulations require (61.56), and the waste classification is derived on the assumption that the more hazardous

wastes be "stable" (for example, do not degrade or promote slumping and collapse of the disposal unit) for at least 150 yr. In fact, for purposes of analysis, the stability is assumed for 500 yr.

Where does this divergence in concept of NSD lead? Part of the answer is that it leads to (1) optimistic projection of the long-term satisfactory performance of waste forms or disposal structures claimed to be stable even in the hostile environments of burial trenches in order to permit burial of greater quantities of LLW than would otherwise be allowed, (2) denigration of legitimate concerns for the consequences of placing radioactive materials in the near-surface environment, to obtain better acceptance of their new concept, and (3) a tendency to avoid full consideration of overall LLW management consequences of the proposed NSD regulation, especially the necessity for a disposal system for the new category of LLW "unacceptable for NSD by virtue of failing to meet the requirements of 10 CFR 61." That is, instead of taking full acount of the proper role that mined cavity disposal (MCD) or other greater confinement disposal (GCD) technologies could play in handling problematic waste forms for NSD, an attempt was made to maximize the number of technological "fixes" to make these wastes acceptable for NSD. The unforeseen and unhappy result of this new approach may well be to seriously undermine the efforts of NRC to provide a credible, workable, publically acceptable commercial NSD system. A clear alternative NRC has before it is to identify and build on the strengths and advantages of NSD, and avoid attempting to apply "fixes" to the weaknesses.

To reiterate what has been said in specific terms, it becomes clear upon reading these requirements that, despite the definition of "wastes" appropriate to these regulations in Sec. 61.2 (for example, *all* non-high-level wastes), there are really two categories of wastes to be anticipated in the commercial LLW waste streams in the future: (1) all the varied low-level wastes from institutional and reactor sources that meet the Performance Objective of this Part, and (2) those wastes that we thought, because of their origin, would be disposed by NSD, but cannot, because of failure to meet the provisions of the waste classification system (61.55) *and/or* waste characteristics (61.56). Thus what we might have expected to be considered in this proposed regulation is a closely linked waste management system that provides for long-term acceptable disposal of both forms of non-high-level waste, including that which must

be disposed in a system that provides greater isolation than NSD. The result might have been that, were MCD or other GCD concept properly and fully considered, it would have been found that all of the waste stream here considered to be unacceptable for NSD in simplest form (something like Spectrum 1), without special treatment in other words, could be accommodated in MCD along with all the wastes that will *have* to go there for lack of meeting the most stringent requirements for NSD. In the present Draft EIS the MCD concept is treated as an *alternative* of sorts to NSD, whereas it is really a necessary adjunct or part of it. Without MCD (or equivalent GCD) the industry will be facing yet another bottleneck to closure of the fuel cycle, much as it has with the spent fuel problem.

2. Review of the Main Report and Appendixes

There are far more details in the analysis and discussion presented in the main report and supporting documents than can be readily analyzed and assessed independently in a limited time. A thorough review could take several months. A major shortcoming of this EIS is, in fact, that the general turgid style, constant bombardment with ultimately trivial detail, and the absence of clear summary computations and discussions, makes the salient points of the supporting arguments practically untraceable. The complete derivation of the crucial waste classification concentration limits is *totally* untraceable because of the unavailability (at the time of the release of the EIS for public comment) of key reference documents, especially the *Data Base for Radioactive Waste Management*, Vols. 1, 2, and 3 from Dames and Moore. The absence of this data and methodology base renders impossible the full assessment of the suitability of the waste classification scheme, and thereby seriously undermines the credibility of the overall effort. The following discussions focus specifically on the tracing, to the degree possible with the given (and referable) data, the derivation of the concentration limits that form the basis for the waste classification system.

a. The Strategy of the EIS. In the discussion of the need for and description of the proposed action (page 1-5), the perceived need is for a disposal process that will accommodate *all* of the wide range of radioactive wastes that fall under the rubric "low-level waste." It is clearly foreseen that some of this waste is relatively innocuous ("room trash") and some may be intensely radioactive

(for example, sealed teletherapy sources). What is *not* advanced at this point, or ever in umistakable plain English, is that the "Proposed Action" is foreseen by NRC to provide a response to only *part* of the need, as discussed above. The remainder, and there will be a considerable remainder, of the LLW disposal need fades into a fuzzy background in the EIS, often confused with alternatives to the proposed NSD technology. The description of the proposed action (Sec. 1.2.5, page 1-12), therefore, is misleading and incomplete. It could be amended to reflect that only a portion of the present LLW stream will be accommodated by the proposed action. However, if an alternative proposed action were countenanced involving the proposal of a NSD technology only for the least hazardous component of LLW, coupled with an MCD technology for the remainder of the LLW, including both most of Class B and C waste and what comes to be known in this EIS as LLW unacceptable for NSD, then the whole tenor of the EIS would change. One could then say without qualification that the proposed action addresses "the standards of performance that should be met in LLW disposal," (but as it is, NRC can not.)

Then, in the codification of requirements into a new Part 61 Rule (Sec. 3.2.3), the full story would be out. We would be told that the new LLW disposal requirements will force us to provide a greater confinement disposal system along with a NSD system, and we would not be deceived into thinking that with some "modest increased effort and cost" we can provide technological fixes to make NSD adequate to the challenges of LLW disposal. More importantly, by keeping the focus of the analysis on the overall problem, the risk-benefit-cost analysis will certainly change, be more realistic, and, as will be shown below, the technology will be more effective in providing credible, long-term confinement of LLW.

b. Waste Source Term. It is certainly true that LLW in general is very diverse in terms of volume, activity, and characteristics (Sec. 3.4.2, page 3-8). Not only is it diverse, but there exists a large uncertainty in each of these aspects for most of the LLW streams. In many cases the relevant data are nonexistent, leading to estimates based on "scaling factors from known composition of similar or related waste streams" (page 3-10). The level of resultant uncertainty in concentration, form, and volume is enormous, and yet this fact is poorly acknowledged in all portions of the EIS. The reported "As Generated (Untreated) Isotopic Concentrations" of Table 3.3 (page 3-14) are not truly known to the

precision indicated (up to 1 part in 10^{11}!) in each of the waste streams, and, in many cases they are merely guesses. The issue here is not precision, however. It is the apparent disregard and even misuse of the uncertainties in the data that are of concern. We are assured in Appendix D that the data are "generally conservative" and sufficiently accurate to make decisions regarding performance objectives and technical criteria (page D-28). Such a vague statement may be defensible, but since the required data base is unavailable (Ref. D-25), because it is unpublished by NRC, this cannot be assessed. What one can do is note that, for example, whereas the NRC best estimate for ^{90}Sr in pressurized water reactor IX-resins is 1.94×10^{-4} Ci/m^3, a recent AIF estimate (Aikens, et al., Table 2-1, January 1979) based on a review of much of the same data is 8.5×10^{-2} Ci/m^3, or over 400 times larger. If two orders of magnitude is to be the guide for "sufficiently accurate," then the NRC should so state it, and what is more, carry this order of uncertainty through the remainder of the analysis, much as AIF did in theirs.

Of course, this would be difficult, inasmuch as so many of the estimated concentrations are only educated guesses. The guessing process consists of a concatination of geometric averages derived from a basic data set. First, we are told, the concentrations of the basic isotopes are calculated as the geometric mean of the data for each (page D-35). Then, a scaling factor (ratio of poorly known isotope concentration to known isotope concentration) for each isotope to be estimated from the basic set is computed by reactor type, etc. as a geometric average of a set of ratios of scaled to basic isotopes. Finally, geometric average scaling factors are then applied to the geometric average of the basic radio-nuclide concentration to obtain estimated concentrations of scaled radionuclides in all waste streams (page D-36). We are told the result is a conservative estimate, yet the uncertainties in the data must propagate nonlinearly in such a scheme. Normally, one would suppose that if the scaling factors were so divergent as to suggest the need for geometric averaging of the ratios, the conclusions would be that there is no good correlation between the basic and scaled isotope concentrations, and accordingly, that the scaling scheme will not work or must be used with caution.

Generally, uncertainties in the data or computational schemes should be properly accounted for in this EIS, yet nowhere are. The most direct effect of uncertainty in radionuclide concentration would likely be in estimating the volumes of waste falling in one or another of the

waste classes, or failing to be suitable for NSD. But indirectly, the uncertainties influence the parameters of the waste classification itself because of the additional factors NRC reserves for itself to "properly" account for the consequences of the highly heterogeneous nature of LLW, by allowing concentrations in certain classes that are higher than calculated intruder limits. This additional "flexibility," unwarranted in light of NRC accounts of the quality of the necessary data, will be discussed below.

c. *NSD and Human Intrusion Impacts.* Simply because it is a near-surface engineered artifact, any NSD facility is now widely perceived to represent a locus for eventual human reuse. Almost all the land features that make a site an excellent candidate for locating a NSD facility are also attractive for siting other land uses. The section discussing background on human intrusion (Sec. 4.2, page 4.2) is decidedly remiss in failing to point this out. Instead, we are told that even the "possibility" of human intrusion into a closed NSD facility is "only hypothetical," and this is reiterated many times. Such a claim is utterly false; the possibility is very real and, moreover, is substantial given the very large time frames over which the hazards of such a facility are appreciable. The alleged hypothetical character of anticipated future human interaction with a site is no more, and even possibly less, than the hypothetical character of the projected stability of "improved" waste forms over periods of hundreds of years based on a few tens of years of experience at most.

The choice of particular intrusion scenarios on which to base impact evaluations is admittedly a difficult one, and is difficult to deliver from arbitrariness. NRC scarcely even tries in this EIS, but there are paradigms of site intrusion and reuse in human history including: (1) many, many instances of human habitation sites being reused (perhaps with the sense of intrusion into the old site being inadvertent with respect to what may be buried there, but deliberate because of the desirability of the site for its location, soils, climate, etc.), and (2) the numerous known instances over millenia of grave, tomb, and monument robbers all over the globe (here with the sense of deliberate search for recoverable materials).

Note here that there are a number of senses of deliberate intrusion besides the perjorative one that NRC stresses, where the intrusion proceeds with knowing, willful disregard of hazards involved. This brief rehearsal of the paradigms of human intrusion is meant to make three points, that contrary to the tone of Chapter 4 of the EIS, human intrusion is not without precedent, that the

scenario in which the waste is contacted for extended periods of time is not "rather extreme" as stated (page 4-3), and that human intrusion is not a "*kind* of hypothetical event" (although a particular description of one might be) any more than "home construction" is.

The evident NRC paranoia about intruder consequences of NSD in Chapter 4 (especially page 4-6) is unwarranted and should be eliminated from the discussion. The convoluted and contradictory appeals to "conservatism" might then be eliminated. For example, one might not need to wonder what the result is supposed to be of NRC "conservatively" assuming certain intrusion scenarios to occur (page 4-6), and then with an evident twist in meaning, that "reasonably conservative" actions on the part of the intruders occurs. Is the result "conservative" or not?

In spite of the claim (page 4-7) that the intrusion scenarios being proposed were developed upon consideration of work of previous investigators of waste classification, NRC has chosen to depart from such consensus as there is in this area, and devise a series of separate scenarios replete with special time limitations on exposure, quantities, and types of waste contacted, and even exposure limits on which to base concentration limits. Although it would be presumptuous to say that the NRC intruder scenarios are unacceptable, it is fair to say that they derive from an unwarranted phobia for intrusion considerations, and fail to represent something resembling a consensus view on how to represent them in impact analyses. Wide acceptance of the means for accounting for intruder impacts is crucial for the rest of the program to develop a waste classification system.

An especially noteworthy development in this regard is presented in the first fruits of the combined detailed estimation of radionuclide source strengths in various waste streams and the application of specified intruder scenarios. It is a turning point in the EIS that passes without fanfare. For what emerges in pages 4-8 through 4-34, beginning with an analysis of the base case (no action) alternative and leading up to a consideration of means to mitigate impacts of human intrusion, is the realization that if one can reasonably depend on an institutional control period lasting at least 100 yr, there will be only a limited number of recognizable, separable waste streams that will present significant hazards to future intruders. In Table 4.7 (page 4-19) are shown calculated potential intruder exposures from four groups of waste streams under a category (waste spectrum) of wastes representing readily achievable improvement in the form of waste shipped for disposal. There are (1)

LWR process waste streams, (2) trash waste streams, (3) miscellaneous low-activity waste streams, and (4) "special" or high-activity waste streams. The expected hazard from these groups is nonuniform; for groups 2 and 3 it is relatively insignificant (less than 200 mrem after 100 yr), but potentially significant (greater than 1000 mrem even after 150 yr) for groups 1 and 4. Included in group 4 are LWR decontamination resins and failed reactor core components, which have long been putatively unacceptable for NSD. It is group 4 where much of the high TRU levels are expected in LLW and it is candidly admitted by NRC that it is these waste streams about which there is considerable uncertainty concerning radionuclide concentration (page 4-20). However, it is only in a one sentence afterthought where we find the admission that another waste disposal system (a greater confinement disposal facility) will probably be needed for these wastes (page 4-20)! Here is the turning point. With this realization, NRC would appear to have had two paths before it:

Path I. Recognize that a certain fraction of LLW will prove unacceptable for NSD, and that another disposal technology (GCD) will have to be advanced alongside NSD. Acknowledge this commitment and avoid compromising the credibility of NSD for group 2 and 3 wastes by applying technological "fixes" to portions of group 1 and 4 wastes to make them marginally acceptable for NSD. Formulate the waste classification system so that only class A and portions of class B wastes would be disposed in NSD, and class C or worse would have to be prepared for GCD. Account for the fact that the costs associated with preparing certain group 1 and 4 wastes to meet the stringent stability requirements of class C wastes would be saved by proper and full use of GCD, which will be needed anyway for those wastes that will inevitably not meet these requirements.

Path II. Recognize that although a certain fraction of LLW will prove unacceptable for NSD, one might, by one fix or another, fit more of group 1 and 4 wastes into NSD than would otherwise be acceptable. This will create conflicts on the fronts of intruder exposure consequences and costs, but these can be dealt with. The former by claiming that it is "incredible" that an intruder would interact with a "clearly recognizable high-integrity" waste form and stonewall any objections to this, and the latter by setting aside any serious consideration of GCD and the cost offsets associated with disposing of all of group 1 and 4 wastes in this system as

possible on the grounds that it is not "relevant" to the discussion of NSD.

Unhappily, as has been seen and will be further discussed, NRC has chosen the second path, to the detriment of both NSD and GCD planning and regulation.

d. Waste Form and the Intruder. Having made this decision in effect at this juncture, NRC has proceeded by first attacking the idea that waste form stability is a liability when it comes to expected intruder exposure, because stable, relatively highly contaminated wastes would not be diluted with ordinary trash and backfill soil (pages 4-28 and 4-29), and then by extolling the virtues of improved waste forms and segregated disposal (pages 4-29 - 4-37). The attack on the question of the consequences of waste form stability is really disingenuous. For, although they question the validity of the conclusion of certain studies that degradable packaging is of lower potential hazard and seem to be asserting that little effective dilution can be expected anyway, they later favorably cite these same studies (pages 7-12 and 7-13) in support of the claim that an additional dilution factor (beyond that already implicit in the geometric averaging fiasco discussed above) is warranted, because considerable dilution with uncontaminated materials *would* be expected to occur. Perhaps the debate would not be of great consequence were it not for the fact that, for different reasons, they really do need to have it *both* ways! Beyond the relative merits of both sides of the debate is the fact that the question would be foregone had NRC chosen the first path described above. Then these very wastes that the NRC is so stubbornly trying to assure us will cause no harm because "its incredible" that the intruder could contact "hunks of waste" and not realize that "something is wrong" (page 4-37), would be beyond the reach of the intruder because such wastes would not be accessible in a near-surface environment. Even the NRC confidence in near-surface confinement from intrusion has its limits, apparently around 500 yr. This too would not be an issue were the long-lived wastes disposed in GCD, as the NRC admits (page 4-47).

The dismissal of consideration of GCD on the grounds of cost (Sec. 4.3.5.4, page 4-47) is a weak counter to the question of why it was not properly considered in this EIS. Costs were not properly developed on the basis of full disclosure that the larger plan for LLW must include GCD. The implication in this section that the choice with which we are faced is

disposing all LLW by either NSD or mine cavity disposal is a false dilemma. We must have both acceptable NSD and acceptable GCD. The real issue is whether the present EIS described a suitable use of both.

e. Intruder Performance Objectives. The discussion of intruder dose limitation guidelines begins by emphasizing the appropriateness of *not* following the EPA guidance for releases to the general environment from other components of the fuel cycle, or the occupational radiation exposure guidelines. But, for some inexplicable reason, they fail to point out both in the narrative of Sec. 4.5, and the text of the proposed regulation (Sec. 61.42) that the exposure limits actually applied to the development and implementation of the waste classification system is a triad consisting of (1) dose equivalent *rate* limits of 500 mrem/yr for total body *and* bone, (2) dose equivalent *commitment* limits of 500 mrem total body and bone, and (3) 1500 mrem/yr and 1500 mrem committed to other organs. This fact emerges only parenthetically in Chapter 7 (page 7-6).

The EPA standards for the rest of the fuel cycle are said to be inappropriate for NSD because they do not seem appropriate to localized "accidental" exposure, which is probably true. This is backed up by the results of a cost effectiveness analysis, but in so doing, the issue of mined cavity disposal arises once again. The observation (page 4-61) that no analysis was performed to determine if waste unacceptable for NSD would be acceptable in a commercially operated MCD site is made in the course of indicating why the MCD cost estimates that are presented may not be correct. Putting another light on this same observation, it amounts to a confession that NRC has failed to provide for full regulation of all LLW and that, indeed, waste generators will face unknown additional costs for storage of wastes unacceptable for NSD until NRC acts. Even with the caveat on MCD costs in mind, a revealing picture of how a more complete LLW system might be costed out can be gleaned from Table 4.20 (page 4-58), "Comparison of Cases."

Suppose one does consider the case of applying the EPA standards to NSD of wastes in their present form (Spectrum 1), or nearly so (Spectrum 2). The volume of wastes unacceptable for NSD without special "fixes" then becomes larger by about an order of magnitude (about 9.7×10^4 m^3) than if a 500-mrem limit is used. The volume that still is acceptable for NSD is on the order of $3-5 \times 10^5$ m^3. For comparison purposes, the NSD costs (design, operation, long-term care) are about $459/m^3 at the high end and for MCD about $512/m^3. No matter what options are considered outside of the technological fixes, NSD appears to have a total cost on the order of 2.3×10^8. The MCD costs for the unacceptable wastes range from 1×10^7 (500 mrem limit) to 5×10^7 (EPA limit), but this factor of five must be evaluated in light of the fact that the true costs of total LLW disposal include both NSD costs and MCD costs to handle the unacceptable volume. Because the startup costs for MCD are estimated to be so much greater than for NSD (4×10^7 vs 7.5×10^6), the consequence of paying somewhat more per cubic meter for MCD is considerably offset. With that kind of investment already committed, a better use of the MCD resource, involving eliminating the requirement for "hot cell" facilities, grouting, and special waste stabilization requirements spelled out in Sec. 4.6.4 and Chapter 5, in favor of MCD options, would be in order.

The whole complexion of the migration and long-term stability discussion of Chapter 5 would then change for the better as well.

The premier siting requirement that the site shall be capable of being characterized, analyzed, and monitored (page 5-92) will then become consistent with the waste forms being disposed. For the very reason that "simple subsurface media are preferred for disposal sites" (page 5-92), one would have the greatest confidence in being able to predict the long-term performance capability of a site if the presence of high-activity "stabilized" waste forms (many of which will be buried at the bottom of trenches to avoid intruders, but where they will be more apt to be in constant contact with infiltrating water and closest to the water table) were eliminated. How does NRC anticipate the applicant will be able to demonstrate the performance capability of a proposed site when much of the radionuclide inventory can be expected to be tied up in waste forms whose long-term integrity can only be guessed?

f. Waste Classification Based on Intruder Consequences. The waste classification system in this EIS is the keystone of the LLW disposal edifice of NRC. The system is recognized in the EIS as the culmination of the Part 61 rulemaking effort (page 7-1). It is perceived as the mechanism that helps assure that the overall performance objectives will be met over the long term through collective reflection of the technical requirements and controls established for NSD.

Given the importance of this classification system, it is disastrous for the credibility and comprehensibility of the

proposed rulemaking that the published waste classification limits of Table 7.1 (page 7-7) and 7.2 (page 7-18) are neither fully traceable to published data bases nor reproducible from data and formulae published in the main report or appendixes of the EIS. The latter claim will be substantiated below in a discussion of the intruder methodology.

The basic strategy of the intruder scenario methodology is to identify "biota access locations" for wastes disposed in a reference NSD facility. These "locations" (contaminated soil, air, water, etc.) are taken to be such that there are very few controls possible on the movement of radioactivity in pathways to man beyond these points. Therefore, it is claimed (page G-10), resulting human exposures may be expressed through radionuclide-specific pathway dose conversion factors (PDCFs). These individual PDCFs represent the combined consideration of physical and biological transfer processes (deposition, plant uptake, etc.), human physiology (inhalation, ingestion, etc.), exposure modalities (direct gamma exposure, etc.), and fundamental dose conversion factor and models (for example, 50-yr dose commitments per pCi inhaled using the Task Group Lung Model). As such, they are totally unique to this document, and the many data and assumptions represented in them are not tabulated here. The details are apparently to be found in the unpublished data bases previously mentioned.

Those parts of the pathways that can be controlled are identified for each release/transport scenario by interaction factors. These are f_o = time delay factor to account for radioactive decay during the period of institutional control, f_d = site design factor, which accounts for engineered barriers and site practices, f_w = waste form consideration, which affect release/transport, and f_s = site selection factor, which includes effects of the natural site environment that contribute to reducing contaminant concentrations.

There are basically only two intruder scenarios that are said (more on this claim later) to determine a generic non site-specific classification system (page 7-5). These are intruder concentration limiting pathways (intruder construction and intruder agriculture). Potential ground water migration pathway analysis could, the NRC says, be used to limit concentration through a somewhat more roundabout process of first computing a limiting inventory for the site and then dividing by the reference site volume of wastes. But as NRC observes (page 7-5), it is considerably less straightforward to set out categories of waste on this basis because individual

wells on site and at the boundary (with one set of dose limits appropriate) and population water supply wells off site (with another set of limits applying) must be considered. These pathways are much more a function of site-specific environmental and geohydrological conditions than the reference site construction or agricultural concentration limiting pathways. The stated (page 7-5) approach, then, is to first determine waste classification requirements (based on concentration limits) considering protection of a potential inadvertent intruder. Then, for certain nuclides determined to be important from the standpoint of migration in ground water, inventory limits can be established on a site-specific basis.

On the basis of an analysis of the basic equations described in Appendix G (pages G-58 and G-62) for computing intruder dose limited concentrations, carried out for a variety of radionuclides, several interesting facts have emerged. Perhaps the most startling is that for the process of plants growing in composted soil-waste mixtures created during intrusion, the assumption is made that only a portion of the radionuclide content of the waste fraction is ever available for plant uptake, and what is more, that fraction is the same as the partition fraction observed between leachate and waste in the trench waters of Maxey Flats and West Valley. For some radionuclides this is a very small number indeed; for ^{239}Pu it is 4.67×10^{-4}. In the case of radionuclides for which plant uptake is the most limiting pathway, such as ^{14}C, the consequence of assuming only 5.76×10^{-3} of the waste activity is available for uptake, or that it all essentially is, is a two-order-of-magnitude difference in maximum average concentration limits. This approach is questionable in that normally one speaks of a concentration ratio between plant concentration of contaminant and soil concentration of contaminant, not between plant and a part of the contaminant thought to be leached into interstitial water (page G-63). This calls into question the availability of data on which to base such a scheme.

In contrast, for radionuclides such as ^{239}Pu, which are limited by inhalation pathways, the issue becomes one of how resuspension is handled. For the humid reference site the concentration limit is about 10 nCi/g, limited by the intruder construction scenario in which plant uptake has no role, but for the arid southwest site the limit is calculated to be only 0.79 nCi/g, an order of magnitude smaller. The NRC address to this sort of nontrivial difference is curious. They say that this consequence would be "compensated" for by the significantly lower decomposition of disposed waste (pages 7-10 and 7-11). Then allegedly there would be larger volumes of waste

"recognized as something other than dirt," the potential for dispersion would be reduced, as well as the likelihood for intrusion (page 7-11). Such a compensation, even if credible and relevant, seems hardly strong enough to warrant concluding that variations in site-specific environmental conditions should not have been considered in developing the waste classification system.

For some radionuclides such as ^{3}H and ^{90}Sr, it is difficult to reproduce the Table 7.1 values, no matter what is assumed, leading one to wonder if the ground water pathway was used to calculate a concentration limit rather than the intruder pathway. A code for this purpose is to be found in Appendix H.

Once again, the absence of the full details of the concentration limit derivation in the data base documents seriously hampers the analysis. But there is clear evidence in the two specific examples of grounds to be quite concerned about the suitability and defensibility of the results in Table 7.1.

For example, the final move to convert the Table 7.1 values to maximum allowable concentrations in Table 7.2 involves the previously discussed claim that for selected radionuclides, such as ^{137}Cs, the maximum allowable concentration would be raised by an additional factor of 10 to 20 to account for the effects of dilution by uncontaminated waste. Of course, the just-reviewed case of ^{239}Pu, where dilution was, in effect, held not to occur, reveals just how such an argument can be a two-edged sword if improperly handled.

g. Conclusions. The overall impact of the several shortcomings developed in this review of the EIS is to call into question the wisdom and justification for NRC proceeding with the proposed waste classification system and licensing requirements. If these issues remain unresolved, NRC may be placing a more carefully considered NSD option in serious jeopardy of not finding full acceptance by industry and the public.

F. Ensuring Adequacy of Shallow Land Burial by Maximum Concentration Limits

There is no simple answer to the bottom-line question posed in the preceding section, particularly if we recall that the proposed NRC primary dose limits and MACs do not seem to include ALARA considerations. The absence of consensus on assumptions and methods for computing appropriate concentration limits for various radionuclides in the environment, regardless of origin, is a good indication of the difficulties encountered in

attempting to determine defensible MACs. This also explains why it was suggested in the discussion of the IRG objectives that part of the problem may be traced to a lack of sufficient understanding of the technology (shallow land burial).

G. Summary and Conclusions

Determining the adequacy of a given nuclear waste management technology is a unique and peculiarly difficult undertaking. We have suggested that many perceive that the decision making process should be aimed at minimizing the error of failing to predict adverse consequences of apparently safe technology, even though the impacts may be remote in time and have a very low probability of occurrence. In that context it is all the more urgent to have well-defined standards and criteria by which to assess waste management technologies. The necessary objectives for this task have been outlined by the Interagency Review Group.

1. Nuclear waste should be isolated from the biosphere and pose no significant threat to public health and safety.

2. Recognizing that zero release cannot be assured, pre-established release standards (which include the principle of ALARA) must be provided.

3. The waste management technology selected, as well as the reasons for its selection, should be well understood, clearly articulated, and widely accepted. Residual uncertainties must be recognized and anticipated.

These IRG objectives should provide a much needed focus for regulatory activities. The requirements of understanding, articulation, and acceptance may be particularly difficult to achieve.

The NRC has prepared draft performance objectives and technical criteria for the technology of disposal of low-level waste by shallow land burial. These are, however, dual standards, based on existing radiation protection regulations (from EPA and NRC), which have a different set of risk/benefit/technology considerations from those appropriate to shallow land burial. This is particularly clear in regard to dealing with site intrusion.

The NRC attempt to assure the long-term adequacy of shallow land burial by taking into account the possibility of site intrusion includes provision of concen-

tration limits for specific radionuclides. These limits were derived to reduce maximum exposures to intruders to below some prescribed level (500 mrem/yr), while at the same time allowing maximal use of the site.

We analyzed the process by which these concentration limits are derived. Although there may be fair agreement on the pathway most likely to cause the greatest exposure (and thus most limiting), there is little or no consensus on the proper set of data and assumptions to describe that pathway. This is shown by the widely divergent limits resulting from several independent limit derivations. There is simply no way to independently test and verify the assumption set involved in a particular limit derivation, especially since so much depends on intruder characterization.

Our review of the draft environmental impact statement on shallow land burial has led to a number of specific criticisms and recommendations, particularly with respect to the manner in which intruder scenarios were used to derive limits. These comments are included as an appendix to this report.

REFERENCES

W. V. Abeele, "Future Credible Precipitation Occurrences in Los Alamos, New Mexico," Los Alamos Scientific Laboratory report LA-8523-MS (1980).

J. A. Adam and V. L. Rogers, "A Classification System for Radioactive Waste Disposal—What Waste Goes Where?," US Nuclear Regulatory Commission report NUREG-0456 (June 1978).

A. E. Aikens, R. E. Berlin, James Clancy, and O. I. Oztunali, "Generic Methodology for Assessment of Radiation Doses from Groundwater of Radionuclides in LWR Wastes in Shallow Land Burial Trenches," Atomic Industrial Forum report AIF/NESP-013 (1979).

Allied Chemical Corporation, Special Chemicals Division, Docket File, Attachment to Form AEC-2, Application for Amendment to Source Material License No. SUB-526, USAEC Docket No. 40-3392, August 21, 1970.

R. L. Anderson, L. R. Cooley, T. J. Beck, and C. S. Strauss, "Institutional Radioactive Wastes," US Nuclear Regulatory Commission report NUREG/CR-0028 (March 1978).

B. J. Barnhart, E. W. Campbell, E. Martinez, D. E. Caldwell, and R. Hallett, "Potential Microbial Impact on Transuranic Wastes Under Conditions Expected in the Waste Isolation Pilot Plant (WIPP)—Annual Report, Oct 1-Sept 30, 1979," Los Alamos Scientific Laboratory report LA-8297-PR (1980).

R. L. Bates, Geology of the Industrial Rocks and Minerals (Dover Publications, Inc., New York, 1969), pp. 218-219.

T. J. Beck, L. R. Cooley, and M. R. McCampbell, "Institutional Radioactive Wastes - 1977," US Nuclear Regulatory Commission report NUREG/CR-1137 (October 1979).

W. R. Bishop and F. J. Miraglia, "Environmental Survey of the Reprocessing and Waste Management Portion of the LWR Fuel Cycle," US Nuclear Regulatory Commission report NUREG-0116 (October 1976).

S. R. Borrowman and P. T. Brooks, "Radium Removal from Uranium Ores and Mill Tailings," Salt Lake City Metallurgy Research Center, US Department of Interior Report of Investigations 8099 (1975).

R. W. Brown and R. H. Williamson, "Domestic Uranium Requirements," Policy and Evaluation, DOE. Presented at the Uranium Industry Seminar, Grand Junction, Colorado, October 26, 1977.

L. J. Carter, "Debate Over Wastes Imperils 3-Mile Cleanup," Science, Vol. 210 (October 1980).

D. A. Clark, "State-of-the-Art: Uranium Mining, Milling, and Refining Industry," Office of Research and Development, Environmental Protection Agency report EPA-660/2-74-038 (June 1974).

W. R. Chappel, "Transport and Biological Effects of Molybdenum in the Environment," in Heavy Metals in the Aquatic Environment, an International Conference, P. A. Krenkel, Ed. (Pergamon Press, 1975).

Code of Federal Regulations, Title 10, Part 20, "Standards for Protection Against Radiation."

Code of Federal Regulations, Title 10, Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix F.

Code of Federal Regulations, Title 10, Part 60, "Technical Criteria for Regulating Geologic Disposal of High-Level Radioactive Waste—Proposed Rule Making," Federal Register, Vol. 45, No. 94, May 13, 1980.

Code of Federal Regulations, Title 10, Part 61, "Disposal of Low-Level Radioactive Waste and Low Activity Bulk Solid Waste," preliminary draft, November 5, 1979.

Code of Federal Regulations, Title 10, Part 71, "Packaging of Radioactive Material for Transport and Transportation of Radioactive Material Under Certain Conditions."

P. Columbo and A. J. Weiss, "Evaluation of Isotope Migration—Land Burial, Water Chemistry at Commercially Operated Low-Level Radioactive Waste Disposal Sites, Progress Report No. 11, October-December 1978," Nuclear Regulatory Commission report BNL-NUREG-5113 (NUREG/CR-1167).

Dames and Moore, "Development of Monitoring Programs for ERDA Owned Radioactive Low-Level Waste Burial Sites," prepared by Dames and Moore, White Plains, New York under ERDA Contract E(49-1)-3759, July 1976.

J. L. Deichman, "Unresolved Technical Issues in Land Burial of Low-Level Radioactive Waste," prepared by Shallow Land Burial Steering Committee, Oak Ridge National Laboratory report ORNL/NFW-79/62 (October 1979).

Division of Reactor Development, AEC and Public Health Service, Washington, DC, "Sanitary Engineering Aspects of the Atomic Energy Industry," TID-7517, Seminar held at Cincinnati, Ohio, December 1955.

DOE Budget, "DOE FY-81 Congressional Budget Request," US Department of Energy report (1980).

DOE/EDP-0058, "Environmental Development Plan—Uranium Mining, Milling, and Conversion," US Department of Energy report (August 1979).

DOE/EIS-0015, "Final Environmental Impact Statement, U. S. Spent Fuel Policy, Vol. 2: Storage of U. S. Spent Power Reactor Fuel," US Department of Energy report (May 1980).

DOE/EIS-0026, "Final Environmental Impact Statement—Waste Isolation Pilot Plant," US Department of Energy report (October 1980).

DOE/EIS-0046-D, "Draft Environmental Impact Statement—Management of Commercially Generated Radioactive Waste," US Department of Energy report, Vol. 1 (April 1979).

DOE/NE-0008, "Nuclear Waste Management Program Summary Document, FY 1981," US Department of Energy report (March 1980).

J. W. Dollahite, L. D. Rowe, L. M. Cook, D. Hightower, E. M. Mailey, and J. R. Kyzar, "Copper Deficiency and Molybdenosis Intoxication Associated with Grazing Near a Uranium Mine," The Southwest Veterinarian (Fall 1972), pp. 47-50.

C. H. Dowding and J. B. Miller, Comparison of Predicted and Encountered Geology for 7 Colorado Tunnels, (MIT Press, Cambridge, Massachusetts, 1975).

D. R. Dreesen and M. L. Marple, "Uptake of Trace Elements and Radionuclides from Uranium Mill Tailings by Four-Wing Saltbush (Atriplex canescens) and Alkali Sacaton (Sporobolus airoides)," Symposium on Uranium Mill Tailings Management, Fort Collins, Colorado. Geotechnical Engineering Program, Civil Engineering Department, Colorado State University, November 19-20, 1979.

D. R. Dreesen, M. L. Marple, and N. E. Kelley, "Contaminant Transport, Revegetation, and Trace Element Studies at Inactive Uranium Mill Tailings Piles," Symposium on Uranium Mill Tailings Management, Fort Collins, Colorado, Geotechnical Engineering Program, Civil Engineering Department, Colorado State University, November 20-21, 1978.

J. O. Duguid, "Annual Progress Report of Burial Ground Studies at Oak Ridge National Laboratory: Period Ending September 30, 1975," Oak Ridge National Laboratory report ORNL-5141 (October 1976).

J. O. Duguid, "Assessment of DOE Low-Level Radioactive Solid Waste Disposal/Storage Activities," Battelle Columbus Laboratories report BM1-1984 (November 1977).

EDP Commercial Waste, "Environmental Development Plant for LWR Commercial Waste Management," US Department of Energy (March 1980).

EDP Defense Waste, "Environmental Development Plan for Defense Waste Management," US Department of Energy (March 1980).

K. J. Eger and G. E. Zima, "Commentary on Spent Fuel Storage at Morris Operation," US Nuclear Regulatory Commission report NUREG/CR-0956, Pacific Northwest Laboratories report PNL-3065 (October 1979).

EPA, "Proceedings of a Public Forum on Environmental Protection Criteria for Radioactive Wastes," Environmental Protection Agency report ORP/CSD-78-2 (May 1978a).

EPA, "Criteria for Radioactive Wastes, Recommendations for Federal Radiation Guidance," Federal Register, Vol. 43, No. 221 (November 15, 1978b).

EPA-600/5-78-121, "Social Decision-Making for High Consequence, Low Probability Occurrences," National Planning Council, Washington, DC and Haldi Associates, Inc., New York, US Environmental Protection Agency report (October 1978).

EPA-906/9-75-002, "Water Quality Impacts of Uranium Mining and Milling Activities in the Grants Mineral Belt, New Mexico," US Environmental Protection Agency report, Region VI, Dallas, Texas (September 1975).

ERDA-1543, "Final Environmental Statement: Expansion of U.S. Uranium Enrichment Capacity," US Energy Research and Development Administration report (April 1976).

ERDA-1549, "Draft Environmental Statement: Portsmouth Gaseous Diffusion Plant Expansion, Piketon, Ohio," US Energy Research and Development Administration report (June 1976).

ERDA-1555, "Final Environmental Impact Statement—Portsmouth Gaseous Diffusion Plant Site, Piketon, Ohio," US Energy Research and Development Administration report (May 1977).

ERDA-7643, "Alternatives for Managing Wastes from Reactors and Post-Fission Operations in the LWR Fuel Cycle," US Energy Research and Development Administration report, Vol. 4 (May 1976).

Federal Register, "Waste Isolation Pilot Plant; Record of Decision," Vol. 46, No. 18 (1981), pp. 9162-9164.

Federal Register, Vol. 45, No. 197 (October 8, 1980).

Federal Register, "Nuclear Regulatory Commission Uranium Mill Licensing Requirements," Vol. 45, No. 194 (October 3, 1980), pp. 65521-65538.

Federal Register, "Environmental Protection Agency Hazardous Waste Management System, Part III, Identification and Listing of Hazardous Waste," Vol. 45, No. 98 (Book 2 of 3 books, May 19, 1980), p. 33094.

Federal Register, "Environmental Protection Agency, Subchapter N-Effluent Guidelines and Standards, Part 440-Ore Mining and Dressing Point Source Category, Effluent Limitations Guidelines for Existing Sources," Vol. 43, No. 133 (July 11, 1978).

R. M. Fry, "Radiation Hazards in Uranium Mining and Milling," Australian Atomic Energy Commission report AAEC/IP-9 (September 1975).

GJO-100(80), "Statistical Data of the Uranium Industry," US Department of Energy, Grand Junction Office (January 1, 1980).

GJO-100(79), "Statistical Data of the Uranium Industry," US Department of Energy, Grand Junction Office (January 1, 1979).

GJO-100(78), "Statistical Data of the Uranium Industry," US Department of Energy, Grand Junction Office (January 1, 1978).

GJO-100(77), "Statistical Data of the Uranium Industry," US Department of Energy, Grand Junction Office (January 1, 1977).

GJO-100(76), "Statistical Data of the Uranium Industry," US Department of Energy, Grand Junction Office (January 1, 1976).

R. W. Granlund, *Incineration of Waste Scintillation Fluids*, (Pergamon Press, Belfast, Northern Ireland, 1978).

B. D. Guilbeault and B. J. Reckman, "Preliminary State-by-State Assessment of Low-Level Radioactive Wastes Shipped to Commercial Burial Grounds," NUS Corporation, draft report prepared for EG&G Idaho, Inc. (August 1979).

T. E. Hakonson, J. W. Nyhan, L. J. Johnson, and K. V. Bostick, "Ecological Investigation of Radioactive Materials in Waste Discharge Areas at Los Alamos," Los Alamos Scientific Laboratory report LA-5282-MS (May 1973).

J. W. Healy and J. C. Rodgers, "Limits for the Burial of the Department of Energy Transuranic Wastes," Los Alamos Scientific Laboratory document LA-UR-79-100 (January 1979a).

E. H. Hirsch, "A New Irradiation Effect and its Implications for the Disposal of High-Level Radioactive Waste," Science, Vol. 209 (September 1980).

IRG 1979, "Report to the President by the Interagency Review Group on Nuclear Waste Management," National Technical Information Service report TID-29442 (March 1979).

G. H. Jenks, "Effects of Gaseous Radioactive Nuclides on the Design and Operation of Repositories for Spent LWR Fuel in Rock Salt," Oak Ridge National Laboratory report ORNL-5578 (December 1979).

A. B. Johnson, "Behavior of Spent Nuclear Fuel in Water Pool Storage," Battelle Northwest Laboratories report BNWL-2256 (September 1977).

J. K. Johnstone, and K. Wolfsberg, "Evaluation of Tuff as a Medium for a Nuclear Waste Repository: Interim Status Report on the Properties of Tuff," Sandia National Laboratories report SAND 80-1464 (July 1980).

D. R. Kasper and W. H. Engelmann, "Compliance with Environmental Regulations for *In Situ* Uranium Mining: From Discovery to Closure," presented at the Bureau of Mines Technology Transfer Seminar on *In Situ* Leaching and Borehole Mining of Uranium, Denver, Colorado, July 26, 1978.

R. F. Kaufmann, G. G. Eadie, and C. R. Russell, "Effects of Uranium Mining and Milling on Ground Water in the Grants Mineral Belt, New Mexico," Ground Water, 14(5), 296-308 (September-October 1976).

A. H. Kibbey and H. W. Godbee, "A State of the Art Report on Low-Level Radioactive Waste Treatment," Oak Ridge National Laboratory report ORNL/TM7427 (September 1980).

W. C. King and J. J. Cohen, "Interim Report to the NRC on Radioactive Waste Classification," Lawrence Livermore Laboratory report UCID-17497 (April 1977).

E. L. Klepper, L. E. Rogers, J. D. Hedlund, and R. G. Schreckhise, "Radioactivity Associated with Biota and Soils of the 216-A-24 Crib," Pacific Northwest Laboratories report PNL-1948 (March 1979).

T. W. Lambe, "Predictions in Soil Engineering," Geotechnique, 23(2), 149-202 (1973).

M. E. Martin and L. J. Riales, "Tennessee Valley Authority's Low-Level Radioactive Waste Management and Associated Environmental Impacts," in *Waste Management '81*, proceedings of American Nuclear Society Topical Meeting, Tucson, Arizona, R. G. Post, Ed., February 23-26, 1981, pp. 596-579.

J. P. McBride, R. E. Moore, J. P. Witherspoon, and R. E. Blanco, "Radiological Impact of Airborne Effluents of Coal and Nuclear Plants," Science, 202(4372) (December 1978).

R. C. Meritt, *The Extractive Metallurgy of Uranium*, (Colorado School of Mines Research Institute, 1971).

A. C. Miegh, "The Triassic Rocks, with Particular Reference to Predicted and Observed Performance of Some Major Foundations," Geotechnique, 26(3), 391-452, 1976.

NAS, "The Disposal of Radioactive Waste on Land," National Academy of Sciences—National Research Council, Washington, DC (1957).

NAS, "Solidification of High-Level Radioactive Wastes," National Academy of Sciences—National Research Council, Washington, DC (1978).

New Mexico Energy and Minerals Department, "An Overview of the New Mexico Uranium Industry (Draft)," January 1979.

New Mexico Health and Environment Department, Environmental Improvement Division, Water Pollution Control Bureau, "Water Quality Data for Discharges from New Mexico Uranium Mines and Mills," July 1980.

Nuclear Fuel Services, Inc., "Safety Analysis Report, NFS Reprocessing Plant, West Valley, New York," Docket No. 50-201, Section V-8, Vol. I, 1973.

NUREG-0002, "Final General Environmental Statement on the Use of Recycled Plutonium in Mixed Oxide Fuel in Light Water Cooled Reactors, Chapter IV, Environmental Impact Due to the Implementation of Plutonium Recycle," Nuclear Regulatory Commission report, Vol. 3 (August 1976).

NUREG-0216, "Public Comments and Task Force Responses Regarding the Environmental Survey of the Reprocessing and Waste Management Portions of the LWR Fuel Cycle," US Nuclear Regulatory Commission report (March 1977).

NUREG-0511, "Draft Generic Environmental Impact Statement on Uranium Milling," US Nuclear Regulatory Commission report, Vol. 1 (April 1979).

NUREG-0575, "Final Generic Environmental Impact Statement, Handling and Storage of Spent Light Water Power Reactor Fuel," US Nuclear Regulatory Commission report (August 1979).

G. K. Oertel, "Low-Level Waste Management," paper presented at Fuel Cycle Conference '80, Atomic Industrial Forum, Inc., New Orleans, Louisiana, April 15-18, 1980.

ORNL/NUREG/TM-7, "Correlation of Radioactive Waste Treatment Costs and the Environmental Impact of Waste Effluents in the Nuclear Fuel Cycle—Conversion of Yellowcake to Uranium Hexafluoride. Part I. The Fluorination-Fractionation Process," Oak Ridge National Laboratory report (September 1977).

ORNL-TM-4902, "Correlation of Radioactive Waste Treatment Costs and the Environmental Impact of Waste Effluents in the Nuclear Fuel Cycle for Use in Establishing 'As Low as Practicable' Guides—Fabrication of Light-Water Reactor Fuel from Enriched Uranium Dioxide," Oak Ridge National Laboratory report (May 1975).

ORO-684, "AEC Gaseous Diffusion Plant Operation," Oak Ridge Operations Office report (January 1972).

C. F. Park, Jr. and R. A. MacDiarmid, Ore Deposits, (W. H. Freeman and Co., San Francisco, 1970), pp. 59-144

B. A. Perkins, "Evaluation of Environmental Control Technologies for Commercial Nuclear Fuel Conversion (UF_6) Facilities, " Los Alamos National Laboratory report, 1982a (in preparation).

B. A. Perkins, "Evaluation of Environmental Control Technologies for Commercial Uranium Nuclear Fuel Fabrication Facilities," Los Alamos National Laboratory report, 1982b (in preparation).

J. Phillips, F. Feizollahi, R. Martineit, and W. Boll, "A Waste Inventory Report for Reactor and Fuel Fabrication Facilities," Office of Nuclear Waste Isolation report ONWI-20, NUS 3314 (March 1979).

P. L. 95-604, Uranium Mill Tailings Radiation Control Act of 1978.

D. W. Powers, S. J. Lambert, S. Shaffer, R. L. Hill, and W. D. Weast, "Geologic Characterization Report, Waste Isolation Pilot Plant (WIPP) Site, Southeastern New Mexico," Sandia National Laboratory report SAND 78-1596 (August 1978).

"Report to the American Physical Society by the Study Group on Nuclear Fuel Cycle and Waste Management," supplement to Reviews of Modern Physics, 50, 1, Part II, S8 (January 1978).

J. F. Reynolds, M. J. Cwik, and N. E. Kelley, "Reclamation at Anaconda's Jackpile Uranium Mine," presented at the 1st Annual General Meeting of the Canadian Land Reclamation Association, University of Guelph, Ontario, Canada, November 23, 1976.

J. V. Rouse, "Environmental Considerations of Uranium Mining and Milling," Mining Engineering (October 1978).

K. J. Schneider and T. J. Kabele, "Description of Reference LWR Facilities for Analysis of Nuclear Fuel Cycles," Pacific Northwest Laboratory report PNL-2286 (September 1979).

J. J. Swift, J. M. Hardin, and H. W. Calley, "Potential Radiological Impact of Airborne Releases and Direct Gamma Radiation to Individuals Living Near Inactive Uranium Mill Tailings Piles," US Environmental Protection Agency, Office of Radiation Programs report EPA-520/1-76-001 (January 1976).

J. B. Squyres, "Origin and Depositional Environment of Uranium Deposits of the Grants Region, New Mexico," a dissertation submitted to the Department of Geology and Committee on the Graduate Division of Stanford University in partial fulfillment of the requirements for the Degree of Doctor of Philosophy, January 1970.

Texas Department of Health, Division of Occupational Health and Radiation Control, "Texas Environmental Radioactivity Surveillance, Results of Samples Analyzed During the 3rd Quarter, 1976," August 1977.

E. C. Tsivoglou and R. L. O'Connell, "Waste Guide for the Uranium Milling Industry," US Department of Health, Education, and Welfare, Public Health Service, Division of Water Supply and Pollution Control, Robert A. Taft Sanitary Engineering Center, Cincinnati, Ohio, 1962.

M. E. Wacks, "Alternative to Shallow Land Burial for the Disposal of Low Level Wastes," Engineering Experiment Station, College of Engineering, University of Arizona, Tucson, Quarterly Report to Los Alamos Scientific Laboratory, Contract L 28-9766F-1, June 1979.

WASH-1248, "Environmental Survey of the Uranium Fuel Cycle," US Atomic Energy Commission report (April 1974).

S. W. West, "Disposal of Uranium-Mill Effluent by Well Injection in the Grants Area, Valencia County, New Mexico," Geological Survey Professional Paper 386-D, prepared in cooperation with the New Mexico State Engineer Office and the US Atomic Energy Commission, US Government Printing Office, Washington, 1972.

M. L. Wheeler, W. J. Smith, J. L. Warren, and W. J. Whitty, "Burial Grounds," International Symposium on the Management of Wastes from the LWR Fuel Cycle, CONF-76-0701, July 1976.

F. W. Whicker, "Considerations of Ecology, Environmental Radioactivity, and Environmental Monitoring for the Shirley Basin Uranium Mine," Department of Radiology and Radiation Biology, Colorado State University, Fort Collins, Colorado, 1972.

C. B. Woodhall, "U.S. and Free World Discharged Nuclear Fuel, Storage and Transportation Analysis," Nuclear Assurance Corporation, Atlanta, Georgia, April 1977.

APPENDIX

GENERAL COMMENTS ON THE DRAFT EIS FOR
10 CFR 61 (NUREG-0782)

The following comments, grouped as general and technical, are offered as brief summary statements.

General Issues

(1) This EIS is to be commended for trying to bring a voluminous amount of data into one set of documents that can be used as a beginning for open discussion on the issue of disposal of radioactive wastes. Moreover, the presentation of the concept of intruder scenarios is important in determining possible pathways for mobilization of radionuclides. The documents also suggest the necessity for *de minimus* shallow land burial and deeper confinement classification types of wastes, which is important.

(2) The proposed regulations purport to provide a generic waste classification system, which is an "umbrella" (Main Report 7.1) under which the disposal of all types of non-high-level wastes can be regulated. This umbrella approach, in the manner in which the regulations are formulated, is flawed, because in some cases the linking of the waste classification system with the specific disposal requirements for near-surface land disposal (NSD) (Main Report 7.1) has the result of imposing technological fixes (such as waste form, and layering) on classes of wastes that might have otherwise been candidates for disposal by some form of greater confinement disposal (GCD) system. These technological "fixes" are questionable because their long-term containment properties are poorly understood and largely untested under the expected geophysical conditions of shallow land burial and possible future land use.

(3) The regulations in classifying waste concentrations as a function of volume appear to encourage dilution as a means of allowing NSD for some types of wastes. The draft EIS should discuss whether this type of "technical fix" is desirable or whether greater confinement facilities should be the preferred mode of disposal.

(4) The regulations in classifying waste as a function of radionuclide content place an extremely large burden on the accurate measurement of these radionuclides. In many cases the use of scaling factors will not be satisfactory because of the variation of radionuclide distribution as a function of time, operating parameters, specific events during operation, and initial conditions. These accurate measurement requirements (many of which are technically difficult to make) in turn place a large burden on the waste generator, disposal site operator, and finally on the governmental manager (who must accept that the wastes have been accurately classified).

(5) There are many uncertainties in the data base used to develop the proposed classification system. The draft EIS itself contains numerous references to the highly uncertain and often extremely variable nature of much of the data used in determining the concentration, inventory, and hazard potentials of low-level wastes (note in particular the discussion of uncertainties in source characterization in Sec. 3 of Appendix D, of intrusion pathway characterization in the Main Report, Ch. 4, and especially the problems of defining the uncertainties in the proposed methods and parameters used to assign

hazard reduction credits for supposedly stable, non-dispersible, low-leaching waste forms in the Main Report, Sec. 4.3.4). The propagation of these uncertainties through the system to the formulation of waste classification itself is nowhere explicitly evaluated and incorporated in these regulations. The order-of-magnitude increase of waste classification limits for ^{137}Cs over calculated concentration limits (Main Report, Sec. 7.2.5) is particularly questionable in light of these uncertainties.

(6) Although ALARA considerations are mentioned in the EIS (Main Report, Sec. 7.2.5, 7.2.6, and elsewhere), it is always with the qualifiers "in the interests of" or "in the spirit of" ALARA on the part of NRC, rather than where it is perhaps most needed as part of performance objective for waste generators and site operators (as was the case in earlier versions of these regulations). Thus, the requirement to meet or exceed (in the sense of ALARA) the performance objectives of these regulations has been improperly lifted from the requirements of this Part.

(7) An oversight noted is that NRC has not stated its support for regulation of the hazardous, nonradioactive components in low-level wastes, or how these regulations might be integrated with the requirements of this Part.

(8) Neither the documentation of the data bases for waste stream characterization (references 5 and 46, App. D) nor documentation of the data and methodology for the pathway analysis (references 1, 6, and 12, App. G) has been available for review because of a failure by NRC to have them published and available with the release of the EIS. This severely limits a thorough assessment of the suitability of the EIS on the proposed action.

(9) Although the draft EIS covers burial sites that may be licensed in the future, the draft does not discuss the impact of the regulations on sites that are presently in use or have been used. Some type of appendix is needed to indicate how the proposed regulations may affect these existing sites.

Technical Issues

(1) As mentioned in the general comments, there are uncertainties and technical difficulties in determining radionuclide content, yet the regulations are very specific about permitted concentrations in the three waste

categories. The responsibility for correctly segregating and properly certifying that wastes meet the requirements of this Part rests solely on waste generators. The NRC recognizes that waste generators face severe operational difficulties and economic penalties in meeting the requirements by direct measurement and are prepared to compromise in terms of "scaling factors" (Main Report, Sec. 7.5). The stated (Main Report, Ch. 7, App. G) examples of how this might work are fraught with many technical difficulties, including two in particular: (1) Because the data on radionuclide concentration in most waste streams are highly variable (App. G) or completely unknown and must be guessed (Main Report, Ch. 7), resort has been made to dubious techniques such as forming products of geometric averages of basic data with geometric averages of ratios of known to unknown concentrations to obtain what are claimed to be "reasonable" scaled estimates of unknown radionuclide concentrations. These very data bases lead one to wonder if requisite correlations between known and unknown radionuclides can be established with sufficient reliability to make the scaling factor approach acceptable, as attractive as it may be from an operational viewpoint; (2) in application (Main Report, Table 7.6), the scaling factor concept seems to be applied as though there were no other radionuclides present in a given waste stream except the pair of measured and target nuclides, and that the sum of fractions rule (Main Report, Sec. 7.4.2) for mixtures does not apply. For these and other reasons the concept presents a disturbing prospect for quality assurance and enforcement. Ultimately it may force disposal site operators to prepare their own verification system in self defense, as has been seen in some recent temporary site closures in Nevada and Washington. From a larger perspective this issue can be seen to be the result of having drawn a box around the problem of low-level waste disposal and then attempting to force the solution to fit the box. The box is the linking of waste classification and the requirements of waste form and disposal by shallow land burial. The solution then only *seems* to require some means for the generator to practically and economically segregate, identify specific radionuclide content, and modify waste form or package to meet the requirements of disposal in the near-surface environment. Of course, another solution is to reformulate the problem in terms of the constructive role greater confinement disposal technologies can play alongside NSD, and thereby redefine the requirements for disposal, taking into account the many advantages GCD offers with respect to contaminant migration and

human reuse of a site and thus, considerably modify and ease the burdens of measurement and waste form modification for the generators. Such a solution would greatly enhance the prospects for quality assurance (QA) and enforcement as well.

(2) The NRC should directly address the QA and enforcement issues of these proposed regulations in the EIS and not leave them to a proposed Regulatory Guide (Main Report, Sec. 7.5). The NRC has itself identified elsewhere many problems with QA programs in other aspects of the nuclear industry, including unqualified workers and QA inspectors, falsified records, lack of authority, lack of communication, inadequate corrective action systems, lack of supervision, and poor to non-existent procedures. The proposed scheme to implement waste classification minimally sketched out in the EIS (Main Report, Ch. 7) could too easily suffer these sorts of QA deficiencies and should be carefully reconsidered. Preferably, such reconsideration would be done in a context that would make it possible to compare the overall QA and enforcement potential of the preferred alternative (linking waste classification and NSD requirements, plus adding scaling factors to make the system practicable) with that of a system that decouples waste classification and disposal requirements to an extent that permits GCD technology to play a constructive role. Then the possible institutional, economic, and QA penalties of the proposed action of the EIS can be more directly evaluated.

(3) The presumption that any NSD facility will be a manmade artifact whose hazard potential (particularly caused by various forms of inadvertent human reuse) might well outlive institutional control measures is a common feature of many governmental and private industry studies of shallow land burial regulation (Main Report, Sec. 4.2). NRC has commendably followed this lead. However, NRC has significantly limited the intruder scenario. This limitation should not be justified on the grounds that intrusion is "only hypothetical" (Main Report, Sec. 4.2). Artificial restrictions in the basic intruder scenarios, which have been used to set waste classification limits, include: (1) the intruder who builds a house cannot live in it (Main Report, Sec. 4.2.2.1); (2) the intruder who lives in a house cannot drink water drawn from a well on site or nearby (Main Report, Sec. 4.2.2.2); (3) the agricultural intruder cannot grow deep-rooted plants that would contact the wastes through remaining trench covers (App. G, 3.4.2); (4) the

agricultural intruder cannot dig stock ponds, septic tanks, drain fields, or utility trenches (App. G, 3.4.2); (5) the agricultural intruder cannot work or normally stay at home (for example, be a farmer, housewife, or child) (App. G, 3.4.2); and (6) the agricultural intruder cannot occupy the basement he is assumed to construct and thereby be exposed to gaseous releases into that space (for example, tritiated vapors, ^{14}C-labeled gases, etc.) (App. G, 3.4.2). The treatment of intruder contact with artifacts or relatively stable waste forms is unsatisfactory and unconvincing (Main Report, 4.3.4). The only justification given in this EIS for a lack of attention to the consequences of burial of stable waste forms containing high surface contamination or large concentrations of long lived radionuclides is simply that "it is not credible" that extensive human reuse of a NSD site or of extended contact or recovery of persistent waste forms buried in it would occur (Main Report, 4.3.4.3). The EIS should discuss the alternative of limiting NSD disposal to those wastes that, by nature of decay and dilution in trench materials, will not present a hazard to any inadvertent human reuse following loss of institutional controls, and disposing of higher activity and/or longer lived non-high-level waste in GCP systems.

(4) NRC has gone only part way toward bringing their radiological dose assessment methodologies up to date by switching from those used in Regulatory Guide 1.109 to the Task Group Lung Model. To the extent that ingestion doses continue to be based on this Guide (App. G, 2.4.2), they may be based on outdated data and assumptions. For example, current revisions in uranium ingestion dosimetry for environmental sources have resulted in EPA limiting uranium concentration in public drinking water to 10 pCi/ℓ (adopted in these proposed Part 61 regulations in Sec. 61.41). These are not reflected in the pathway dose conversion factors. These considerations also need to be taken into account in considering the disposal of natural and depleted uranium (which is in the proposed regulations up to the natural specific activity concentration, Main Report 7.2.2).

(5) In attempts to reproduce the NRC intruder scenario computation of concentration limits, it was found that, in the agricultural scenarios, plant uptake was based on the leaching of waste to the interstitial water and that only the fractions of radionuclides transferred from waste to water were assumed accessible to roots (Appendix G, 3.4.2). The procedures used in the

NRC calculation do not have a clear basis in the literature and moreover considerably underestimate plant uptake. The NRC calculations should be changed to reflect these considerations.

(6) If the pathway assumptions and models given in the draft EIS for the arid site are used, the calculated arid site concentration limits for plutonium are more restrictive than the generic site by an order of magnitude (data from App. J, Table J.5, methodology from App. G, Ch. 3). There is no technical basis for the NRC conclusion (Main Report, 7.2.4) that this consequence is adequately offset by consideration of differences in intruder behavior at arid sites. The NRC should set generic concentration limits based on the most limiting site conditions.

(7) The proposed flexibility reserved for deciding the final form of waste classification limits is objectionable if it is to be based on the kinds of arguments advanced in the case of ^{137}Cs (Main Report, 7.2.5). The quality of the source characterization data used in this EIS provides no real assurance that an additional dilution factor of 10 to 20 is warranted, particularly in light of the anticipated averaging and scaling practices to be used by waste generators.

(8) A more complete discussion of waste processing than is presented in Appendix G, Sec. 5 is needed. Emissions as a function of processing temperature, off-gas treatment systems, type of process equipment used, and operating conditions should be discussed.

(9) A proposed regulation in 10 CFR 61 requires separation of units so that there is "no interaction between them." The draft EIS should discuss whether this is possible in terms of hydrocarbon (such as methane) migration, ponding of water from subsidence, and subsequent movement into adjacent regions, etc.

(10) The draft fails to address all types of wastes that may need disposal before 2000. It would appear likely for example that at least some pilot plant reprocessing waste might need disposal. Because UF$_6$ facility wastes presently contain radium and thorium and because these will be present as daughters of uranium as the uranium ages, these radionuclides should be considered in the regulations as soon as possible. Wastes from the thorium high temperature gas cooled reactor fuel cycle also need to be considered.

90

Page Range	NTIS Price Code	Page Range	NTIS Price Code	Page Range	NTIS Price Code	Page Range	NTIS Price Code
001-025	A02	151-175	A08	301-325	A14	451-475	A20
026-050	A03	176-200	A09	326-350	A15	476-500	A21
051-075	A04	201-225	A10	351-375	A16	501-525	A22
076-100	A05	226-250	A11	376-400	A17	526-550	A23
101-125	A06	251-275	A12	401-425	A18	551-575	A24
126-150	A07	276-300	A13	426-450	A19	576-600	A25
						601-up*	A99

*Contact NTIS for a price quote.

Los Alamos