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ORNL/NFW-82/18

Contract No. W-7405-eng-26

DOE LOW-LEVEL WASTE MANAGEMENT PROGRAM  
NUCLEAR WASTE PROGRAMS  
OAK RIDGE NATIONAL LABORATORY  
Environmental Sciences Division

ORNL/NFW--82/18

DE83 003193

(Activity No. AR 05 15 15 0; ONL-WL01; WBS OR 3.1.1)

PROCEEDINGS OF  
THE FOURTH ANNUAL PARTICIPANTS' INFORMATION MEETING  
DOE LOW-LEVEL WASTE MANAGEMENT PROGRAM

Convened by  
The DOE Low-Level Waste Management Program

August 31-September 2, 1982  
Denver, Colorado

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Date Published - October 1982

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OAK RIDGE NATIONAL LABORATORY  
Oak Ridge, Tennessee 37830  
operated by  
UNION CARBIDE CORPORATION  
for the  
DEPARTMENT OF ENERGY



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## PURPOSE AND SCOPE

The Fourth Annual Participants' Information Meeting of the Department of Energy Low-Level Waste Management Program was held in Denver, Colorado, August 31 - September 2, 1982. The purpose of the meeting was to report and evaluate technology development funded by the program and to examine mechanisms for technology transfer. The following specific objectives were presented to guide formal discussion during technical sessions of the meeting:

- o Has the Department of Energy defined the problem?
- o Is the Program designed to answer the questions?
- o Is the Program being conducted properly?
- o Who needs the technology and how do we get it to them?

The meeting consisted of an introductory plenary session, followed by two concurrent overview sessions and then six concurrent technical sessions. There were two group meetings to review the findings of the technical sessions. The meeting concluded with a plenary summary session in which the major findings of the meeting were addressed.

One hundred thirty-seven registrants participated in the meeting. In addition to attendance by individuals from various levels within the Department of Energy and by contractors for that agency, there was representation and active participation by the Nuclear Regulatory Commission, the Environmental Protection Agency, the U.S. Geological Survey, and their contractors.

## WELCOME AND OPENING REMARKS

G.B. Levin  
DOE Low-Level Waste Management Program  
EG&G Idaho, Inc.  
Idaho Falls, ID 83415

Good morning, My name is George Levin and I would like to welcome you to the Fourth Annual Department of Energy Low-Level Waste Management Program Participants' Meeting.

For those of you who haven't read the agenda, I will tell you a little bit about the meeting. You're in for something a little different this year.

The Program management team got tired of working at these meetings, and we thought you might like to participate. So we have set up this meeting for you folks to work for the next two and one half days.

Tomorrow is your big day. You are supposed to listen and talk and offer ideas and suggestions. You'll note the agenda is a little different, kind of experimental, and I hope you enjoy it.

Other than that, you are going to see our program from stem to stern, almost everything in it. The Nuclear Regulatory Commission, the Environmental Protection Agency, and the United States Geologic Survey graciously agreed to participate in this year's meeting, so you are really going to see a cross section of what's going on in low-level waste technology development around the country. It's just about everything, I think, that's worthwhile.

There are only three of us who have been around since the beginning of the program. Of those who have been involved with the program longer than I really in the trenches, our keynote speaker today, Dewey Large, is the only one left.

Dewey Large is our keynote speaker. Dewey has survived the longest, so thought it appropriate that he kick things off today. He runs the program from the Oak Ridge Operations Office, and he's going to tell us what's going on today.





**DOE LOW-LEVEL WASTE MANAGEMENT PROGRAM  
PERSPECTIVE ON TECHNOLOGY TRANSFER:  
OPPORTUNITIES AND CHALLENGES**

**D. E. Large  
ORO Radioactive Waste Management Program  
U. S. Department of Energy  
Oak Ridge Operations Office  
Oak Ridge, Tennessee 37830**

**Greetings!**

It is my privilege and pleasure to assist in welcoming you and to help open the Fourth Annual Information Meeting of the DOE Low-Level Waste Management Program (LLWMP). An information meeting is held each year to afford the opportunity to share our work with our fellow participants. These meetings are designed to ensure that we all have a common perspective of the DOE Low-Level Waste Management Program, its purpose, accomplishments, expectations, and the synergisms of the parts fitting together.

We appreciate this particular occasion, arranged to bring together representatives of the DOE LLWMP Technology Team, interested agencies, states, industry, and other organizations involved in the management of nuclear waste. As we exchange information and deal with understanding and moving applicable technology to the points of application and implementation, we will focus upon what the Technology Program is, how we arrived at this point, what we produce, and how to market our products. The assessment of the current status of the overall Program, expectations, and mechanics for transferring technology will be revealed in a later address and treated in various forms and approaches in subsequent sessions and discussions. In keeping with the purpose of this meeting, this discussion will deal with the DOE LLWMP perspective in regard to transfer of LLWMP technology to current and potential users in both the Commercial and Defense sectors. Past, present, and future opportunities and challenges for the whole nuclear waste management community will be indicated.

Elements to be considered include: historical and evolutionary events and activities; the purpose of the Program and its inherent opportunities and challenges; achievements and expected accomplishments; supporters and interactors; packaging and delivering technology; implementing and serving potential users; determining and meeting users' needs; and identifying and responding to opportunities and challenges.

Several events and activities have brought us to this point in LLW management with basic roots extending back into our radioactive heritage in the Manhattan Engineering days. Other notable items in the sequence of our evolution reflect:

- attempts by specific generators and disposers to respond to defined and undefined technology needs.
- AEC Headquarters' formation of an organizational unit to fund and deal with field offices' research, development, demonstration, and operations in radioactive waste.
- Shallow Land Burial Steering Committee to advise the Headquarters unit on research and development needs, principally for Defense wastes.

Concurrently, commercial disposal sites were being established to accommodate radioactive wastes from commercial, institutional, and some federal sources.

The low-level waste management effort to improve shallow land burial technology and deal adequately with our special heritage was begun in FY 1977 and has expanded to include waste treatment and alternative disposal methods.

Administrative facilitation of the ERDA and DOE decentralized management policy resulted in the lead field offices for specific categories of radioactive wastes. This action led to designation of IDO/EG&G and ORO/ORNL as the Program Management Team for the DOE LLWMP initiated in 1979. The technology assessment, development, and demonstration efforts have involved eight field offices and their respective contractors. In the beginning of the LLWMP, the Shallow Land Burial Steering Committee assisted in determining technological gaps and issues in shallow land burial. The technology efforts of the Program have addressed those items along with identifying and considering feasible alternatives.

The Program is dynamic and responsive to adjustment, such as answering to two Headquarters' masters: "Commercial" and "Defense". Even the stated purpose of the Program, although in principle remaining the same, has varied in statement in order to be responsive to both Commercial and Defense interests.

The objectives of our Program continue to include: provide improved techniques for shallow land burial, including development for monitoring and corrective measures methods for low-level wastes; provide criteria for disposal; develop treatment processes to reduce volumes of waste generated and for improved handling; and develop disposal concepts alternative to shallow land burial.

The Technology Team, principally dedicated and competent participants, have taken advantage of opportunities and are meeting challenges. The technology focus has been upon shallow land burial for improved disposal of most low-level waste for both Commercial and Defense. Corrective measures for existing and new disposal areas have received much attention. Considerations of alternatives to shallow land burial have been concentrated into "greater confinement than shallow land burial".

Milestones have been established and are used as principal management control items.

This technology, the Program Product, is being described and made available to our customers.

This year, the Program has drafted criteria for inclusion in a DOE order for radioactive waste management operations at DOE sites. Meeting this challenge involved taking advantage of cooperation with potential users: the DOE contractor site operators.

Moving toward meeting our challenges has resulted in many notable achievements since our previous participants' meeting.

From the beginning, and I am one of the few people who have been around throughout the life of the Program, the Low-Level Waste Management Program has attempted to meet the needs of our customers. Our initial challenges and efforts were aimed at determining the customers needs and expectations. Some of you remember the survey conducted by the transuranic and low-level waste management programs concerning the technology needs of all the DOE facilities. The commercial customers were part of the strategy development process, and we continue to maintain active contact with these groups. Some of them are represented here for the second year, and hopefully they will continue to provide input to the Program. At this point, we believe that the basic needs of the customers are addressed in the Program and, although some fine tuning may be needed, the Program is oriented toward meeting these needs.

We hope that, here, those of you who are customers of the Program, in the sense that you may need or want to use some of our products, and those of you who are the participants in the Program, in the sense that you are the developers of the products, will exchange views on the various projects of the Program. We hope that we have provided the appropriate forum for some candid and frank exchange between these interest areas with the objectives of further refining our information on customer needs and fine tuning the Program to meet these needs. The Program staff is here to listen and learn from these exchanges.

The challenge and opportunity ahead of us all is the rapid and cost-effective applications of the emerging technologies, an activity which we call "technology transfer". The Program must make technology transfer a smooth and effective operation in order for our technology to be implemented.

"Technology transfer", which appears to many to be simple, is complex and very challenging. Even within the DOE system, it is not an every day occurrence for one facility to use applicable technology developed by another facility. In these times of tight budget and in the interest of good management, it is in our collective interest and to our mutual benefit to develop technology only once.

The Program and the Interim Operations Program in DOE Headquarters are working together to promote more interchange of information from the lead technology offices to the site operators and among the operators. We hope that by learning about technology work throughout the system, the various sites will use the results instead of undertaking new development projects themselves.

It is more difficult to transfer DOE technology to the commercial world because of certain institutional constraints and attitudes. A "free and easy" information exchange system between the Department and the commercial sector is not in effect. This has led to misunderstandings and difficulties in the past that do not contribute to the technology transfer process. Improved mutual respect between the two groups is needed so that communications can be enhanced. The Program has established contacts with the commercial sector to attempt development of technology transfer.

This meeting is intended to promote technology transfer by describing to each other and to our customers the work we are performing. We believe this interchange will be useful to our customers. Customers should examine the products and make certain the information is in a usable nature and an acceptable form. We are interested in any thoughts, ideas, and concepts that any of you have to bring about technology transfers. Our desire is that participants present their work in a professional and candid manner and that all of you will participate in frank and professional discussions of the work. We have scheduled times for discussion in order to receive your opinions of individual projects and the overall program so that we can improve both. We solicit constructive criticism and suggestions to improve the fiber and structure of the LLWMP.

We hope you will enjoy the meeting and learn much during the next few days.

4th ANNUAL PARTICIPANTS' MEETING  
DOE LOW-LEVEL WASTE MGMT. PROGRAM  
DENVER, COLORADO - 8/28-9/2/93

PERSPECTIVE ON TECHNOLOGY  
TRANSFER: OPPORTUNITIES  
AND CHALLENGES

## ELEMENTS TO BE TREATED:

- HISTORICAL AND EVOLUTIONARY CONSIDERATIONS
- PURPOSE OF THE PROGRAM; INHERENT OPPORTUNITIES AND CHALLENGES
- ACHIEVEMENTS AND EXPECTED ACCOMPLISHMENTS
- SUPPORTERS AND INTERACTORS
- PACKAGING AND DELIVERING TECHNOLOGY
- IMPLEMENTATION AND SERVICING
- POTENTIAL USERS
- DETERMINING AND MEETING USERS' NEEDS
- IDENTIFYING AND RESPONDING TO OPPORTUNITIES AND CHALLENGES

95. ANNUAL P&E GRANTS MEETING  
DOE LOW-LEVEL WASTE MGMT. PROGRAM  
DENVER, COLORADO - 5/29-31/92

PERSPECTIVE ON TECHNOLOGY  
TRANSFER, OPPORTUNITIES  
AND CHALLENGES

## DOE LLWMP TECHNOLOGY TEAM

- HEADQUARTERS
  - OFFICE OF DEFENSE WASTE AND BYPRODUCTS MANAGEMENT
  - OFFICE OF TERMINAL WASTE DISPOSAL AND REMEDIAL ACTION
- LEAD
  - ID/EG&G
  - OR/ORNL
- PARTICIPANTS
  - ALO: LANL, ML, RI
  - CHO: ANL (E&W), BNL
  - IDO: EG&G, USGS
  - NVO: FBD, REECO
  - ORO: ORNL
  - RLO: PNL, RHO
  - SAN: LLNL
  - SRO: SRL

4th ANNUAL PARTICIPANTS' MEETING  
DOE LOW-LEVEL WASTE ADAPT. PROGRAM  
DENVER, COLORADO - 8/28-9/2/82

PERSPECTIVE ON TECHNOLOGY  
TRANSFER, OPPORTUNITIES  
AND CHALLENGES

**A DYNAMIC PROGRAM TO PROVIDE APPROPRIATE  
TECHNOLOGY FOR RADIOACTIVE WASTE HANDLING  
AND DISPOSAL TO MEET COMMERCIAL AND  
DEFENSE NEEDS BY PROVIDING TECHNOLOGY FOR:**

- WASTE GENERATION REDUCTION
- WASTE TREATMENT, PACKAGING AND HANDLING  
FOR SHALLOW LAND BURIAL
- ESTABLISHING SHALLOW LAND BURIAL SITES
- CORRECTIVE MEASURES FOR SHALLOW LAND  
BURIAL SITES
- WASTE TREATMENT, PACKAGING AND HANDLING  
FOR GREATER CONFINEMENT THAN SHALLOW  
LAND BURIAL
- ESTABLISHING A SITE PROVIDING GREATER  
CONFINEMENT THAN SHALLOW LAND BURIAL



ANNUAL PARTICIPANTS' MEETING  
FOR LOW-LEVEL WASTE MGMT. PROGRAM  
DENVER, COLORADO - 8/2-9/2/92

PERSPECTIVE ON TECHNOLOGY  
TRANSFER, OPPORTUNITIES  
AND CHALLENGES

## **TECHNOLOGY PROGRAM PRODUCTS:**

### **LOW-LEVEL WASTE TECHNOLOGY HANDBOOKS**

- VOLUME 1 - BACKGROUND INFORMATION
- VOLUME 2 - SHALLOW LAND BURIAL
- VOLUME 3 - CORRECTIVE MEASURES FOR DISPOSAL SITES
- VOLUME 4 - DISPOSAL SITE MONITORING
- VOLUME 5 - WASTE TREATMENT
- VOLUME 6 - GREATER CONFINEMENT

DEL. 8/29/92

4th ANNUAL PARTICIPANTS' MEETING  
DOE LOW-LEVEL WASTE MGMT. PROGRAM  
DENVER, COLORADO - 8/31-9/2/82

PERSPECTIVE ON TECHNOLOGY  
TRANSFER, OPPORTUNITIES  
AND CHALLENGES

## **LLWMP TECHNOLOGY PROGRAM FY 1982 PRINCIPAL ACHIEVEMENTS (CONT.)**

### **WASTE GENERATION**

- PREPARED WASTE GENERATION REDUCTION MANUAL - RLO/PNL

### **WASTE TREATMENT**

- SET UP A DEMONSTRATION OF INCINERATION OF INSTITUTIONAL WASTE - IDO/EG&G
- DESIGNED A MEMBRANE PLANT - ALO/ML
- STUDIED APPLICATION OF MICROWAVE PLASMA TORCH INCINERATOR - RLO/RHO
- REPORTED ON VOLUME REDUCTION AND PRODUCT ACCEPTABILITY ATTAINED BY GLASS MELTER - ALO/ML
- INVESTIGATED SOLIDIFICATION - CHO/BNL
- DRAFTED WASTE TREATMENT HANDBOOK - ORO/ORNL
- STUDIED NITRATE DECOMPOSITION - ALO/RI(RF)

ANNUAL PARTICIPANTS' MEETING  
FOR LOW-LEVEL WASTE MGMT. PROGRAM  
DENVER, COLORADO - 8/31-9/2/82

PERSPECTIVE ON TECHNOLOGY  
TRANSFER, OPPORTUNITIES  
AND CHALLENGES

## **LLWMP TECHNOLOGY PROGRAM FY 1982 PRINCIPAL ACHIEVEMENTS**

### **SHALLOW LAND BURIAL**

- DRAFTED SLB HANDBOOK - ORO/ORNL
- DRAFTED ENVIRONMENTAL MONITORING MANUAL - CHO/ANL-E
- PREPARED SLB TEST REPORTS - ALO/LANL, ORO/ORNL, RLO/PNL, SRO/SRL
- STUDIED MICROBIAL EFFECTS IN SLB - CHO/BNL

### **CORRECTIVE MEASURES**

- PREPARED CURRENT PRACTICES HANDBOOK - ORO/ORNL
- DRAFTED CRITERIA FOR CORRECTIVE MEASURES - ORO/ORNL

DEL 8/31/82

4th ANNUAL PARTICIPANTS' MEETING  
FOR LOW-LEVEL WASTE MGMT. PROGRAM  
DENVER, COLORADO - 8/31-9/2/82

PERSPECTIVE ON TECHNOLOGY  
TRANSFER, OPPORTUNITIES  
AND CHALLENGES

## **LLWMP TECHNOLOGY PROGRAM FY 1982 PRINCIPAL ACHIEVEMENTS (CONT.)**

### **GREATER CONFINEMENT**

- DRAFTED METHODS FOR SELECTING SITES FOR GREATER CONFINEMENT DISPOSAL - SRO/SRL
- PREPARED TEST PLAN FOR GCDF EXPERIMENT - NVO/REECO
- PREPARED GCDF MONITORING PLAN INCLUDING DETAILED DESIGN - NVO/FBD
- INITIATED GCDF CONSTRUCTION - NVO

4th ANNUAL PARTICIPANTS' MEETING  
DOE LOW-LEVEL WASTE MGMT. PROGRAM  
DENVER, COLORADO - 8/21-23/82

PERSPECTIVE ON TECHNOLOGY  
TRANSFER, OPPORTUNITIES  
AND CHALLENGES

## DOE LLWMP CUSTOMERS

- DOE INTERIM WASTE MANAGEMENT
  - PLANNERS
  - GENERATORS
  - HANDLERS
  - DISPOSERS
  - CUSTODIANS
- COMMERCIAL SECTOR
  - GOVERNMENT
    - LOCAL
    - STATE
    - FEDERAL AS RELATED TO NON-DOE
  - INDUSTRY
    - PLANNERS
    - HANDLERS
    - DISPERSERS
    - CUSTODIANS

DEL 8/21/82

DEFENSE PROGRAM TECHNOLOGY TRANSFER  
NEEDS AND EXPECTATIONS

G.K. Oertel  
Defense Waste and Byproducts Defense Programs  
U.S. Department of Energy  
Washington, D.C. 20545

I would like to welcome all of you to the Fourth Annual DOE Low-Level Participants' Information Meeting. I think the sessions will be useful for sharing information among developers and DOE operators of sites, and of course, I'm delighted to see that the commercial side is represented here too.

I want to thank the Idaho Operations Office, the Oak Ridge Operations Office, and their contractors, EG&G and ORNL, for making the meeting arrangements. I hope the meeting will turn out to be useful for everyone.

I would also like to take the opportunity to introduce a new member of my staff whom you haven't met. He's the new director of the Research and Development and Byproducts Division. Texas Chee reports to him. His name is John Jicha.

My topic is the Headquarters' perspective of program needs and technology transfer. Technology has to be useful. Beyond convincing himself that the technology is useful, the developer must assure that the prospective user really believes that.

It's not enough for you in the laboratories to be convinced that this is useful technology. If your customers, the operators of the sites, in DOE or elsewhere, don't know about it, or can't afford it, they may not use it.

There is usually a world of difference between the people who develop technology and those who operate burial grounds. Different actors and institutions are involved. They have different priorities and perceptions of what is important. They are working under different constraints, often money. They need to get a job done, like managing so many drums of waste, putting them under the ground every week, or whatever. It is not easy to get them to do business differently.

We have been asked by our program review committee to look into this matter and to see whether the right people are talking to each other. I'm not convinced they are. That is the focus of this meeting.

As you know, waste management got separated into two parts, commercial and defense, a year ago. The cooperation has been excellent, particularly in low-level waste. We are using the same lead sites and the same lead contractors, and that helps assure that we don't duplicate technology development. It also helps assure that new technology will

be reviewed from the point of view of whether anybody is ready to use it. In fact, there are various efforts in place to demonstrate such use. Technology development, the work in the laboratory, be it in the National Laboratories or universities or wherever, tends to be funded by defense programs. The nuclear energy budget is being used to help the states and the industry cope with their new responsibilities under the Low-Level Waste Policy Act of 1980. Can this division be justified? It turns out that so much of the technology development is useful to our burial grounds. We, after all, have a lot more than the commercial sector. We feel no hesitation to fund most of the development of new technology because if it is useful anywhere, it will be useful on our sites.

At the same time, it is our contribution to the nuclear energy side's efforts to make the Low-Level Waste Policy Act work, and we are cooperating fully with the commercial side. What mechanisms are there for a smooth transfer of technology? The technical reports, participants' meetings and appraisals, and the existence of lead sites all have led to better communications. That doesn't necessarily assure that technology developed in one place is being used in another, but there is close cooperation in the development of technology and of operations among some sites. The working groups today will show if there are additional needs for technology transfer.

I was going to talk briefly comparing this situation with another one that many of you are familiar with. The role of technology development is much like the support that a tailor receives from the people that develop fibers, so that he can make clothing. If the tailor produces clothing that cannot be seen, then his technology support has failed him.

I hope that you will participate, make recommendations, and not be reluctant to tell us if you think we are going in the wrong direction. We'd like to make the system work.

Finally, I'd like to say that low-level waste management is mature. The waste is being disposed of in a very responsible, safe manner. I really do not know of any exceptions. Some past operations have left us with situations we wish were better, and we are doing something about them as part of the technology development job.

In high-level and TRU waste management, however, we have not yet dealt with inventories. We are very excited to finally have the opportunity, in this administration, to deal with them. The Waste Isolation Pilot Plant and certain processing facilities that will be built at Idaho will permit us to take care of much of the TRU waste that is in storage. The defense waste processing facility at Savannah River will immobilize the high-level waste there.

We like to think that low-level waste management is working well in technology development and in practice. If that is not really the case, then I hope you will make it happen.

**COMMERCIAL LOW-LEVEL RADIOACTIVE WASTE MANAGEMENT**

Joseph A. Coleman  
Acting Director  
Division of Storage and Treatment Projects  
U.S. Department of Energy

My purpose is to review the goals, objectives, and activities of the Department of Energy's Low-Level Radioactive Waste Management Program. I will touch on the new organizational structure of the Office of Terminal Waste Disposal and Remedial Action, where the program is located, briefly cover the current low-level waste (LLW) management situation and outline our continuing role in providing assistance to the states in this area. Institutional and technological needs related to LLW management have, for the most part, been identified, and I will discuss the Department's plan in meeting these needs. Finally, I wish to emphasize how our program priorities are designed to accomplish our primary objective.

The goal of the overall LLW Program is to support development of an acceptable, nationwide, near surface waste disposal system by 1986. To be effective the system should provide for all low-level waste materials -- whether defense or commercial -- to be managed in such a way as to protect public health and safety over the long-term. My remarks will focus on the commercial waste management system.

The reorganization of the Nuclear Energy programs under the Assistant Secretary, Shelby Brewer, has been completed. Franklin Coffman is Director of the Office of Terminal Waste Disposal and Remedial Action. Within this Office, I am the Director of the Division of Storage and Treatment Projects. My responsibilities, in addition to the commercial LLW program, include the West Valley Demonstration Project, and high-level and transuranic waste treatment programs.

At Headquarters, Betsy Jordan manages the commercial LLW program. Idaho Operations continues as the lead office for overall LLW program coordination and implementation. We recognize that technologies and procedures are being developed and implemented through Idaho according to the recognized needs of the defense and commercial LLW programs.

The commercial LLW program has two major functions: 1) application of the technology improvements for waste handling, treatment and disposal, and 2) assistance to states as they carry out their responsibilities under the Low-Level Radioactive Waste Policy Act of 1980.

In fulfilling these functional responsibilities, we have at the present time two principal customers: the States and the commercial generators of waste. The States have been given the primary responsibility for safe disposal and management of non-defense LLW. The



decisionmaking authority rests with the State governments. The Department's role is one of support for providing technical assistance, analytical systems, information, and other materials so that the States will be able to meet their new responsibilities efficiently. The commercial customer exists in that utilities and other users of radioactive waste materials generate a variety of low-level wastes. Technical improvements in LLW management will help these generators meet new regulatory requirements, provide protection of public health, and achieve more economical treatment and disposal operations. A third customer for our products will be those firms that will treat and dispose of the waste.

The basic policy structure for low-level waste management was set less than 2 years ago in December of 1980 with the passage of the Low-Level Waste Policy Act. The States and industry have moved forward with impressive speed to cope with the disposal problem. They are now in the analysis and planning stage wherein future management activities and facilities are being projected. The appropriate support for these activities may be categorized quite generally as data collection and analysis during which great premium is put on our efforts to accurately project the magnitude and priorities for resolving the LLW management problem over the next 20 years.

In the last 5 years the regulations concerning low-level waste disposal (10 CFR 61) have been developed and the disposal economics have undergone enormous changes. The result of these economic changes and the new regulations have been to make volume reduction far more attractive economically and likely to make physical containment of the waste mandatory. The commercial LLW generators have new technology needs and the Department's roles to assist in bringing these new technologies to the point where they can be readily adopted. These new technologies should meet the needs of the customers and should be economically attractive. You will hear detailed presentations on these technology development tasks over the course of this meeting.

The situation in commercial LLW management -- increased rate of waste generation, increased number of shipments, shortage of disposal sites, and the States' responsibility to provide for disposal capacity -- dictates that the Department deliver useful and appropriate products for use by the States and industry. Our customers will judge us on the quality, usefulness, timeliness, and our responsiveness in developing these products.

Your contributions to developing these products is a critical element in our efforts to resolve the low-level waste problem in this country. It is your work that will become the technical, analytical, and managerial resource that the States and private industry will rely on to meet their responsibilities for an effective LLW management system. We are currently in the forefront of day-to-day relations with the States and industry. Our performance in this role -- of providing

transfer of an improved LLW management technology -- will set the tone for other Department programs in the next 2 or 3 years.

We are committed to providing support to the States, a role identified in the Low-Level Radioactive Waste Policy Act and endorsed by the States, and to transferring technology developed for Departmental programs to the commercial sector. Information from our Defense Program activities is of significant value. You have the basic role of providing the technology for managing the Department's low-level wastes. Our role on the Nuclear Energy side is to adapt that technology for possible use by the private sector. We want to provide as much useful information as possible in the timeframe set by the Low-Level Radioactive Waste Policy Act.

To summarize, the priorities for the commercial side of the Low-Level Waste Management Program have been established to meet one goal: Do our best to support development of an effective commercial management system by 1986. The first priority is being given to supporting State efforts in forming the institutional structures needed to manage the system. A second, equally important, priority is the State and industry role in transferring and demonstrating treatment and disposal technologies.

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## INDUSTRY EXPECTATIONS OF THE DOE LLWMP

Max M. DeLong  
Northern States Power Company  
Minneapolis, Minnesota

## ABSTRACT

Time will be limited for regional commissions to become fully operational and site developers to license and open disposal facilities. DOE and its contractors have extensive experience and capability for LLW management that can be utilized. Priority activities for new site development, waste generation reduction, waste classification and decontamination residue disposal are discussed. DOE should confer or work jointly with organizations to provide needed information.

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INTRODUCTION

In reviewing the program outline for this conference, I am impressed with the amount of work underway related to the disposal of low-level radioactive waste (LLW). As one representative of those potential users of a commercial LLW disposal facility, it's encouraging to see a conference such as this that has as part of its focus the transfer of technology from basic programs and actual disposal experience to commercial LLW disposal operations.

I would be absolutely ecstatic if the conference had in its title words to the effect that these were all final report presentations on the various subjects. Then there would be only the task of implementing that information between now and January 1986 when the industrial/institutional segment must have access to additional disposal sites. When one examines a time line during which a number of significant accomplishments must be completed, there are not too many words that are useful in summarizing the situation. The need for action is at the very least, urgent when assessed optimistically and may border on panic when evaluated pessimistically.

For example, consider new commercial LLW disposal site access. There is a substantial amount of work that must be done before that time. It will be perhaps mid 1983 before many of the state compacts are ready to

go to Congress, and another period of time will be taken for Congressional review. What it means is that there are only two years for regional commissions to become fully operational and site developers to license and open facilities. Two years time for those kinds of activities to be completed are at best the very minimum. It will take a significant degree of cooperation between potential users of commercial LLW facilities, state and local governments and the DOE if that schedule is to be met.

I've organized my remarks for this morning in three different sections. First, I would like to provide a LLW disposal site user's perception of why we consider DOE to be in a very advantageous position and why we would like to see DOE utilize that position to the utmost. There is a vast amount of experience residing with DOE and its contractors on a variety of aspects dealing with low-level radioactive waste disposal, and put simply, we must capitalize on it. Secondly, I would like to discuss and focus on those activities that we think are needed to achieve operation of new commercial disposal sites. Third, I would like to discuss some of the characteristics of low-level waste management activities that are unique to this technological, high public visibility, government/industry effort.

#### DOE'S ADVANTAGEOUS POSITION

DOE has a vast amount of technical expertise and experience when it comes to managing LLW. For example, simply scan the topics that are being covered at this conference. The topics indicate there is activity underway in waste treatment, packaging, handling and disposal considerations. Although much of that work applies specifically to DOE and sometimes has unique characteristics, I think it's fair to say that most of the technical work that is underway does have direct application to the disposal of commercial waste.

But besides the obvious technical expertise and first hand experience, the DOE LLWMP participants are in a unique position to be approached by and share with all the other parties involved in the efforts to developing new commercial low-level waste disposal sites. Don't underestimate the uniqueness of this aspect. Federal Law 96-573 placed a substantial responsibility on state government but only a few states have had direct prior experience with LLW disposal. Further, as compact committees deliberate and as compact commissions are formed they need a source of information so that they can conduct the needed activities. They have available to them information from waste generators and site developers and others, but often that is only a part of the information they require in order to conduct the activities they have underway for the formation of regional compacts.

Third, there are R and D programs which can provide some of the fundamental information that waste generators need as this new era of low-level waste disposal gets underway. As DOE continues to conduct

these programs, you will be in contact with both users as well as regulators and the results of those programs thus will have the benefit of the input that will satisfy both user requirements and regulator requirements. DOE as an independent party, with actual experience, has a unique role to fulfill.

## PRIORITY ACTIVITIES

Considering the short time frame for having new commercial disposal sites in operation, that is between now and January of 1986, the country simply does not have the luxury of continuing to study various approaches or alternatives to the various tasks associated with disposal of low-level radioactive waste. The fact that the country has had successful low-level waste disposal sites in operation for a number of years is sufficient demonstration that adequate technology and regulation is available and being practiced. The real focal point of industry's concern is that there must be access to additional disposal sites by January 1 of 1986. In order to have those sites operational by that time there are a number of specific tasks that need to be done and ones which DOE must play a key role.

## NEW SITE DEVELOPMENT

The following discussion is predicated on DOE seizing the initiative and 1) backfitting an existing DOE LLW site, or 2) opening up a new DOE site, or 3) at the request of a regional commission, work toward developing a new commercial site. Since the topic of the conference is the DOE LLWMP, with emphasis added to MANAGEMENT, it would seem the following suggestions are quite workable even though not necessarily originally envisioned. It's not that regional commissions couldn't do it, it is a matter of expediency.

### Organization and Management

Even though a regional commission could put all the organizational aspects into effect, the operation of a commission would surely be expedited if sample detailed management and programmatic materials were in existence. If DOE were to start with the viewpoint of a regional commission and prepare the necessary material even if it were for an existing DOE site or a new one, the task could start immediately. Until a compact has had Congressional review and approval, a regional commission will not be fully functional. The difficulty is that when Congressional review is completed, a commission will have a minimum of time to complete these organizational tasks. If sample by laws, organization manuals and regulatory guidelines were ready, the process of having a site operational by January 1986 would be realistic. Sound presumptuous? Well, maybe, but then the situation calls for innovation. DOE and its contractors could look outstanding by simply extrapolating on existing experience. Do it

cooperatively as a joint effort and the technology transfer takes place as the project proceeds. This effort could start now.

### Site Selection

The task of site selection for a LLW disposal facility in a new region is certainly the most public and political visible process that a regional commission will be involved in. The effort is going to take skillful leadership on the part of a commission. However to make the process most efficient, it would help to have a benchmark siting study conducted in accordance with latest siting techniques and regulations.

Site selection techniques and guides exist now either in descriptive form, such as the soon to be released EPRI handbook, or in the experience embedded in a number of organizations. In either case, an immediate effort ought to be set in motion to exercise and test their approaches. Again it could be applied to an existing DOE site, a new DOE site, or if a regional commission finds it appropriate, it could work jointly with DOE and others and focus on a specific new region. The important thing is to have tried and tested approaches and sample final work products available when regional commissions begin these tasks.

### WASTE GENERATION REDUCTION

Most of the compacts will probably have as a basis the intent to strive towards reduction of waste generation and/or volume reduction. Waste generation reduction is probably the most cost effective approach. DOE is in a unique position to review both their own position and those of industry and identify which are cost effective in reducing the waste stream volume.

Volume reduction of plant waste residues before disposal is going to continue to be a topic of much interest. What LLW shippers need on this topic is for DOE to add their perspective on cost benefits and public acceptability considerations for candidate options. Given the relatively fixed cost structure of opening and operating regional disposal facilities, smaller annual disposal volume may not reduce the total disposal costs.

### WASTE CLASSIFICATION

The proposed 10CFR-61 regulations that will be out soon have very definitive requirements for waste classification. Waste classification has been extended beyond just the identification of the source or waste stream involved. For waste streams such as those from power reactors, the task of sampling and measuring concentrations of key nuclides and inferring or developing correlations for describing remaining nuclide content could be a significant one. With DOE having a number of waste

streams, application of this approach could provide perspective on workable approaches on this topic. This would provide both industry and regulators with insight on adequate approaches.

#### DECONTAMINATION TECHNOLOGY

As nuclear plants and other facilities generating radioactive waste begin to mature, there will be increasing decontamination efforts on a variety of equipment and structures. DOE and their contractors, having actual experience, and planning new decommissioning jobs (such as the ETR reactor) can provide very realistic data this industry can use in similar efforts in the future. How the residue can be processed for disposal and what site conditions would have to be present for acceptance of this type of material are all questions that require answers.

#### SIGNIFICANT LLW DISPOSAL PROGRAM REQUIREMENTS

I highlighted in preceding sections what results from many of the items to which I have attached priority, are urgently needed. The five years provided by Congress for development of new commercial LLW sites is almost 40% gone. Significant progress has been made but there is much yet to be done. The way DOE contributes to the program over the next three years is crucial.

Reports covering the management and administrative aspects of regional facilities would be very timely if available before the end of 1983. If those reports were the result of a joint effort between DOE and others, it would be very beneficial. Exercising and demonstrating existing LLW disposal facility siting techniques is also urgently required. A joint effort between DOE and others would add a certain degree of realism even if it pertained only to DOE's property.

Programs addressing waste generation reduction, volume reduction, chemical cleaning, waste classification and decommissioning must reflect actual industry situations. DOE should confer with industry or undertake joint programs to provide the necessary realism for eventual application of the results.

Finally, with programs such as LLW disposal, that have high public visibility, demonstration not explanation is the key persuasive element. DOE has an opportunity to utilize its expertise, its facilities and its Congressional mandate to participate in moving the nation's LLW disposal activities from the conceptual stage to reality.

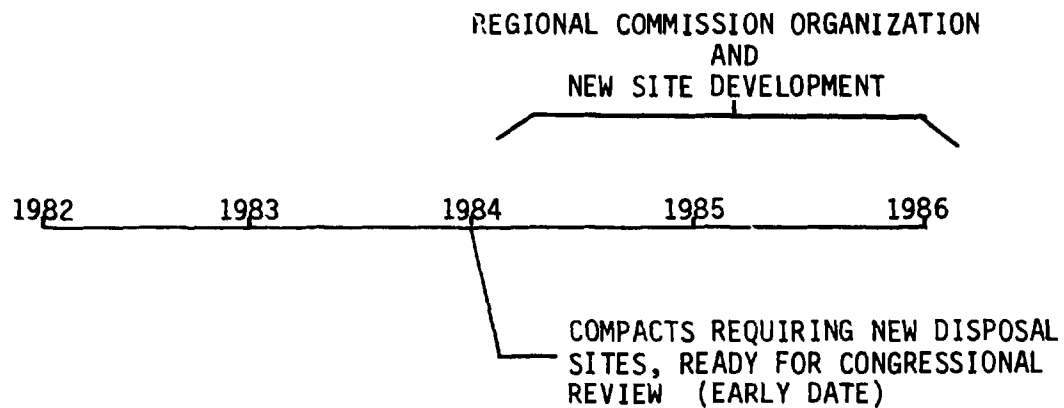


4th ANNUAL PARTICIPANT'S MEETING  
DOE LOW-LEVEL WASTE MGMT. PROGRAM  
DENVER, COLORADO - 8/31-9/2/82

INDUSTRY EXPECTATIONS OF THE  
DOE LLWMP

# TIME CONSTRAINTS

(FIGURE 1)



4th ANNUAL PARTICIPANTS' MEETING  
DOE LOW-LEVEL WASTE MGMT. PROGRAM  
DENVER, COLORADO - 8/31-9/2/82

INDUSTRY EXPECTATIONS OF THE  
DOE LLWMP

## EXPECTATIONS

(FIGURE 2)

- UTILIZE AN ADVANTAGEOUS POSITION
- GIVE PRIORITY TO TASKS FOR OPENING  
AND OPERATING NEW DISPOSAL SITES
- RECOGNIZE THE SIGNIFICANT PROGRAM  
REQUIREMENTS

4th ANNUAL PARTICIPANTS' MEETING  
DOE LOW-LEVEL WASTE MGMT. PROGRAM  
DENVER, COLORADO - 8/31-9/2/82

INDUSTRY EXPECTATIONS OF THE  
DOE LLWMP

PRIORITY TASKS

(FIGURE 3)

NEW SITE DEVELOPMENT

WASTE GENERATION REDUCTION

WASTE CLASSIFICATION

DECONTAMINATION TECHNOLOGY

NRC EXPECTATIONS OF THE DOE  
LOW-LEVEL WASTE MANAGEMENT PROGRAM

E.F. Hawkins  
U.S. Nuclear Regulatory Commission  
Washington, D.C. 20555

It is a real pleasure for me personally to be here this morning for this meeting. It is a culmination of some rather quick activities on our part over the last few months. I have very high hopes on what I can learn here at this meeting about the DOE work, and I also hope that this will be the start of many ways in which DOE, NRC and the other agencies will interact on all low-level waste programs.

This is, as has been mentioned in passing this morning, however, not the first step in NRC-DOE interactions. We have been meeting on a regular basis for several years. Four agencies in particular, NRC, DOE, EPA, and USGS have been discussing areas of mutual concern in low-level waste for some time. That interagency group sponsored a workshop here in Denver about two years ago on modeling the low-level waste. Since then, we have also continued to have a series of discussions trying to coordinate and figure out what each agency is doing and how we can coordinate those programs more closely.

NRC and DOE have also had a series of informational exchange meetings where we have brought together DOE and NRC contractors. In those, so far, we have covered waste treatment, waste form and shallow land burial technology. Some of you here have participated in those. We plan to do more of these types of meetings in the future, in specific technical areas where we think we can both benefit.

Last year at this meeting in New Orleans, Dale Smith discussed the proposed 10 CFR Part 61 regulations and John Stewart discussed, in general terms, the NRC program in low-level waste and the attendant program support and research contracts that we have underway. That was the extent of our participation last year.

This year, as you are aware, NRC contractors are also participating, although I will admit to a rather modest extent. I take the blame for that, partly. We didn't really decide until rather late in the game that we were going to participate. And we weren't able to really get our contractors on board in time to participate. That wasn't their fault; it was really mine.

Hopefully, if this works out as well as I think it will, we will have a full year to plan how we can participate in this meeting next year.

By the way, as I mentioned, and several of you have discussed this morning or asked me questions about, there have been some changes in NRC. Dale Smith, who gave you the talk last year on 10 CFR Part 61 is

no longer in the Low-Level Waste Licensing Branch. Dale has been named the new director of the Denver Field Office for Uranium Recovery Licensing, and he's already in that role. He will physically be moving to Denver sometime within the next year along with about half of the Uranium Recovering Licensing Branch people that are now in Washington, D.C. Ross Scarano, who was heading the Uranium Licensing in Washington, is the new acting branch chief in low-level waste. Ross planned to be here this week to try to get caught up on what was going on. Unfortunately, Ross was badly needed in Washington to carry on some uranium recovery work, so he was not able to make it. I am sure you will be hearing more about, and from, Ross in the future.

The objective of having NRC participate in this meeting is to coordinate, to the extent possible, the NRC and DOE programs; to take advantage of each other's work and expertise; to avoid any unnecessary duplication and to try to identify any gaps in the programs.

As far as the printed title of this talk, "NRC Expectations of the DOE Low-Level Waste Management Program," I think the use of the word "expectations" is well chosen. George Levin cautioned Mr. DeLong to say things diplomatically. Well, I will say something diplomatically, but it's going to be rather blunt too. NRC is not relying on any of the DOE programs to accomplish our missions. I think that why will become clear as I go along. However, let me emphasize now, that from what we know of DOE programs we expect to see a lot of useful and essential information that we will use to build upon and use in our programs in the future.

NRC's contractual efforts can be broken into two areas; confirmatory research to support our regulations and program support to assist us in performing licensing actions. Examples of each kind are represented in the talks from our contractors over the next couple of days.

NRC does have expectations of the DOE program. In general we're looking to the DOE programs to resolve information that can be used by the states, the compacts and industry in all aspects of low-level waste disposal. We also expect DOE to pretty well carry the ball, almost exclusively from the federal standpoint, in performing research that extends to the state-of-the-art. We don't see that as a part of the NRC mission.

If we break down the low-level waste disposal program into general areas, we can discuss in more specific terms what our interests and directions are and where we are looking to DOE for research, development, demonstration and technology transfer.

First, in the areas of siting and site characterization. In terms of site selection, NRC has said very little about and is supporting very little work that could be called site selection. Any guidance that is provided we expect to come from DOE and consultants, to applicants, states and compacts. Our regulations do not specify how a site is

selected. We are less concerned with how a site is selected but rather that the selected site meets the minimum technical requirements and to assure that the performance objectives are met. We realize that help is needed by states, etc., in this area and a lot of questions will arise on site selections and guidance. But we believe that those should be provided by others.

For site characterization, that's a different story because we must be able to make independent licensing decisions on the acceptability of proposed sites. To be able to do this we need good, reliable, comprehensive information on the site characteristics. Because of this we have funded many studies on methods that could be used to characterize the sites and, again, those will be illustrated over the next couple of days. The studies are directed at providing NRC with the ability to evaluate the types of field and lab work that are needed, the accuracy of the measurements, their reliability and what can be learned from the measurements and analyses. As you can see, that would be somewhat different from the DOE program, in the same area. We developed a technical position paper on site selection, characterization and suitability and that provides some preliminary guidance in these areas. Our programs are all directed at enabling us to evaluate the sufficiency and completeness of a site characterization program and providing us with the necessary tools to determine site suitability.

What do we expect of the DOE program? We don't think that we are relying on any of the DOE programs to provide information that we need to evaluate site acceptability. However, we would expect advances in the state-of-the-art to come from DOE and private industry. We also expect, as Mr. DeLong talked about, that we could learn a lot from how DOE has, and is, characterizing their own sites and learning both from the mistakes and successes.

Next let me discuss the general areas of facility design and operations. 10 CFR Part 61 also has minimum technical requirements in these areas. And, we have several projects underway to assist us in evaluating proposed facility design and operations. We are also preparing a technical position paper that will be ready probably within this month in draft form. And, we are sponsoring a symposium in this area at the end of September in Washington, D.C. Our interests here are to be able to evaluate independently proposed designs and operations to assure that the minimum technical requirements are met and the performance objectives are also met.

Our expectations of the DOE program would be in developing and demonstrating these technologies. With many years of experience at hand, we would very much like to have information on the design and operations of your sites. And we do have a lot of this information, I don't mean to imply that we don't have any information. We do. This information will be very helpful to us in building a data base that we could use to develop and test evaluation procedures.

In the areas, that I will call generally "properties of low-level wastes," Part 61 has minimal technical requirements on waste form and/or waste containers. And our interest is to be able to evaluate proposed waste forms and has supported quite a bit of work in this area. We also have an interest, of course, in waste treatment and waste generation techniques and volume reduction methods. Our interest here is in knowing just what the properties of such waste will be so we will be able to analyze their performance in a disposal site environment.

Probably more so at present than the others, we are expecting quite a bit of useful information from the DOE program. For instance, the data base system that EG&G is developing, the incineration studies and the waste form and treatment studies should prove very valuable to us and very useful. Frankly, we would like to know much more about the programs. For some we have already initiated some coordination. I must emphasize again, we are looking at these programs as providing useful but not essential information. Since many of your programs will take several years to complete we can't rely on them.

In the area of alternatives to near surface disposal our programs are really just getting started. Our plans call for looking at both those wastes that are not suitable for near surface disposal because they have high activities and/or concentrations, and those that are not of regulatory concern -- that is, those that have such a small hazard that they need not be disposed of in a commercial low-level waste site.

We have followed the DOE program in greater confinement disposal, have received several reports on it, and have had several discussions with many of the principal investigators in this area. We are hopeful that we can utilize this information in the future. That program, as most of you know and you will hear about later, is pretty much just getting started. However, since our schedule calls for publishing amendments on alternatives to near surface disposal within the next year, we can't really rely on that program to give us information to feed into that task. What we are looking at is a potentially valuable demonstration of the technology.

Let me emphasize that although I have discussed the above areas in separate parts, we are really viewing it as a whole system consisting of the site, the facility design, the operation and the wastes that will be disposed. Our primary purpose is to assure that the performance objectives of Part 61 will be met. There are, of course, minimum technical requirements, and in those areas our review process will first determine that they have been met.

Again, so I don't leave you with the wrong impression, we believe that the DOE program is essential, important, and useful and will provide the needed guidance and information to states, compacts and the industry. We are, however, not dependent on your program. This is primarily because, as I have tried to illustrate, our missions and timing are somewhat different. Our mission is to be able to evaluate

proposed low-level waste disposal facilities and to make regulatory decisions. The DOE program is obviously not directed in the same way. We do recognize where the similarities are and where we can benefit from your program. If I didn't think so we would not be here. I think that we have worked very well together to date. What we want to do is make the same thing true between you, who are actually doing the work, the contractors and researchers who are doing the research and technology development. That's why we have supported, enthusiastically, this meeting. We hope that what we start here will develop into a closer relationship in the future.

Let me cover one other subject quickly. What is the status of Part 61? Part 61 has been submitted to the Commission for action. We have reviewed all comments received, analyzed them, and incorporated them into Part 61 and the Environmental Impact Statement. It was submitted to the Commission early this summer and there has been one briefing before the Commission on it. Based upon what we know now, optimistically, we believe that Part 61 will probably be published as a final rule within the next couple of months.



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THE U.S. ENVIRONMENTAL PROTECTION AGENCY'S  
PROGRAM FOR THE DEVELOPMENT OF LOW-LEVEL  
RADIOACTIVE WASTE DISPOSAL STANDARDS

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OBJECTIVES OF THE PROJECT

The U.S. Environmental Protection Agency has been involved in the examination of low-level radioactive waste practices since the Agency was created in 1970. We have only recently gotten under way, however, with a delineated active program for establishing standards and/or guides for such disposal. Although many facets of this program have been initiated, it has not been approved in its entirety at the Agency level.

Beyond the generic purpose of protecting the public health and safety, we intend to (1) establish appropriate limits on radiation exposures to the public for low-level waste disposal and (2) determine if there are exposures related to disposal of extremely low-level radioactive waste that are sufficiently small that they do not need to be regulated as to their radiation hazard and such that the wastes can be disposed of with minimal controls. Such limits would be applicable only to the low-level waste disposal operation itself, it would not apply to auxiliary functions such as waste treatment, packaging, or transportation.

The wastes that the EPA standards program would apply to are presently defined by exclusion. That is, they would apply to all radioactive wastes except: high-level radioactive wastes and high-activity transuranic wastes covered by proposed EPA "Environmental Standards and Federal Radiation Protection Guidance for Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes;" uranium mill tailings; and radioactive wastes regulated under the Resource Conservation and Recovery Act of 1976. The Low-Level Waste Standard is intended to handle disposal of all Atomic Energy Act materials not covered by the other standards.

As our development efforts continue, we plan to maintain an ongoing dialog and exchange of information with the NRC to assure that our directions do not diverge. Meetings, like this one, where staff and related contractors from all the involved agencies get together to exchange information are invaluable in this respect. We are also engaged in an interagency headquarters based information exchange program which is most valuable. Only recently, staff representatives from four agencies (Environmental Protection Agency, Nuclear Regulatory Commission, Department of Energy, and the U.S. Geological Survey), met to discuss needs and future directions in the modeling of releases and transport pathways for low-level radioactive waste disposal.

Our intention to set low-level waste disposal standards was formally established in 1980 with the Presidential endorsement of the proposed plan of the Interagency Review Group on Nuclear Waste Management. This plan charged EPA with issuing low-level radioactive waste disposal standards.

The desire for EPA to establish such standards has also been expressed by the States through the Presidentially established State Planning Council on Radioactive Waste Management and by the members of the Conference of Radiation Control Program Directors. We have committed ourselves in congressional testimony to do the job, and the Government Accounting Office has expressed interest in the need to complete this work.

#### AUTHORITIES FOR EPA STANDARDS AND GUIDES

In the process of creating the Environmental Protection Agency the Presidential Reorganization Plan No. 3 of 1970 transferred two authorities to EPA which are potentially applicable to low-level waste disposal.

The first authority, which came from the Atomic Energy Act of 1954, as amended, is for "generally applicable environmental radiation protection standards." Such standards could establish numerical limits on radiation exposure or levels, or concentrations or quantities of radioactive materials, in the general environment outside the boundaries of locations used for low-level radioactive waste disposal. This authority was used to set EPA standards for uranium fuel cycle facilities (40 CFR 190). Its use is restricted to Atomic Energy Act materials.

The second authority, which came from the former Federal Radiation Council (42 U.S.C. 2021(h)), is for the Administrator to "advise the President with respect to radiation matters, directly or indirectly affecting health, including guidance for all Federal agencies in the formulation of radiation standards." This authority is the same one that the Federal Radiation Council used to issue its reports, numbers one and two, which established the basic limits on exposure used by the Federal agencies today. We used this authority to promulgate joint Environmental Protection Agency--Food and Drug Administration guidance for diagnostic x-rays.

Both authorities can be used for setting exposure limits but the Presidential Guidance approach would also allow the Agency to give more general admonitions and direction on the "how to."

We have not decided to use either one of these authorities to the exclusion of the other. We plan to keep our options open as we continue our analysis. It may be that we would want to use both authorities for different aspects.

Before leaving the area of authorities, let me briefly run down the other EPA authorities that are being applied to various aspects of radioactive waste disposal and explain why there are differences.

EPA standard-setting efforts for high-level radioactive waste have been developed using the same authorities as we have discussed here.

Standards for uranium mill tailings disposal are specifically authorized and mandated by the "Uranium Mill Tailings Radiation Control Act of 1978." This Act requires EPA to establish criteria for disposal of both designated inactive tailings and tailings at active mills.

The Ocean Dumping Act of 1972 gives EPA the specific authority to issue permits for ocean disposal of radioactive waste and to certify dump sites for this purpose.

For radioactive materials not covered by the Atomic Energy Act and for other toxic materials, we can control disposal activities under the Resource Conservation and Recovery Act. This Act contains specific exclusions for Atomic Energy Act materials and for activities controlled under the Atomic Energy Act. EPA and NRC are working

together to resolve the applicability of this law to nonradioactive toxic wastes when they are mixed with low-level radioactive waste under NRC license.

#### ISSUES IN SETTING A LOW-LEVEL WASTE DISPOSAL STANDARD

There are, of course, numerous issues or questions to be resolved in this standard-setting activity. I will highlight a few of the ones that we have identified at this time, realizing that we will probably uncover many more in the days to come.

The major questions are related to the primary objectives of the activity. First, what is the appropriate and justifiable allowable exposure for members of the general public from low-level radioactive waste disposal operations? The second is whether there is some level of exposure related to low-level waste disposal, below which regulation is unwarranted, or below which regulation can be minimal, and if so, what is that level?

The form of the standard will receive a good deal of attention during its development. The standard must both accomplish the desired radiation protection goals and it must lend itself to practical application. Whether it is best expressed in terms of annual maximum individual exposure, lifetime dose commitment, radionuclide release rate, total radionuclide release, or other terms will be determined in the course of the analyses.

How to handle other hazardous chemicals contained in low-level radioactive waste is a question that many have asked. We do plan to examine the use of the Resource Conservation and Recovery Act to see if it can be applied or at least adopted by reference. The EPA work group which has been assigned to participate in developing the standard includes participation from the Office of Solid Waste which will be helpful in this analysis.

Another decision that must be made is whether the standard should apply to all low-level waste disposal facilities, both existing and new, or just apply to new facilities or license renewals. We also want the opinion of both the Nuclear Regulatory Commission and the Department of Energy on this matter. One key factor in this decision will be evaluating how dependent the standard may be on the site selection, initial design, and construction of the facility.

## THE TECHNICAL ANALYSIS

During the rest of this week you are going to be hearing from some of the contractors and others who are assisting us in the development of this standard. Most of these activities are just recently underway, so not much can be expected in the way of results. With the exception of the Oak Ridge National Laboratory effort, they will primarily be presenting plans and perhaps some progress. In order to have some perspective of context within which to evaluate these presentations, I will outline some of the primary things that must be accomplished for us to set a standard for low-level radioactive waste disposal.

We must be able to estimate the risk resulting from low-level waste disposal as it has been conducted, as it might be conducted without our standard, and with our standard in place. To provide a mechanism for such risk assessment, Oak Ridge National Laboratory has been working with us to develop a dynamic model for near surface disposal.

Because a "generally applicable standard" must not be method specific or dependent, TRW will expand our risk estimation procedures to encompass other disposal methods such as deep well disposal and hydrofracturing. Our initial evaluation indicates there are at least eight land based disposal methods that must receive some analysis. These are engineered surface storage, sanitary landfill, conventional and improved shallow land disposal, intermediate-depth disposal (greater than 10 meters), deep geologic disposal in a mined cavity (greater than 100 meters), hydrofracturing, and deep well injection. Not all of these methods will be analyzed to the same extent. If one method of disposal for a particular class of waste can be shown to meet the standard cost effectively, then it would not be necessary to analyze other more costly methods for that waste class. The U.S. Geological Survey has agreed to assist by evaluating our hydrogeological models.

Our analyses will evaluate the relationship between the reduction of risk through the application of technology and the cost of accomplishing that reduction. This cost-effective analysis will be carried out by TRW for a series of combinations of model facilities and appropriate waste streams. This will be a key element in the decision process of setting a standard. The U.S. Geological Survey will assist in this effort by providing site data and descriptions for the model facilities.

As an adjunct to this cost-effective analysis, TRW will also perform a series of analyses on waste form, treatment, transportation, and related cost factors.

As part of its compliance with Executive Order 12291 EPA must include in its overall analysis an evaluation of the economic impact of various levels of a standard and a determination of whether any entities, industrial or institutional would be severely impacted by the standard. Initial work on this effort is being performed for us by Energy Resources Company.

All of the above analyses must be consolidated and incorporated in a Draft Environmental Impact Statement to support the proposal of a standard. This will be followed by public hearings and subsequent issuance of a final standard after appropriate modifications.

I have chosen to leave out of this presentation any discussion of the intra-agency formalities and procedures that are necessary to the process in EPA. There are also external clearances monitored through the Office of Management and Budget. The intent of these checks and reviews is to develop as complete, consistent, and valid a standards package as possible.

#### EPA EXPECTATIONS FROM THE LLWMP

We have been asked to include as part of these presentations some assessment of the assistance we may need from DOE and its Low-Level Waste Management Project (LLWMP) contractors. The program I have described emphasizes the conduct of environmental assessment and cost-effective analysis.

To do either of these tasks well, we need detailed technical information about:

- o The quantities and characteristics of low-level wastes.
- o Engineering design, operational, closure, and cost information on the disposal methods available.
- o Reliable data for use in our assessment models; and, so far as possible.
- o Validation and verification of our models.

Fortunately, most of the information which we need is already available and only needs to be collected, collated, and put into a proper format for our analyses. The NRC produced an extensive body of information on the characteristics of low-level waste and on a shallow land disposal methods to support the development of its 10 CFR 61 licensing requirements. Also, extensive technical information on low-level waste management and disposal has been published by DOE contractors and the national laboratories under DOE LLWMP. We have, however, identified several gaps in the information we require for our analyses; a brief discussion of these follows.

Information on the quantities and characteristics of certain types of low-level wastes is very limited. Much is available on reactor and fuel cycle wastes. However, very little technical or inventory information is readily available on the physical, chemical, and radiological characteristics of commercial, institutional, and industrial low-level wastes. DOE's high-level wastes have been described in detail but very little has been published on the characteristics of their low-level wastes. We have already formally requested information from DOE on their own wastes. These two groups of low-level wastes will be significant both in our consideration of a generic environmental standard and in our determining if there is a level of exposure related to low-level waste disposal, below which regulation of the waste is unwarranted or can be minimal.

We have identified a number of methods, in addition to shallow land burial, which are potentially suitable for disposing of low-level wastes and which we believe should be considered in our analyses of cost-effectiveness. However, several have not been well documented for this use. Specifically, published information is limited on hydrofracture, deepwell injection, and intermediate depth disposal methods. We need additional information on engineering design, operations, closure, and costs of these disposal methods. We have included this need in our previously mentioned request to DOE.

Our assessment model for the shallow land disposal method is in the final stages of development by ORNL. TRW has just begun modifying it for use with other land disposal options. We will have to begin using these models for our environmental impact and sensitivity analyses as soon as they are completed to meet our planned schedule. Although we have the utmost confidence in the capabilities of ORNL and TRW, it



is not prudent to use any systems type model as large as the ones they are developing without extensive testing, comparison, and validation.

We have already asked the U.S. Geological Survey, through an interagency agreement, to critically evaluate the hydrogeological portions of our model. We are aware that NRC has been evaluating some of the commercial disposal sites. Likewise, the DOE management has had assessments conducted for disposal facilities on some of their reservations. We believe that a critical review and comparison of our new models with their own assessments by these groups would be very helpful.

Lastly, we can use help in our efforts to identify and categorize those low-level wastes which do not need to be controlled because of their radioactive hazard. We believe this is a project of great interest to all of us here. Our work plans on this particular effort are not complete yet and we do not know specifically how NE, DOE, and the national laboratories can help us on this effort yet. However, staff level discussions with representation from both agencies have indicated their willingness to assist. As soon as our plans are complete, we will be contacting the appropriate program offices.

## DOE LLWM PROGRAM MANAGEMENT EXPECTATIONS

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I would like to thank the previous speakers and accept their challenge to us to attempt to improve the process we refer to as technology transfer. You, the participants, have been involved from the inception of the program and know that we have tried to be responsive to the needs of our clients and that we have conducted these reviews to improve our program. The consensus both in this country and abroad is that technology exists to dispose of low-level waste and our challenge is to provide timely transfer of information. We have learned to better refine our understanding of the performance of the disposal system and to practically apply the innovations we develop. These needs are to establish a reliable low-level waste management system by 1986; and to develop regulations and technology to provide safe and cost effective management of the Department's low-level waste from Defense Programs.

In my talk I would like to cover the following points:

- o Summarize our program efforts;
- o Respond to some recommendations from last year's session;
- o Provide an overview of our schedules for technology transfer; and
- o Touch upon some of the progress we have made on improving technology transfer.

The low-level waste program is structured to provide solutions, for example:

Commercial Thrust

- o The program assisted most of the regional compact groups by providing financial support or technical assistance.
- o We have assisted some states in establishing their plans by making program staff available for consultation with the state and by working directly with states in developing management plans as was done in the case of North Carolina and Tennessee, or

by providing funding for the region to develop plans as was the case in Colorado.

- o We have provided funding for Rocky Mountain West, Midwest, and Northeast regions, as well as to the Southern States Energy Board for assistance in the Southeast and Plains regions, and have worked with Tennessee and Pennsylvania to develop siting process documents.
- o We are developing information management systems to provide computerized information on a real-time basis on waste volumes and waste characteristics. The Southern States Energy Board and Conference on Radioactive Control Program Directors are assisting us in developing methods to obtain better manifest data on a state level. We are continuing to pursue the development of cost-effective volume reduction techniques and are funding a project to develop a liquid waste treatment process; these technology efforts will be documented in a handbook which should be available in early next year.

### Defense

The defense program objectives are to continue to pursue the development of criteria, develop cost effective technology in the area of waste treatment, waste generation reduction, shallow land burial, and evaluation of greater confinement technology.

In last year's meeting several recommendations were made concerning the development of criteria:

- o That we attempt to minimize inconsistencies between DOE and NRC requirements
- o That we establish de minimis criteria for solid waste
- o That we recognize the difference in each site's waste characteristics and operations and allow flexibility for each site to implement its own special requirements.

In response to these recommendations, we have worked with the NRC during the comment process of 10 CFR 61 and believe that the NRC has done a commendable job in bringing the regulation to its present stage.

The draft DOE criteria are not completely consistent with all the NRC requirements; for example, the NRC excludes cardboard containers from disposal. However, the rationale for excluding these containers is stated by the NRC to be the operational consideration of possible exposure of site operators due to damage in transit. When DOE wastes are shipped within the site boundary we should not have the same problem.

The concern for de minimis criteria is reflected by the addition of the Solid Waste Environmental release limits to our criteria development efforts. In response to input from both the participants at last year's meeting and other contacts outside the program, we are evaluating an approach to a low-cost high-integrity package. We are also reviewing our program on waste form evaluation and have a laboratory program at Brookhaven and field studies with actual forms at Savannah River and plan similar tests at Richland, Washington.

We have not been able to incorporate all the suggestions into our plans. For example, we evaluated the possibility of conducting mock licensing reviews at a DOE site, and concluded it was not feasible. We have, however, attempted to deal with the problem of validating parts of the various models used for low-level disposal. The USGS has agreed to compile and evaluate the data bank of geotechnical technical data, and PNL is providing an assessment of various models and guidelines to assist the program in determining how to validate the various models.

Our approach in establishing LLW management guidance as shown in the next viewgraph has been to provide the basic requirements in the DOE order and to augment the order with the criteria and evaluation factors required to meet the requirement/criteria in the order and to follow-up this process with a series of technical handbooks. This process should provide the necessary level of safety and still provide flexibility for sites to evaluate site specific needs.

The next viewgraph shows the major milestones for the program. The criteria and greater confinement areas support defense programs and data systems support the commercial programs. The other technical activities areas include both defense and commercial projects. The projects and tasks in each of these areas will be covered in more detail later in the workshops.

The low-level waste criteria are being revised and each field office should receive a revised draft of the LLW Order and criteria reflecting the latest HQ guidance next month. The de minimis criteria and greater confinement criteria are being developed in consultation with contractors from the various DOE sites as we did with the near surface criteria and should be available next year. We are continuing to develop a waste classification system and expect to publish a conceptual approach later this year.

In addition to the actual demonstration and tests we are performing in the technology areas we are producing handbooks. These handbooks are shown as major milestones.

As many of you are aware, we have included preparation of early versions of these documents in our plans with later drafts reflecting what we have learned from 1982 to 1984. Copies of the waste reduction handbook prepared by PNL and the Environmental Monitoring Handbook prepared by Argonne are in the final review and printing stage and should be available later this year. We involved industry and state personnel in the peer review of the Environmental Monitoring Handbook and I would like to thank those of you who participated in that process and particularly Jake Sedlet of Argonne for a job well done.

In the shallow-land burial task we are planning to provide assistance to states and one of the defined tasks is assisting Pennsylvania in developing a siting process. In data base activity the Southern States Energy Board and Conference of Radiation Control Directors are working with states to develop a computerized manifest system.

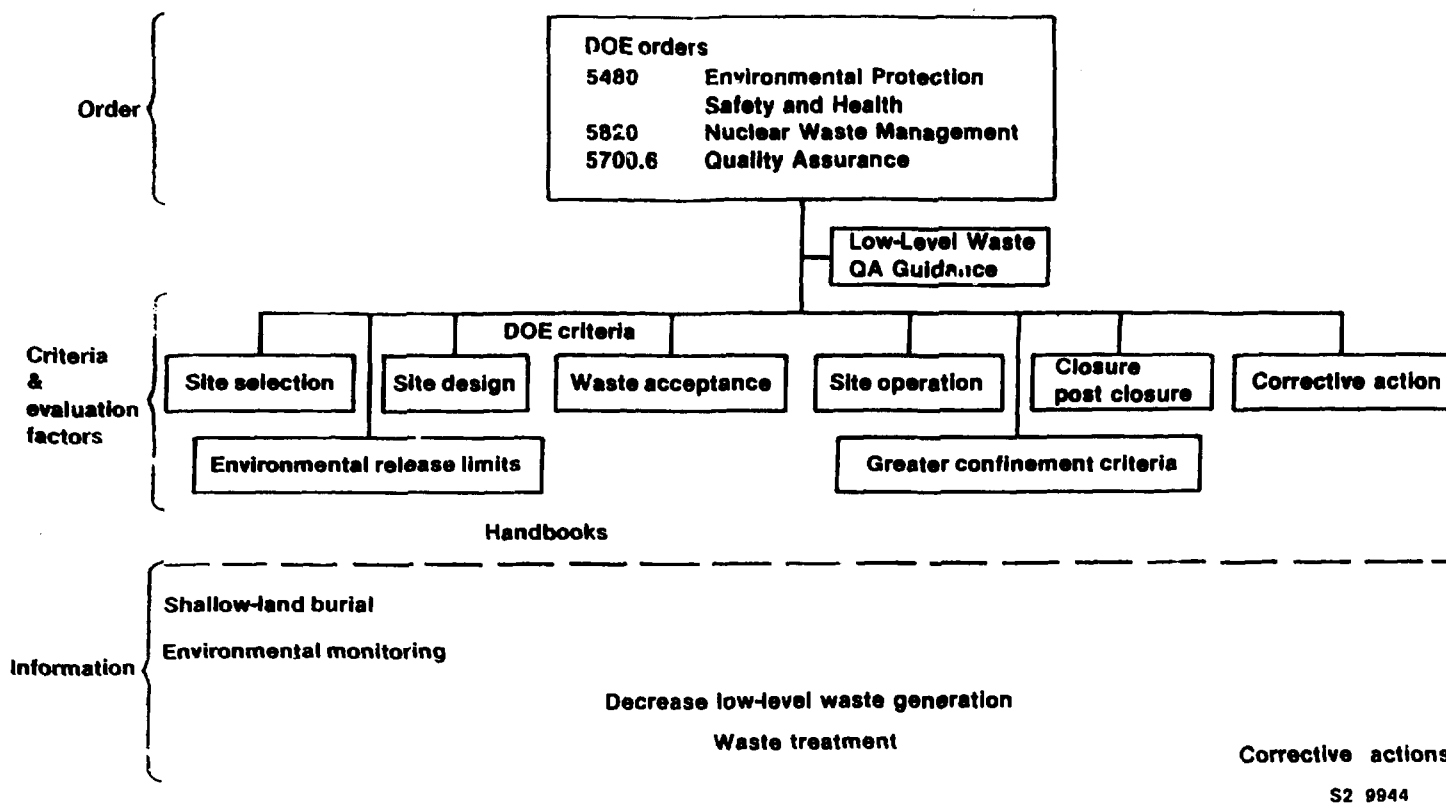
The program is moving to address the concerns raised last year of technology transfer. We recognize the need to improve our efforts in this area and will continue to work with states, industry, as well as the DOE laboratory to promote technology transfer. Each of you are encouraged to continue to share your experiences. Tom Hakonson of LANL has done a particularly good job and I would like to use some of Tom's work as an example: Tom's research is to develop empirical shallow land burial technology and he has built several demonstrators at LANL. He has been visiting the DOE sites to describe his research in shallow-land burial technology, and has participated in various forums with DOE, the NRC, and the commercial sector by taking visitors on tours of the LANL test facility.

Betsy Jordan of HQ has worked for the past year to obtain a distribution list for LLW technology documents and we now have an approved list UC-70B, of 287 names including state, industry, as well as governmental organizations. Texas Chee and the defense program staff have been instrumental in convening an ad hoc group of DOE contractors to review the LLW program with the objective of better assisting the DOE laboratories.

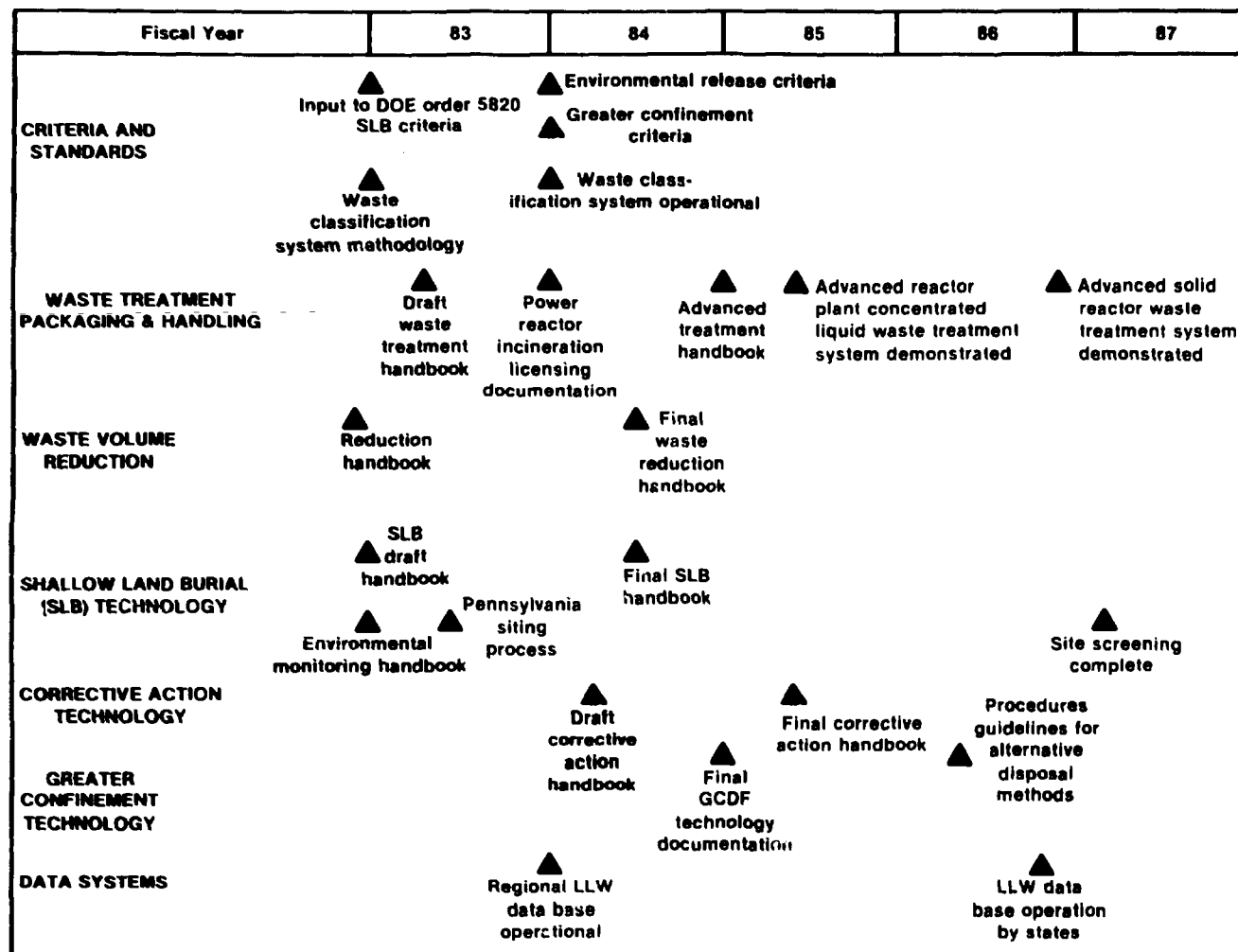
The internal operations activities of the Department can assist us as Rockwell Hanford has done by publishing the results of site specific tasks and making them available to others. We will require your assistance to make this process work. You will have to help us by developing your research program to meet the needs of a broader audience, to review the reports of other and by providing feedback to the program when you don't understand what we are asking for and to work more closely with the technology integrators of the ORNL staff. We believe in the Peer Review process and have requested that our

contractor developing the Handbooks to involve the users outside DOE (such as states, industry, and industry representatives) in the review process, and that the final submission to the program include the results of this review. The Peer Review Process will add this broader perspective to what we produce. I want to echo Dewey's challenge to all of us to work together in improving technology transfer and encourage all of you to consider the needs of other programs as you participate in the workshops.

# LLW Management Guidance



## Low Level Waste (LLW) Program Schedule





OBJECTIVES OF THE  
FOURTH ANNUAL DOE LLWMP  
PARTICIPANTS' INFORMATION MEETING

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Ladies and gentlemen. We have just heard the speakers at this morning's Plenary Session describe the expectations of their organizations with respect to technology transfer from the DOE-LLWMP. These papers set the stage for the remainder of the meeting.

Purpose and Objectives of the Meeting

The purpose of this year's meeting is somewhat different than that of previous year's meetings when the emphasis was primarily on reporting and evaluating the DOE LLWMP technology development activities. This year, in addition to the evaluating the status of program technology development activities, the scope of the meeting is expanded to include identifying mechanisms for transferring program developed technology to potential users. Specifically, the objectives of the meeting are (1) to determine if the program has comprehensively identified the problems, gaps, and issues associated with the management of LLW; (2) to determine if the program as structured addresses the identified issues; (3) to determine if the specific tasks being performed address and resolve the issues and if they are technically sound, (4) to identify information or technology developed by the program that should be transferred to potential users, (5) to identify the potential users of the information or technology, and (6) to identify the most appropriate mechanisms for transferring the information or technology.

Organization of the Meeting

In order to accomplish these objectives, we have organized the meeting into three types of sessions: (1) plenary sessions this morning and Thursday morning, (2) technical overview sessions this afternoon and Thursday morning, and (3) a series of concurrent, technical sessions all day Wednesday.

This morning we discussed the need for technology transfer, the objectives of the meeting, and the "rules of the game" for the conduct of the meeting. Technical session chairmen will summarize the conclusions and recommendations resulting from the meeting in the final plenary session on Thursday morning. This afternoon's technical

overview sessions will provide background technical information necessary to support and stimulate Wednesday's more specialized discussions. Panel discussions will be held at the end of the day and should address (1) whether or not the program has comprehensively identified the problems, gaps, and issues; (2) whether or not the program as structured addresses the problems, gaps, and issues identified; and (3) what information or technology developed by the program is ready and suitable for transfer to potential users, who are the potential users, what form should the information be in, and what mechanisms are most appropriate for technology transfer.

The concurrent, technical sessions are designed to function as workshops on these specific topics. Technical highlights will be presented to a small group to encourage more inspection, evaluation, and discussion of the specific tasks. Once again, the questions to be answered are (1) does the task address an identified problem, gap, or issue and (2) is the work technically sound. Papers, grouped by topic, will then be followed by panel discussions in order to provide a broad basis for evaluating a technology and to discourage focusing discussions on a specific task. Discussions in the panel sessions should focus on (1) determining if the program as structured addresses the identified problems, gaps and issues, (2) identifying information or technology ready and suitable for transfer to potential users, (3) identifying potential information and/or technology users, and (4) identifying mechanisms for transferring the technology. The results of these smaller sessions will be synthesized and summarized for presentation and discussion at two group sessions beginning at 8:30 Thursday morning. The two groups will be the same as those planned for this afternoon on A) Waste Treatment, Packaging, and Handling and B) Waste Disposal.

The final reports from the two groups emphasizing specific recommendations on technology development and transfer will be presented at the plenary session beginning at 10:30 Thursday morning.

#### Audience for the Meeting

The program has discussed technology development and transfer needs with the federal agencies and their contractors, state representatives, and industry representatives in the past in order to make the program responsive to the needs of the potential users of the program outputs. Because we wanted this meeting to serve as a forum for discussions between researchers and potential information users, we have brought together the full spectrum of participants in LLW management including researchers, regulators, and operators representing several federal agencies, the states, and private industry. Representatives of the Nuclear Regulatory Commission (NRC) and Environmental Protection Agency (EPA) have already indicated what their expectations regarding outputs of our program are and how the federal LLW program fits together. In addition, we heard what the states and industry need and expect relative to technology transfer.

More detailed presentations on specific components of the NRC, EPA and USGS programs will be presented in the concurrent technical sessions.

Bringing the various federal agencies together on a program like this certainly enhances the timely exchange of information and the cross-fertilization of ideas still in the formative stages. But we as a program really want more than that. We want the users in the states and commercial sector as well as the various federal agencies to tell us and each other what they expect from the DOE LLWMP, when they need it, what form it should be in, and whether the program is being responsive to those needs. We believe that by bringing this group together we have created an opportunity to have a meaningful exchange of information on LLW management between information generators and users.

At the end of this meeting, the results of these discussions will be reported to the program and evaluated and factored into the program planning process in much the same way as the results of last year's workshops influenced program activities. This is your chance to identify what problems really exist, what information and technology is really needed and how should program outputs be transferred to potential users.

Meeting these objectives will allow us to more effectively plan the future of this program. These discussions will also allow researchers to develop proposals which are more responsive to program needs, and allows all participants to have a direct and positive impact on where the program is going.

#### Closing Comments

By now, it should be clear to you whose meeting this really is. The DOE LLWMP with the cooperation of many others arranged the meeting, brought you all together, and provided a forum for your discussions. We have provided copies of the technical highlights where possible for your use during the meeting and for your reference until the proceedings are available. We will help the technical session chairmen document the results of the meeting and we will print and distribute the meeting proceedings. But the actual work, the discussions and exchange of information, can only be accomplished by the participants. I hope no one thought that this meeting was going to be low-key or relaxing. Please approach this meeting aggressively and with the desire to accomplish the objectives.



OVERVIEW OF DOE LLWMP  
WASTE TREATMENT, PACKAGING,  
AND HANDLING ACTIVITIES

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This paper presents an overview of Department of Energy (DOE) Low-Level Waste Management Program funded activities on the treatment, packaging, and handling of low-level wastes. Other papers in the Group A sessions will present the details of this work.

The waste management system is summarized in Figure 1. The system approach is demanded here because what can be done at each step is heavily influenced by what has been done previously and will in turn affect what can be done in later steps. The choice of a waste treatment method must consider not only the nature of the waste to be treated, but also the effect of the treatment process on packaging requirements and the restrictions on waste form that may be imposed by transportation and disposal requirements. Throughout the system the choices are interlocked.

A major purpose of the program is to reduce the commitment of burial ground space while protecting the human health and safety in a cost effective manner. This cost effectiveness requires balanced decisions at each point in the waste management system. For instance volume reduction is an appealing way to reduce the cost of packaging transportation or burial, but the volume reduction process invariably produces secondary waste which must cycle back into the system. The cost of handling secondary waste could destroy the cost-effectiveness of an otherwise attractive treatment scheme. Only experience from actual operations or from engineering-scale demonstrations can tell us enough about the secondary waste problems to clearly indicate that a given treatment process is cost-effective.

The final product of the DOE Low-Level Waste Management Program will be documentation of the various options available for each step in the waste management system. This documentation will permit each responsible site whether government or commercial to tailor a waste management system that is safe and cost-effective for the waste mix unique to that particular site.

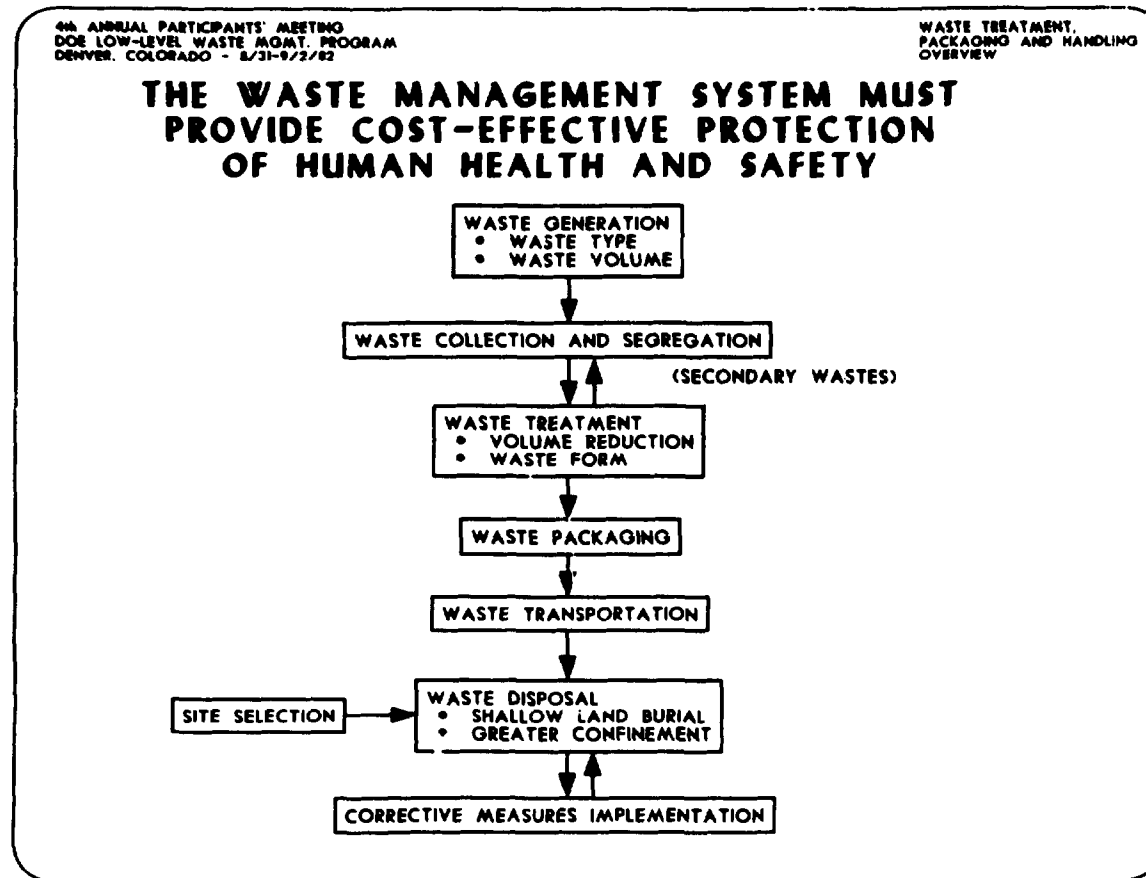


Figure 1.

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The efforts in the waste treatment, packaging, and handling technology include the selection and documentation of techniques for reducing the quantity of waste generated. The efforts also include the development and documentation of improved techniques for processing low-level waste to reduce its volume and increase the stability of the resulting waste form. This includes any techniques for reducing nonradioactive hazards associated with the waste form. The program is also involved in the investigation of waste packaging to improve transportation safety, increase disposal site stability and contribute to waste retention after disposal. The review and documentation of waste handling techniques is a part of the program since proper handling will reduce the potential for radiation exposure to both operating personnel and the public.

Three top-level objectives have been defined for technology development: (1) to develop and document the technology for waste generation reduction by September 1984; (2) to develop and document the technology for waste processing, packaging, and handling for shallow land burial by March 1984, and (3) to develop and document the technology for waste processing, packaging, and handling for greater confinement than shallow land burial by September 1985.

The technology development work is focused on development and demonstration applicable to the entire low-level waste stream from generation to disposal. The major components include waste generation reduction, waste processing, waste packaging, waste handling and waste form testing.

The waste generation reduction work has resulted in the preparation of a final draft of the Waste Generation Reduction Handbook by Pacific Northwest Laboratory (PNL) this year. No further activity is planned for this area until FY 1984 when the draft handbook will be reviewed and any necessary updating or revisions will be made.

The types of waste processing can be described as:

- o mechanical processing which includes baling, shredding, compaction or any mechanical treatment which makes the waste easier to package or to further process,
- o incineration which includes those processes which involve burning the waste, but as a more general term is sometimes used to include the digestion process,
- o chemical processing which includes ion exchange and biological treatment,

- o metal volume reduction which includes sawing objects into pieces suitable for packaging and melting and casting metal waste into solid ingots,
- o waste form development which includes the preparation and characterization of candidate waste forms especially the various solidification techniques, and
- o general waste processing guidelines which document the various procedures sufficiently to provide guidance to the individual waste generator.

The waste packaging studies can be divided into general packaging guidelines, the transport package and the disposal package. The transport package is essentially completely devoted to the protection of the operating personnel and the public from any release of radioactive material during transportation. The design of the transport package addresses such off-normal situations as resistance to impact and fire and is not closely related to the properties of the waste. The disposal package is however in direct contact with both the final waste form and the disposal medium. In some cases such as in-drum mixing of solidification agents, the disposal package is directly involved in the waste processing. Thus the disposal package and the guidelines pertaining to it are of greater interest in LLW treatment technology.

Waste handling can be divided into general guidelines, generator site handling, transport handling, and disposal site handling. Here, as with the packaging, the interest for waste treatment technology is primarily the handling which takes place at the generator site and immediately precedes or follows or is part of waste processing.

Waste testing involves the development and documentation of general testing guidelines, the definition of treatment methods (i.e., the characterization of the waste form produced), the development of standardized test procedures and the determination of whether there is a need for testing center for quality assurance. This work is at an early stage, but it is obvious that one cannot compare candidate processes without information about the nature of the waste form produced. The proper operation of a waste treatment process will also almost surely require some minimal information about the product of that process. The requirement for testing centers for quality assurance has been suggested, but is not yet resolved.

Several state-of-the-art reports were prepared early in the program to provide baseline information and guidance to the development work. (1-3) These reports assisted in the identification of critical technology areas which most needed continued study or development work:

- o The reduction of the volume of waste generated which has already been mentioned.

- o The incineration of organic liquid wastes which essentially destroys a waste form that is difficult to handle in other ways.
- o The incineration of combustible solids which is applicable to a large fraction of the low-level waste and which provides a large volume reduction factor.
- o Metal waste volume reduction which applies to a waste type not amenable to other treatment methods.
- o Waste solidification which is needed for incinerator ash, aqueous concentrates and miscellaneous wastes.
- o Waste handling and segregation which is a necessary precursor to most other waste processing operations.
- o And finally, treatment of special solids prior to disposal which includes the removal of metallic sodium or the destruction of nitrate in salt wastes.

These critical technology areas have been the subject of studies or development tasks at the various participating sites around the country. The critical areas and participating sites are shown in Figure 2.

In the area of waste generation reduction, both Oak Ridge National Laboratory (ORNL) and Rockwell Hanford Operations (RHO) provided documentation of their on-site practices early in the program. PNL work has more recently provided comprehensive documentation of waste generation reduction practices in the Waste Generation Reduction Handbook.

RHO is developing a microwave plasma incinerator for the incineration of organic liquids. This task will complete hot bench-scale tests early in FY 1983 and will begin the assembly of a larger unit. Both EG&G Idaho and Savannah River Laboratory (SRL) are developing the incineration of organic liquids in multipurpose incinerators. The emphasis of the work under the purview of EG&G Idaho is on institutional incinerators which are being tested at the University of Maryland and at Purdue University to determine the practicality of such waste treatment on a small scale by individual institutions. The incinerator at SRL, known as the Solvent/Solids Incinerator Facility Test (SWIFT), is expected to handle both solids and liquids. The glass furnace at Mound incorporates the ash into a bed of molten glass thereby providing volume reduction, fixation and solidification simultaneously. The cold demonstration and product acceptability tests are expected to be completed in FY 1983.



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## STATUS SUMMARY OF WASTE TREATMENT, PACKAGING, AND HANDLING TECHNOLOGY ACTIVITIES

| TECHNOLOGY AREA                  | ANL-W | BNL | EG&G | MOUND | ORNL | PNL | RFO | RHO | SRL |
|----------------------------------|-------|-----|------|-------|------|-----|-----|-----|-----|
| WASTE GENERATION REDUCTION       |       |     |      |       | ●    | ●   |     | ●   |     |
| INCINERATION LIQUIDS             |       |     | ●    |       |      |     |     | ●   | ●   |
| INCINERATION SOLIDS              |       |     | ●    | ○     |      |     |     |     | ●   |
| CONC. & TREATMENT : LIQUIDS      |       |     |      | ●     | ●    |     |     |     |     |
| METAL VOLUME REDUCTION           |       |     | ○    |       |      |     |     | ●   |     |
| SOLIDIFICATION (TESTING)         |       | ○   |      |       |      |     |     |     |     |
| HANDLING                         |       | ●   |      |       | ○    |     |     |     |     |
| TREATMENT OF SOLIDS FOR DISPOSAL | ○     |     |      |       |      |     | ●   |     |     |

● SOME RESULTS AVAILABLE

○ TASK UNDERWAY

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Figure 2.

For the concentration and treatment of liquids, Mound is investigating ultrafiltration and reverse osmosis. Ultrafiltration removes very fine particulate matter while the reverse osmosis removes material from solution. The biological treatment of liquids was investigated at ORNL both for the destruction of nitrates and the concentration of heavy metals.

Metal volume reduction at EG&G consists of melting stainless steel scrap and casting it into ingots. At ORNL, work was directed toward both decontamination of low-level metal waste and volume reduction by melting. At RHO the arc saw and vacuum furnace were tested for volume reduction of metal waste.

Waste form development work at Brookhaven National Laboratory (BNL) includes identifying and investigating potential solidification agents and processes and testing and evaluating the resulting waste forms.

BNL has also prepared a study of waste handling methods for low-level wastes. The Waste Treatment Handbook which has just been completed in draft form at ORNL contains chapters on both waste handling and waste packaging as well as waste treatment.

Argonne National Laboratory - West (ANL-W) is currently investigating the preparation of sodium contaminated waste for shallow land burial. The present rules do not allow the burial of waste with any degree of contamination by metallic sodium, so both removal methods and the amount of residual sodium are of interest. The Rocky Flats Plant is investigating the destruction of nitrate in low-level waste streams. Concerns relate to the prevention of the generation of oxides of nitrogen in the off-gas and the conversion of the waste to a soluble form.

In summary, the program objective is to develop the best available technology for waste treatment, packaging, and handling to meet the needs of shallow land burial disposal and for greater confinement than shallow land burial. The program has reviewed many of the hardware options for appropriate usage with low-level waste, but promising options remain to be evaluated. The testing of treatment technologies with actual radioactive process wastes has been initiated. The analysis of the interaction of treatment, solidification and disposal needs to be completed.

## REFERENCES

1. Kibbey, A. H. and H. W. Godbee, 1980. A State-of-the-Art Report on Low-Level Radioactive Waste Treatment, ORNL/TM-7427.
2. Gilbert/Commonwealth, 1980. State-of-the-Art Review of Radioactive Waste Volume Reduction Techniques for Commercial Nuclear Power Plants, ORNL/SUB-79/13837/2.
3. Pitt, W. W. Jr., H. W. Godbee, C. H. Hancher, B. D. Patton, and S. E. Shumate II, 1980. Survey of Needs for Biological Treatment of Radioactive Low-Level Liquids in Nuclear Facilities, ORNL/NFW-79/76.

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# **OVERVIEW OF DOE NATIONAL LOW-LEVEL WASTE MANAGEMENT PROGRAM**

## **WASTE TREATMENT, PACKAGING AND HANDLING TECHNOLOGY ACTIVITIES**

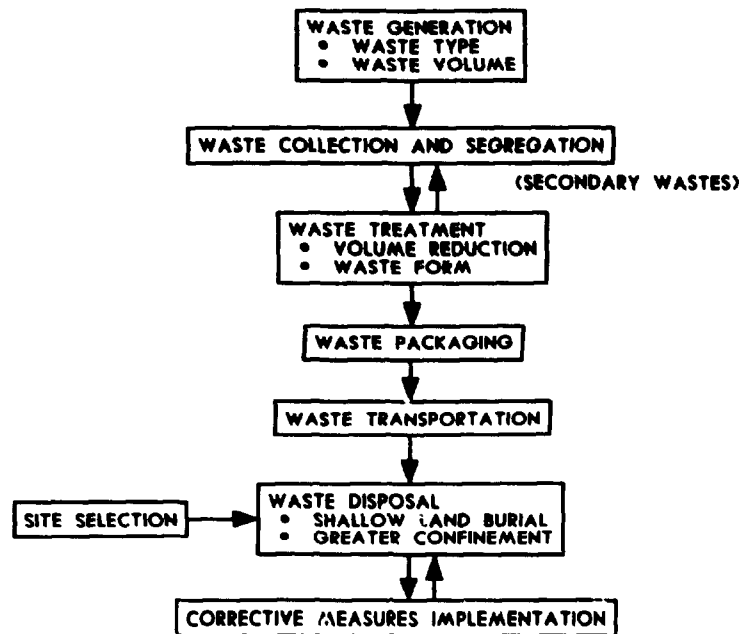
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# THE WASTE MANAGEMENT SYSTEM MUST PROVIDE COST-EFFECTIVE PROTECTION OF HUMAN HEALTH AND SAFETY



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## **WASTE TREATMENT, PACKAGING, AND HANDLING TECHNOLOGY ACTIVITIES INCLUDE THE RELATED AREAS OF:**

- IMPROVED TECHNIQUES FOR REDUCING THE QUANTITY OF WASTE GENERATED
- IMPROVED TECHNIQUES FOR PROCESSING LLW TO REDUCE ITS VOLUME; INCREASE THE STABILITY OF WASTE FORMS; AND REDUCE OTHER NON-RADIOACTIVE HAZARDS ASSOCIATED WITH THE WASTE
- PACKAGING FOR LLW IN ORDER TO IMPROVE TRANSPORTATION SAFETY; DISPOSAL SITE STABILITY; AND WASTE RETENTION AFTER DISPOSAL
- HANDLING TECHNIQUES FOR USE WITH LLW IN ORDER TO REDUCE THE POTENTIAL FOR RADIATION EXPOSURE TO OPERATING PERSONNEL AND THE PUBLIC

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## OBJECTIVES

- DEVELOP AND DOCUMENT TECHNOLOGY FOR WASTE GENERATION REDUCTION (9/84)
- DEVELOP AND DOCUMENT TECHNOLOGY FOR WASTE PROCESSING, PACKAGING, AND HANDLING FOR SHALLOW LAND BURIAL (3/84)
- DEVELOP AND DOCUMENT TECHNOLOGY FOR WASTE PROCESSING, PACKAGING, AND HANDLING FOR GREATER CONFINEMENT THAN SHALLOW LAND BURIAL (9/85)

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## **WASTE TREATMENT, PACKAGING AND HANDLING TECHNOLOGY**

- FOCUSED ON THE DEVELOPMENT AND  
DEMONSTRATION OF TECHNOLOGY  
APPLICABLE TO THE ENTIRE  
LOW-LEVEL WASTE STREAM FROM  
GENERATOR TO DISPOSAL
- MAJOR COMPONENTS OF THIS  
EFFORT INCLUDE:
  - WASTE GENERATION REDUCTION
  - WASTE PROCESSING
  - WASTE PACKAGING
  - WASTE HANDLING
  - WASTE FORM TESTING



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## **WASTE GENERATION REDUCTION INVOLVES METHODS TO ASSIST IN REDUCING GENERATION OF LOW-LEVEL WASTE**

- MAJOR COMPONENTS OF THIS EFFORT INCLUDE:
  - PREPARATION OF WASTE GENERATION REDUCTION HANDBOOK (INCLUDING STATE-OF-THE-ART ASSESSMENT; COST BENEFIT ANALYSIS; AND IDENTIFICATION OF PROBLEMS, GAPS, AND ISSUES)
  - DRAFT 9/81 (PNL)
  - FINAL 4/82 (PNL)
  - RESOLUTION OF PROBLEMS & GAPS 9/84

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**WASTE PROCESSING STUDIES ARE  
FOCUSED ON THOSE ACTIVITIES  
WHICH CHANGE THE PHYSICAL OR  
CHEMICAL FORM OF THE WASTE**

- MECHANICAL PROCESSING
- INCINERATION
- CHEMICAL PROCESSING
- METAL VOLUME REDUCTION
- WASTE FORM DEVELOPMENT
- GENERAL PROCESSING GUIDELINES

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## **WASTE PACKAGING INCLUDES:**

- GENERAL PACKAGING GUIDELINES
- TRANSPORT PACKAGE
- DISPOSAL PACKAGE

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## **WASTE HANDLING INCLUDES:**

- GENERAL HANDLING GUIDELINES
- GENERATOR SITE HANDLING
- TRANSPORT HANDLING
- DISPOSAL SITE HANDLING

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## **WASTE TESTING INCLUDES:**

- GENERAL TESTING GUIDELINES DEVELOPMENT
- TREATMENT METHODS DEFINITION
- STANDARDIZED TEST PROCEDURES SELECTION
- TESTING CENTERS ESTABLISHMENT (FOR QA)

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## **EARLY INITIATIVES PRODUCED PROGRAM OUTPUTS WHICH WERE VALUABLE IN ASSESSING NEEDS FOR FUTURE WORK**

- A STATE-OF-THE-ART REPORT ON LOW-LEVEL RADIOACTIVE WASTE TREATMENT, ORNL/TM-7427, A. H. KIBBEY AND H. W. GODBEE (9/80)
- STATE-OF-THE-ART REVIEW OF RADIOACTIVE WASTE VOLUME REDUCTION TECHNIQUES FOR COMMERCIAL NUCLEAR POWER PLANTS, ORNL/SUB-79/13837/2, GILBERT/COMMONWEALTH (9/80)
- SURVEY OF NEEDS FOR BIOLOGICAL TREATMENT OF RADIOACTIVE LOW-LEVEL LIQUIDS IN NUCLEAR FACILITIES, ORNL/NFW-79/76, W. W. PITT, JR., H. W. GODBEE, C. H. HANCHER, B. D. PATTON, AND S. E. SHUMATE II (5/80)

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## **CRITICAL TECHNOLOGY AREAS HAVE BEEN SUPPORTED AS A RESULT OF EARLY NEEDS ASSESSMENTS**

- WASTE GENERATION REDUCTION
- INCINERATION OF ORGANIC LIQUIDS
- INCINERATION OF COMBUSTIBLE SOLIDS
- CONCENTRATION AND DESTRUCTION OF LIQUID CONTAMINANTS
- METAL WASTE VOLUME REDUCTION
- WASTE SOLIDIFICATION (TESTING)
- WASTE HANDLING AND SEGREGATION
- TREATMENT OF SOLIDS FOR DISPOSAL

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## STATUS SUMMARY OF WASTE TREATMENT, PACKAGING, AND HANDLING TECHNOLOGY ACTIVITIES

| TECHNOLOGY AREA                  | ANL-W | BNL | EG&G | MOUND | ORNL | PNL | RFO | RHO | SRL |
|----------------------------------|-------|-----|------|-------|------|-----|-----|-----|-----|
| WASTE GENERATION REDUCTION       |       |     |      |       | ●    | ●   |     | ●   |     |
| INCINERATION LIQUIDS             |       |     | ●    |       |      |     |     | ●   | ●   |
| INCINERATION SOLIDS              |       |     | ●    | ○     |      |     |     |     | ●   |
| CONC. & TREATMENT : LIQUIDS      |       |     |      | ●     | ●    |     |     |     |     |
| METAL VOLUME REDUCTION           |       |     | ○    |       |      |     |     | ●   |     |
| SOLIDIFICATION (TESTING)         |       | ○   |      |       |      |     |     |     |     |
| HANDLING                         |       | ●   |      |       | ○    |     |     |     |     |
| TREATMENT OF SOLIDS FOR DISPOSAL | ○     |     |      |       |      |     | ●   |     |     |

● SOME RESULTS AVAILABLE

○ TASK UNDERWAY



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## **WASTE GENERATION REDUCTION**

- DOCUMENTATION OF ORNL PROCEDURES (ORNL)
- DOCUMENTATION OF RHO PROCEDURES (RHO)
- WASTE GENERATION REDUCTION MANUAL (PNL)

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## INCINERATION OF LIQUIDS

- MICROWAVE PLASMA TORCH (RHO)
- INSTITUTIONAL INCINERATOR (EG&G)
- SOLVENT SOLID WASTE INCINERATOR  
FACILITY TEST - SWIFT (SRL)

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## INCINERATION OF SOLIDS

- INSTITUTIONAL INCINERATOR (EG&G)
- GLASS FURNACE (MOUND)
- SOLVENT SOLID WASTE INCINERATOR  
FACILITY TEST - SWIFT (SRL)

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## **CONCENTRATION AND TREATMENT OF LIQUIDS**

- ULTRAFILTRATION/REVERSE OSMOSIS  
(MOUND)
- BIOLOGICAL DESTRUCTION OF NITRATE  
IN SOLUTION (ORNL)

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## METAL VOLUME REDUCTION

- SMELTER (EG&G)
- ARC SAW/VACUUM FURNACE EXPERIMENTS (RHO)

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## **SOLIDIFICATION (TESTING)**

- **WASTE FORM DEVELOPMENT AND TESTING (BNL)**

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## WASTE HANDLING

- WASTE SEGREGATION STUDY (BNL)
- WASTE TREATMENT HANDBOOK (ORNL)

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## TREATMENT OF SOLIDS FOR DISPOSAL

- SODIUM DISPOSAL (ANL-W)
- WASTE GENERATION REDUCTION -  
NITRATE (RFO)



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## **THE CURRENT STATUS OF LLWMP WASTE TREATMENT, PACKAGING, AND HANDLING TECHNOLOGY ACTIVITIES IS AS FOLLOWS:**

- MANY OF THE HARDWARE OPTIONS FOR WASTE TREATMENT HAVE BEEN REVIEWED FOR APPROPRIATE USAGE WITH LLW. SOME PROMISING OPTIONS REMAIN TO BE EVALUATED.
- TESTING OF TREATMENT TECHNOLOGIES WITH ACTUAL PROCESS RADIOACTIVE WASTES HAS BEEN INITIATED. CONSIDERABLE WORK REMAINS TO BE COMPLETED.
- ANALYSIS OF THE INTERACTION OF TREATMENT, SOLIDIFICATION AND DISPOSAL NEEDS TO BE COMPLETED.



## WASTE CLASSIFICATION SYSTEM FOR LOW-LEVEL RADIOACTIVE WASTES

L. E. Wickham  
Low-Level Waste Management Program  
EG&G Idaho

### ABSTRACT

Some low-level radioactive waste represents both a chemical and radiological hazard, and it is desirable that management practices take this into account. This paper describes a waste classification system that could provide a basis for such management practices.

The classification method formulates the innate hazard of waste as the ratio of potential exposure or intake to some selected limit. Potential exposure or intake relates two factors: 1) the nature and amount of toxic materials disposed in the soil, and 2) the available biological hazard posed when those materials leave the soil and enter a biological pathway potentially exposing humans. The system will provide guidance on how wastes could be separated for disposal, and the degree of disposal isolation each waste material or shipment needs. An example of the systems use is provided.

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### INTRODUCTION

Besides being radioactive, some low-level waste is chemically hazardous or toxic. For example, liquid scintillation waste ("scintillation cocktails") contains small amounts of radioactivity in toxic organic chemicals. These toxic organic chemicals, such as xylene, may be more hazardous than the small amounts of radioactivity contained within them. Likewise, some chemical forms of uranium are more hazardous or toxic chemically than radioactively.

Since some low-level waste represents both a chemical and radioactive hazard, it is desirable that management practices take this into account. This desire has spurred many attempts to develop a waste classification system that classifies waste by total hazard. Such a classification system could provide a basis for management practices that would reflect the radiological, biological, and chemical hazards of the waste.

In this paper a new method for classifying low-level waste is introduced. The method is based on the geological hazard index introduced by Smith et al. (1980), and provides a measure of the total innate hazard of the waste after disposal. The classification system is designed to be a simple, cost effective, and efficient waste screening tool that is conservative and applicable for the great majority of waste streams.

## METHODOLOGY

The basic methodology for the system has been developed. The heart of the system is the hazard index. The hazard index,  $HI_n$ , is composed of two terms that describe: 1) the nature and amount of toxic materials disposed in the soil; and, 2) the available biological hazard posed when those materials leave the soil and enter a biological pathway with a potential for exposing humans. This is formulated as

$$HI_n = Q_n \times AH_n, \quad (1)$$

where

$Q_n$  = the effective concentration of each toxic material  $n$ ,

$AH_n$  = the relative available hazard that could result from exposure (ingestion, inhalation, or direct exposure) to toxic material  $n$ .

### EFFECTIVE CONCENTRATION FACTOR

The effective concentration of a given toxic material,  $Q_n$ , is a function of the initial concentration of the hazardous material, time, and containment provided by engineered features, the waste form and waste container. This may be expressed as

$$Q_n = Q_{on} \times P_n \times WC_n \times WF_n, \quad (2)$$

where

$Q_{on}$  = the initial concentration of toxic material  $n$  in the waste (Ci or mg/m<sup>3</sup>)

$P_n$  = the persistence through time of material  $n$  (unitless),

$WC_n$  = the waste container factor (unitless), and

$WF_n$  = the waste form factor (unitless).

### Initial Concentration

$Q_{on}$  is the measured concentration of a radionuclide or chemical in the waste. In general, the concentration would be measured in individual waste containers, although application of the formula to entire waste shipments or disposal areas is also possible.

### Persistence

The term persistence describes the longevity of a material over a

specified period of time. In this study, the control period used is 500 years. This is the longest time period identified by the U.S. Nuclear Regulatory Commission that a low-level waste disposal facility must perform. One hundred years may be a more appropriate time period.

For radioactive substances, longevity is a function of each nuclide's decay rate. Short-lived nuclides such as  $^{60}\text{Co}$  or  $^3\text{H}$  do not persist over the 500 years and therefore have small persistence values. The longer lived  $^{239}\text{Pu}$  and  $^{238}\text{U}$  have values of 1, indicating that essentially all the material will be available for 500 years and beyond.

Chemicals present different problems. Some substances may undergo a degradation or volatilization process while others will persist indefinitely. Currently, it may not be possible to reliably predict the degradation processes, rates, or site specific characteristics contributing to degradation; therefore, using a persistence factor of 1 is the reliably conservative option for generic assessment at this time.

#### Waste Container Factor

The third and fourth terms of the effective concentration equation account for engineered barriers that may mitigate release of waste materials into surrounding soil. The waste container serves as the first barrier to waste migration. During the time the container is intact, there is no release of waste and the waste safely decays in place. As soon as the container is breached or opened to the surrounding soil, the waste may enter biosphere pathways to man.

The waste container factor is defined as the fraction of original waste material remaining when a container is breached; and, hence accounts for the effect of the lifetime of the container on waste material concentration. The values of WC will vary between 0 and 1. When the container life is long and the material life short, then the majority of the material will decay or degrade in the container without being exposed to the soil and the WC value approaches 0. Likewise, when the container life is short and the material life is very long, then the value of WC approaches 1.

#### Waste Form Factor

The waste form is another barrier to potential waste migration. When a material is mixed with a fixation agent like concrete, the rate that the material is released to the surrounding soils is greatly reduced. Waste forms such as loosely packaged paper trash do not inhibit release of waste materials; hence, they can readily enter the soil once the waste container is breached. The waste form factor,  $WF_n$ , is defined as the maximum fraction of material which could be released from a container in the first year after the container is breached. WF values are based upon laboratory tests and may not reflect field conditions for solidified waste in soil. Also, since WF values have been calculated for only a small number of materials, using a waste form factor of 1 may be the reliably conservative option at this time.

## AVAILABLE HAZARD FACTOR

The second major factor in the hazard index is the relative available hazard. The relative available hazard of a given material,  $AH_n$ , is a function of the availability of material  $n$  in the biosphere and the ratio of potential biological effects normalized to a selected limit. This may be described mathematically as

$$AH_n = A_n \times \frac{DCF_n}{DC_n}, \quad (3)$$

where

$A_n$  = the rate that toxic material  $n$  could leave the soil and enter a transport pathway to man ( $m^3/yr$ ),

$DCF_n$  = the dose conversion factor for  $n$ , which relates the dose received per unit intake or exposure (dose/unit intake), and

$DC_n$  = the selected criteria (dose/year).

### Availability

The availability factor describes the rate that a substance can leave the soil and enter a transport pathway to man. Pathways considered in the system are ingestion, inhalation, and external exposure.

The specific approach selected to describe availability in the ingestion pathway uses concentration factors, which are simple ratios that describe the equilibrium concentration of a material in one part of the biosphere such as soil, relative to another part, such as vegetation. Concentration factors have been empirically obtained from field and laboratory studies for a number of radionuclides and stable elements. Less generic information is available for nonradiological hazardous wastes.

A number of assumptions also were made regarding the activities and environmental conditions used in calculating the availability rate. It was assumed that the receptor is a "homesteader" who lives on a disposal area for 50 years. Radionuclide concentrations in the soil are equal to that in the buried waste. The homesteader obtains all his food from the farm, including meat, milk, and vegetables. He breathes resuspended contaminated soil. No radionuclide decay occurs during the 50 year exposure period, and no depletion of radionuclides in soil occurs due to soil resuspension and dispersion, harvesting of crops, or leaching into deeper soil layers.

For radiological substances, the equations used to describe terrestrial food chain transport are the same as those used by the

Nuclear Regulatory Commission (NRC) to evaluate doses from nuclear power plants (NRC 1977). The end result of the analysis is a yearly ingestion rate of each radionuclide which corresponds to a unit concentration of that radionuclide in soil. For nonradiological substances, the methodology is the same as that applied to radiological substances, assuming that any chemical substance has an infinite half-life and a decay constant of zero. Parameter values used in calculating ingestion rates are the same as those for radiological substances.

A mass loading approach was used to estimate the availability of radionuclides from inhalation of contaminated soil. The availability of contaminated dust is a measure of the availability of a material attached to that dust. Hence, a single availability value represents all non-volatile substances.

It can be argued that buried wastes, by the time they reach the soil surface, are essentially "part" of the soil, or at least sorbed strongly to surfaces of soil, and that the mass loading approach is a good choice for the waste classification index. However, inhalation of volatile substances will never be satisfactorily handled using the mass loading approach, since those compounds do not adsorb strongly to soils.

The external exposure pathway, which for the present is considered valid for radioactive substances only, also has been calculated. Exposure rates from gamma-emitting radionuclides have been measured and extrapolated from known quantities of radionuclides homogeneously distributed in soil (Beck and de Plangue, 1978).

#### Dose Conversion Factor

A dose conversion factor is included in the equation to convert the material concentration to a dose. For nonradioactive substances, DCF's will be equal to 1. For radioactive substances, the dose conversion factor derives the internal radiation dose that results per unit of radionuclide ingested or inhaled, or the external dose from exposure to gamma radiation 1 meter above the surface (Oztunali, et al. 1981).

The assimilation, distribution, and retention of elements in the body is considered, as well as irradiation of various organs by different isotopes of the same element. Organ doses are weighted according to their radiosensitivity (ICRP30), and then summed to give an effective whole-body dose. This effective whole-body dose is intended to represent the same risk as if the entire body were irradiated uniformly.

#### Dose Criteria

Dose criteria are used as a way of equating the effects of radiological and chemical waste substances on a similar scale. The dose criteria selected for this purpose are Toxic Dose Low ( $TD_{10}$ ) for ingestion and Toxic Concentration Low ( $TC_{10}$ ) for inhalation.  $TD_{10}$  and  $TC_{10}$  are the lowest reported cumulative ingestion and inhalation doses that have been observed to cause any toxic effect in humans or carcinogenic,

neoplastic, tumorigenic, or teratogenic effects in laboratory animals. Data are reported for both acute and chronic effects depending on the exposure period; chronic exposure studies (i.e., greater than one year in length) are preferentially selected when available because these studies are perceived as most representative of exposure conditions at a waste disposal site.

In order to develop a unified index to evaluate both radioactive and chemical materials, both types of materials must have  $TD_{10}$  and  $TC_{10}$  values. While these values are prevalent for chemicals, a separate evaluation was necessary to select a radiation exposure value comparable to chemical  $TD_{10}$  and  $TC_{10}$ . Evidence from studies on genetic abnormalities, teratogenic cancer induction and stochastic carcinogenesis suggest that one rad is a good estimate for a radiological  $TD_{10}$  and  $TC_{10}$ .

For the chemical compounds considered (General Research Corporation, 1980), 43 substances had either a  $TD_{10}$  or  $TC_{10}$ . However, few substances had both, which greatly limits a full pathway analysis of any particular chemical substance at the present time.

$TD_{10}$  and  $TC_{10}$  values are in theory ideally suited for use as a dose criteria. They are reported for a large number of substances (although a limited number of radioactive waste substances), are updated quarterly, and report chronic exposure, many of which are applicable to human exposures. Further evaluation, however, indicates there are three principal problems with this approach. First, not all chemical  $TD_{10}$  and  $TC_{10}$  values are equivalent in representing chronic toxicity. For some toxic effects, there is no indication of the severity or reversibility of the reported effect. Second, data are reported for a variety of animal species, as well as humans. At present, there is no universally accepted method for determining human toxic effects based on laboratory animal experiments. Third, reported data covers a variety of exposure times and may not include all exposure times or effects for each laboratory animal. For example, a  $TD_{10}$  may be reported for an acute exposure period (1 week or less), while a chronic study may go unreported.

#### EXAMPLE APPLICATION

Recapitulating, the hazard index is formulated as:

$$HI_n = (Q_{on} \times P_n \times WC_n \times WF_n) \times \frac{(A_n \times DCF_n)}{D_n} \quad (4)$$

Hazard indices are calculated for each material,  $n$ , in the waste and for all transport pathways. The individual indices are summed for a total hazard value and the final waste classification index (WCI) is formulated as:

$$WCI = \log_{10} HI \quad (5)$$

To illustrate the use of the Waste Classification Index a hypothetical waste mixture containing an arsenic pesticide (arsenic trioxide) and various radionuclides ( $^{14}\text{C}$ ,  $^{60}\text{Co}$ , and  $^{131}\text{I}$ ). The materials are assumed to be contained in a 55-gallon drum without any special waste form treatments. Hazard Index values are calculated in Table 1.

Table 1. Hazard Index Calculations For Hypothetical Waste Mixture

| Substance        | Initial Concentration  | HI <sup>a</sup><br>(ingestion) | HI<br>(inhalation) |
|------------------|------------------------|--------------------------------|--------------------|
| $^{14}\text{C}$  | 0.80 Ci/m <sup>3</sup> | 3.3E3 <sup>b</sup>             | 1.4E-3             |
| $^{60}\text{Co}$ | 0.12 Ci/m <sup>3</sup> | 4.6E-2                         | 1.1E-4             |
| $^{131}\text{I}$ | 0.08 Ci/m <sup>3</sup> | 0 <sup>c</sup>                 | 0 <sup>c</sup>     |
| AR $\text{O}_3$  | 3000 mg/m <sup>3</sup> | 2.4E-3                         | 4.5E-9             |
|                  | Total                  | 3.3E3                          | 1.5E-3             |

a HI = hazard index

b  $3.3\text{E}3 = 3.3 \times 10^3$ .

c Because the half-life of  $^{131}\text{I}$  is very short (8 days), the hazard when placed in the 55-gallon drum is effectively zero.

From Table 1 it can be seen that the hazard from ingestion of  $^{14}\text{C}$  dominates the hazards from all other substances in all other pathways. The hazard of  $^{131}\text{I}$  is also noteworthy. Because the half-life of  $^{131}\text{I}$  is very short (8 days), the hazard from this radionuclide is effectively zero.

The total hazard is the sum of each pathway hazard:

$$\text{HI}_{\text{total}} = 3.3 \times 10^3 + 1.5 \times 10^{-3} = 3.3 \times 10^3.$$

Hence the Waste Classification Index is

$$\text{WCI} = \log_{10} \text{HI} = \log_{10} (3.3 \times 10^3) = 3.5.$$

## DISCUSSION

The waste classification index compares the hazard of toxic material in a land burial situation to a selected standard. The WCI is a relative scale for comparing the innate hazards of toxic materials and/or types of waste. This relative scale can help determine the degree of disposal isolation needed to contain the waste so that the hazard to man is minimized. A container of waste calculated to have a WCI of 3 would be disposed of at a location that accepts drums having classification numbers of, for example, 10 and below.



The majority of the classification systems methodology has been developed and presented here. Three major tasks still remain to be done. First, values for the persistence of chemicals need to be developed. Second, availability factors must be derived for the ingestion of organic compounds and inhalation of volatile compounds. Third, a thorough review and evaluation of the toxic dose level methodology must be done.

The waste classification index has been developed to allow some quantification of total waste hazard. The methodology has been designed to be as general as possible, so that the index would be applicable over a broad range of waste streams. The approach is intuitively simple and conservative. The waste classification index can be very useful as a measurement tool to aid in waste management decisions.

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## LOW-LEVEL WASTE TREATMENT HANDBOOK

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## INTRODUCTION

At the present time, one of the major obstacles for nuclear power expansion is public concern regarding radioactive waste disposal. Wastes are generated in all phases of the commercial nuclear fuel cycle, ranging from large volumes of slightly radioactive uranium mill tailings to relatively small volumes of high-level radioactive wastes from the processing of nuclear fuels. Commercial nonfuel-cycle wastes result from operations by private institutions and industries that are licensed to use radioactive materials in research, medical, and various industrial activities. The governmental operations that generate radioactive wastes include defense activities as well as basic research and development studies at several Department of Energy (DOE) sites.

The largest volume of radioactive wastes generated is termed low-level waste (LLW) and is the subject of a handbook in preparation at Oak Ridge National Laboratory. Low-level waste is defined in a negative sense as radioactively contaminated material that is not classified as high-level waste, transuranic waste, spent nuclear fuel, or by-product material, as defined by the amended Atomic Energy Act of 1954 (1).

The radionuclides in LLW vary widely. Uranium (natural and depleted) and its daughters dominate in the conversion, enrichment, and fuel fabrication steps of the fuel cycle. Reactor operation and spent fuel storage generate LLW containing mostly activation products and fission products. A significant fraction of institutional LLW is contaminated with small quantities of tritium and carbon-14. Institutional and industrial LLW also contains cobalt-60, cesium-137, and other gamma-emitting nuclides. Low-level waste generated by the government is similar to that of institutions, industry, and the commercial fuel cycle. Of the 166,000 m<sup>3</sup> of LLW buried in the United States in 1980 (2), approximately 50 vol % was attributed to governmental activities, 30 vol % to the U.S. commercial fuel cycle, and 20 vol % to institutional and industrial organizations. The treatment and disposal of special LLW, such as active uranium mill tailings and LLW generated by remedial action programs, are beyond the scope of the handbook.

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\*Operated by Union Carbide Corporation under contract W-7405-eng-26 with the U.S. Department of Energy.

## OBJECTIVE

The objective of the Low-Level Waste Treatment Handbook is to present an overview of current practices related to the segregation, classification, volume reduction, solidification, handling, packaging, and transportation of LLW for disposal in a shallow land burial facility. The handbook is intended to serve as a guide to individuals who have no previous knowledge in handling LLW and who need an introduction to the available technology and regulations. The handbook will not explicitly tell the user how to design and operate LLW treatment facilities, but rather will identify (1) kinds of information required to evaluate the options, (2) methods that may be used to evaluate these options, and (3) limitations associated with the selection of the treatment options. The focus of the handbook is on "how to do" waste treatment for disposal by shallow land burial.

## BACKGROUND

Shallow land burial techniques similar to those employed by municipalities and industry for the disposal of solid wastes are now used to dispose of essentially all low-level radioactive waste. Commercial wastes are currently buried in privately operated burial grounds licensed by the U.S. Nuclear Regulatory Commission (NRC) or by a state, if the state has qualified as an Agreement State under the Atomic Energy Act. Governmental LLW is mainly generated and buried at DOE facilities. Existing DOE burial grounds are not licensed by NRC or the Agreement States; however, they are regulated by the DOE under the Atomic Energy Act. In 1980 about 74,000 m<sup>3</sup> of LLW was buried at DOE sites and about 92,000 m<sup>3</sup> at commercial sites. Projections of future LLW generation indicate that DOE wastes will remain relatively constant at 80,000 m<sup>3</sup>/year for the next 20 years, whereas commercial production is estimated to increase to about 225,000 m<sup>3</sup>/year by the year 2000 (2).

A major concern with the projected increase in commercial waste generation is the limited amount of disposal space available at existing sites and the fact that the geographic distribution of disposal sites is not well matched with current waste sources. Although the problem is not yet severe at DOE sites, existing burial space will also be used up in the near future at certain sites. As a result, new burial grounds will be required in the near future.

To date, six commercial sites have been licensed to dispose of LLW; however, three of these sites have been closed. Of the remaining three, Barnwell, South Carolina, has placed limits on the amount of waste accepted for burial each month, and the other two operating sites (at Beatty, Nevada, and at Richland, Washington) are remote from waste sources in the eastern United States. It is estimated that from five to seven regional disposal sites will be required to meet 1990 commercial LLW disposal requirements (3).

The loss of the most convenient LLW disposal sites for the majority of the commercial waste generation facilities resulted in immediate transportation cost increases. These cost increases were compounded by what has become a steep spiral in the burial charges for the three remaining sites, aggravated by the recent addition of local government taxes and restrictions, inspection requirements, and burial volume limitations. During the past five years, the radioactive waste burial charges per unit of volume have increased more than 500% (4).

With the diminishing availability of LLW disposal facilities and continuing escalation of burial cost, a quotation from the NRC's policy statement (5) is apropos: "The Commission hereby adopts a policy calling on all generators of low-level radioactive waste to reduce the volume of waste for disposal ... NRC believes it is in the best interest of licensees and the public that licensees extensively explore means by which volume may be reduced." One goal of volume minimization is to exercise better administrative controls to limit the production of waste, such as contaminated dry trash. Volume minimization can also be achieved by more efficient waste treatment options. However, maximum volume reduction will dictate equipment modifications involving capital investment and licensing considerations. The commercial introduction of more sophisticated systems, such as incineration for combustible wastes, can significantly reduce the LLW volumes for subsequent disposal.

The NRC recently proposed licensing requirements for the commercial disposal of LLW (6). The most direct impact of these requirements on volume reduction systems will be the limitations imposed on specific isotope activities per unit volume of LLW buried. Depending on input waste activities, some of these isotopic limits would typically be exceeded if maximum volume reduction were achieved. As a result, the NRC proposal could challenge the viability of high-technology volume reduction systems. In addition, the proposed classification system, which requires segregation of certain wastes, will impact the operating philosophy of volume reduction systems that significantly concentrate the radioactivity (e.g., incinerators that reduce combustibles to an ash form). Both of these potential concerns may be partially offset by blending input wastes and accounting for the dilution in a solidification binder and/or waste container. However, additional operational controls will probably be required to measure critical isotopic concentrations.

### QUALITY ASSURANCE

Quality assurance is defined as the planned and systematic actions necessary to ensure confidence that a structure, system, component, facility, or operation will perform satisfactorily. A quality assurance (QA) program should be included in all major steps of LLW handling from generation to successful disposal. This program should develop a QA plan, which is a document that describes potential significant quality failures, presents the QA actions required to provide confidence that the

postulated failures are unlikely to occur, and specifies the responsibility and timeliness for carrying out the QA actions. Quality assurance is a key feature in ensuring satisfactory treatment, handling, packaging, and shipment of generated LLW to the burial facility.

## ORGANIZATION OF THE HANDBOOK

The six chapters of the handbook describe LLW treatment from source to ultimate disposal. An abbreviated Contents is shown in Table 1. In general the handbook applies to both commercial and governmental sources of LLW, although some of the treatment practices may be unnecessary at DOE facilities, which are not licensed by the Agreement States nor the NRC. The critical step in handling LLW is to become thoroughly familiar with the rules concerning waste classification requirements (e.g., concentrations may be averaged over the package volume). With a characterized waste for disposal and a known volumetric burial charge, a cost-effective waste treatment procedure can be developed by the radioactive source generator. The treatment options available to achieve the permissible volume reductions will probably require equipment modifications involving capital investment and licensing considerations.

An introduction to the LLW treatment problem is given in Chapter 1. Chapter 2 deals with the sources and characteristics of LLW, disposal criteria at the shallow land burial facility, and treatment options for dry and wet wastes. Included are discussions of wastes generated in the commercial fuel cycle; institutional, industrial, and governmental wastes; and special category wastes such as uranium mill tailings and remedial action wastes.

Conceivable volume reduction techniques for both dry and wet wastes are described in Chapter 3. The unit operations involving physical and chemical treatments will have a significant impact on the volume of LLW shipped to the shallow land burial facility for disposal. Included are discussions of compaction, incineration, evaporation, filtration, reverse osmosis, ion exchange, coagulation, and sedimentation.

Chapter 4 presents details on LLW solidification techniques including a discussion of sorbents, cement, asphalt, and thermosetting organic polymers. This chapter has significance for the ultimate disposal form since the proposed NRC requirements specify that all LLW liquids must be packaged in sufficient absorbent material to absorb twice the liquid volume. In addition, several waste classifications require structural stability which can be attained by processing the waste to a stable form or by placing the waste in a structurally stable disposal container.

Chapter 5 treats the general subject of LLW handling and deals with the specific topics of waste segregation, monitoring, accountability, and conditioning prior to packaging.

**Table 1. Abbreviated Contents for Low-Level  
Waste Treatment Handbook**

| <b>Chapter</b> | <b>Title</b>                                                                                                                                                                                                   |
|----------------|----------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------|
| <b>1</b>       | <b>Introduction</b>                                                                                                                                                                                            |
| <b>2</b>       | <b>Waste Types, Characteristics, and Treatment Options</b> <ul style="list-style-type: none"><li>• Waste sources and characteristics</li><li>• Criteria for disposal</li><li>• Treatment options</li></ul>     |
| <b>3</b>       | <b>Volume Reduction</b> <ul style="list-style-type: none"><li>• Dry wastes</li><li>• Wet wastes</li></ul>                                                                                                      |
| <b>4</b>       | <b>Solidification Techniques</b> <ul style="list-style-type: none"><li>• Sorbents</li><li>• Cements</li><li>• Asphalt</li><li>• Thermosetting organic polymers</li></ul>                                       |
| <b>5</b>       | <b>Waste Handling</b> <ul style="list-style-type: none"><li>• Segregation</li><li>• Monitoring</li><li>• Records</li></ul>                                                                                     |
| <b>6</b>       | <b>Packaging and Transportation</b> <ul style="list-style-type: none"><li>• Regulations</li><li>• Package type and size</li><li>• Labeling, marking, and documentation</li><li>• Reporting incidents</li></ul> |

The last chapter covers the packaging and transport of LLW to the disposal facility. The shipping requirements are regulated by the NRC, the Department of Transportation (DOT), the Postal Service, and the individual states. These regulations are summarized, and the responsibilities of shippers and carriers are outlined. Procedures for reporting incidents are also discussed.

### STATUS

The initial draft of the Low-Level Waste Treatment Handbook has been prepared and submitted to the DOE Low-Level Waste Management Program for review and comment. A revised draft is scheduled to be delivered to DOE Headquarters in December 1982. The handbook is designed to be useful to all individuals and groups concerned with LLW. It is one of several volumes that will ultimately comprise a Low-Level Waste Technology Handbook.

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CHARACTERISTICS OF FUEL CYCLE WASTE *Deep*

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## BACKGROUND

Since its establishment in March of 1979, the National Low-Level Radioactive Waste Management Program has been working to establish an acceptable commercial low-level waste management system. From the beginning of the program it was recognized that state governments, private industry, and several Federal agencies would each play a major role in commercial low-level waste management. To fulfill this role effectively, each entity would require accurate information on the activities and practices used to generate, treat, or dispose of low-level waste. A lack of policy direction confounded progress toward this goal; it was unclear which organization had primary responsibility for commercial low-level waste management. A further detriment to progress was the overabundance of information from a multitude of sources, none of which seemed to agree.

The passage of the Low-Level Radioactive Waste Policy Act (Public Law 96-573) of December 1980, clarified Federal policy for the disposal of low-level radioactive waste. The act affirmed state responsibility for the disposal of low-level radioactive waste. Each state must provide disposal capacity for the low-level waste generated within its borders. Regional solutions are recommended but individual state disposal sites are not precluded. State responsibility does not apply to low-level waste generated by Federal research or defense activities. In order to fulfill the responsibilities of the Policy Act, states are entering into regional compacts to establish and manage regional low-level waste disposal facilities.

By adding a third level, regions, to the existing two, states and national, the need for a reliable and accurate common information management system increased. As states began to negotiate regional compacts, the first major question to be addressed was how much low-level radioactive waste is generated in each state. Subsequent questions are never ending and entail concerns over how much waste will be generated in the future, what waste management practices could or should be changed, and what effect would such changes have on the waste stream. The Low-Level Waste Management System anticipated such questions and started in 1979 to describe and model the existing commercial low-level waste management system. The system description produced is based on the identification of the different elements making up both the fuel and non-fuel cycle and their relationships to each other. Figure 1 shows the elements and relationships for the fuel



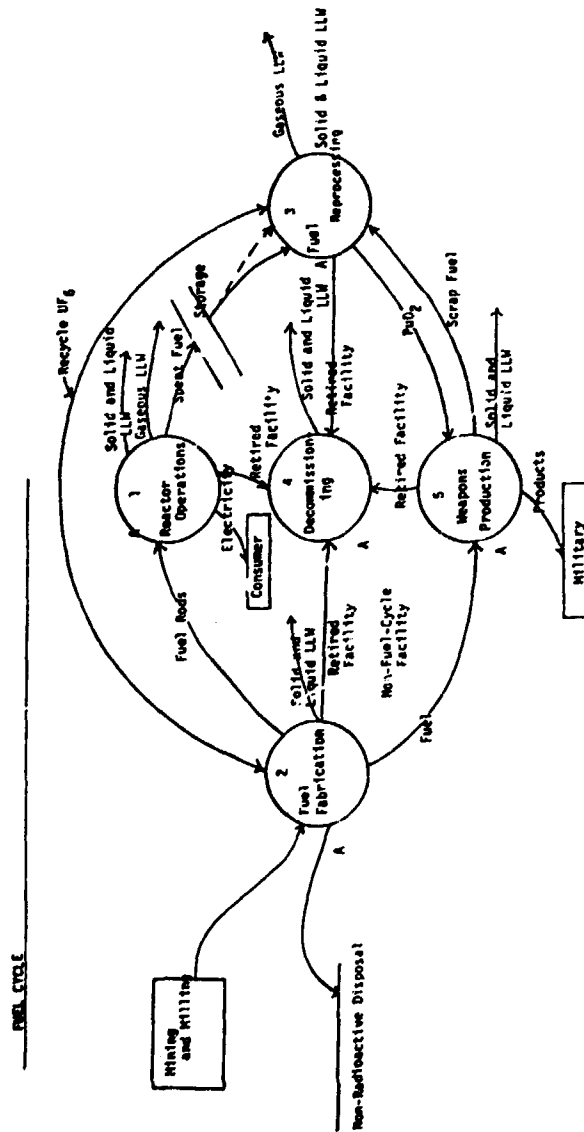


Figure 1.

cycle. A systems model based on the system description can accurately reflect the flow of low-level waste from generator to disposal site and is only limited by the reliability of the information it uses. For both the fuel cycle and non-fuel cycle large quantities of information is required in order to allow the system to operate at its full potential. Work is ongoing to collect this information. Significant progress is being made in the fuel cycle area primarily because the majority of fuel cycle low-level radioactive waste is produced by commercial power reactor plant operations.

### DATA BASE

The low-level waste management system description identifies several points where accurate information is required. A data base tied to these points is maintained by the Low-Level Waste Management Program. For the fuel cycle the data base contains historical data on low-level waste generated by individual reactor plants (Table 1). Annual records of waste generated by each plant from 1972 to present are available. The data in its present form and quality has had limited use, however, the potential use for this information is large. Previous studies on a limited number of operating reactors indicate that on an aggregate basis, boiling water reactors produce approximately 1.21 cubic meters of low-level waste per MWE per year and pressurized water reactors produce approximately 0.61 cubic meters of low-level waste per MWE per year. Analysis are just starting which compare individual reactor plants to these aggregate numbers. As individual records are verified new and more extensive reactor plant sample sets will be compared with these same numbers. The outcome of such studies may go far beyond an estimate of how much waste is generated per megawatt. Other information to be gained is:

- . time relationship of power history versus waste volume generated
- . factors unique to type plant versus management practices
- . trends based on activity of the waste

More accurate projections of reactor plant waste generation are only a small benefit from such studies. Identification of areas amenable to changes which will result in less waste or more easily treated waste hold a much larger potential benefit.

These efforts and analysis deal with the questions of today. Questions of tomorrow are also being addressed. How much and what type of waste will be produced by the decontamination and decommissioning of fuel cycle facilities? How will this vary by type of facility? By decommissioning option? chosen? Once again information exists but it varies in the answers given and also by availability. The Low-Level Waste Management Program is addressing these questions. Information is being collected for input to the data base. All sources are being examined with the purpose to establish a set of generic assumptions. A predictive model based on these assumptions but amenable to change by the user is being established.

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Table 1. Example , Low-Level Waste Generated by Individual Power Plants

LOW-LEVEL RADIOACTIVE WASTE DATA  
WASTE REPORTED AS SHIPPED  
TO COMMERCIAL BURIAL GROUNDS  
FROM COMMERCIAL POWER REACTORS

RUN DATE: 06/23/82  
SCHEMA XREF: 1.1.1

| REACTOR NAME                       | UNITS | REPORT DATE | WASTE CAT | KG OF WASTE | CURIES SHIPPED |
|------------------------------------|-------|-------------|-----------|-------------|----------------|
| ARKANSAS NUCLEAR ONE               | 112   | 06/30/80    | A         | 46.07       | 3.332E+02      |
|                                    |       |             | B         | 72.00       | 5.200E-01      |
|                                    |       |             |           | 118.07      | 3.237E+02      |
|                                    |       | 12/31/80    | A         | 46.07       | 3.332E+02      |
|                                    |       |             | B         | 72.00       | 5.200E-01      |
|                                    |       |             |           | 118.07      | 3.237E+02      |
| DEAVER VALLEY POWER STATION        | 1     | 06/30/80    | A         | 43.00       | 4.420E+02      |
|                                    |       |             | B         | 113.00      | 4.400E-01      |
|                                    |       |             |           | 156.00      | 4.424E+02      |
|                                    |       | 12/31/80    | A         | 21.10       | 9.040E+01      |
|                                    |       |             | B         | 81.40       | 4.400E-01      |
|                                    |       |             |           | 102.70      | 9.104E+01      |
| BIG ROCK POINT NUCLEAR PLANT       | 1     | 06/30/80    | A         | 4.81        | 9.740E+00      |
|                                    |       |             | B         | 7.22        | 2.030E+01      |
|                                    |       |             |           | 30.02       | 8.700E-01      |
|                                    |       | 12/31/80    | A         | 37.24       | 2.117E+01      |
|                                    |       |             | B         | 42.05       | 3.093E+01      |
|                                    |       |             |           |             |                |
| BROWNS FERRY NUCLEAR POWER STATION | 1,213 | 06/30/80    | A         | 372.80      | 3.429E+03      |
|                                    |       |             | B         | 500.00      | 2.800E+02      |
|                                    |       |             | C         | 225.40      | 8.200E-01      |
|                                    |       | 12/31/80    | A         | 375.64      | 5.455E+03      |
|                                    |       |             | B         | 495.00      | 1.679E+02      |
|                                    |       |             | C         | 390.00      | 2.850E+00      |
|                                    |       |             |           | 1,263.44    | 5.624E+03      |
|                                    |       |             |           | 2,791.94    | 9.334E+03      |
|                                    |       |             |           |             |                |

The range of options available to the user will encompass the three standard scenarios: safestore, entombment, and dismantle. The user will be able to construct models based on individual needs and inputs. The output of such models will be able to be used to study the impact decommissioning has on the low-level waste management system. Will there be enough disposal capacity? Will there be enough trucks or casks? What types of waste, A, B, or C, will be produced? How much of these wastes will be produced? These and other questions will be answered - not only for decommissioning but for other activities and for other wastes.

#### CONCLUSION

The Low-Level Waste Management system is only beginning to derive the benefits to be obtained from an accurate low-level waste management information system. As data is verified and analyzed, results on a national as well as individual organization level will be gained. Comparisons to previous studies will be made. Accurate projections of waste volumes and activities to be produced, projected impacts of waste streams of treatment or management changes are only examples of information to be produced.

#### REFERENCE

1. A Waste Inventory Report for Reactor and Fuel Fabrication Facility Wastes, ONWI-20, March 1979.

## STUDY OF NON-FUEL CYCLE WASTES\*

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## ABSTRACT

The low-level radioactive waste generated by many non-fuel cycle industries and institutions is not as well characterized as that produced by nuclear power plants. To better understand the variety of non-fuel cycle waste products now being disposed of by commercial shallow land burial (SLB) and to assess specific packages in advance of the enactment of the proposed regulation, 10 CFR Part 61 (dated June 29, 1981), "Licensing Requirements for Land Disposal of Radioactive Waste," the United States Nuclear Regulatory Commission (NRC) requested Brookhaven National Laboratory (BNL), under FIN A-3165, (in April of 1981), to provide technical assistance in expanding the data base on the physical and chemical characteristics of these wastes. With the cooperation of two major corporations, this program enabled the NRC to examine the achievability of the proposed 10 CFR Part 61 criteria, prior to the enactment of the regulation.

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INTRODUCTION

In April of 1981, Brookhaven National Laboratory (BNL) initiated a program of technical assistance to the U.S. Nuclear Regulatory Commission (NRC) in the area of non-fuel cycle wastes. Non-fuel cycle wastes comprise nearly half of the activity of the low-level radioactive waste buried at commercial shallow land burial sites. Therefore, these wastes constitute a significant part of the source-term evaluation. The non-fuel cycle waste contribution to the source term needs to be better defined so that the NRC may be able to: (1) predict the source term, (2) assess site performance, and (3) provide criteria to regulate the type of waste form generated from non-fuel cycle operations. Generators

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\*Work performed under the auspices of the United States Nuclear Regulatory Commission.

of non-fuel cycle wastes are institutions, hospitals, medical laboratories, radiochemical manufacturers, and research laboratories. Technical assistance was provided in expanding the data base on the chemical and physical characteristics of the low-level radioactive waste of two major corporations. Evaluations were done on the basis of criteria for Class B Stable Waste packages as specified in proposed 10 CFR Part 61 (dated June 29, 1981).

## BACKGROUND

Each regional U. S. NRC Office of Inspection and Enforcement (I and E) requested licensees to respond in 1979 to questions listed in I and E Bulletin No. 79-19, entitled, "Packaging of Low-Level Radioactive Waste for Transport and Burial."

From the response of the Union Carbide Corporation (UCC), it was determined that during 1978 and the first half of 1979, UCC shipped approximately 8,700 cu ft of waste, with a total activity content of approximately 140,000 Ci. This amounted to a contribution by UCC of 15% of the total activity and less than 1% of the total volume of radioactive waste sent for disposal at commercial shallow land burial in that period. UCC produces Mo-99 by the neutron irradiation of targets containing enriched uranium. Approximately 75% of the curie content of the waste was due to short-lived solidified mixed fission products of U-235 (having less than 1% burn-up), which had decayed for 30 days.

From the response of the New England Nuclear Corporation (NEN), it was determined that during 1978 and the first half of 1979, NEN shipped approximately 37,000 cu ft of waste, with a total activity content of approximately 210,000 Ci. This amounted to a contribution by NEN of approximately 24% of the total activity and less than 5% of the total volume of radioactive waste sent for disposal at commercial shallow land burial during that period. The predominant isotope in the waste is tritium, which is used in the production of tritium-labeled compounds.

## PROGRAM DESCRIPTION

The following areas were examined for the generators identified by the NRC (i.e. UCC and NEN):

- A. (1) the identification of concerns and gaps in knowledge with respect to the criteria set forth in proposed 10 CFR Part 61 regarding waste package performance and possible degradative mechanisms (e.g., corrosion, biodegradation, and radiolysis) that are operative during storage and after burial, and (2) the recommendation of actions necessary to resolve these concerns or develop the information necessary to do so.

- B. the identification of the type of information that should be submitted by the waste generator on the shipping form to permit adequate assessment of the relative hazards of both the radioactive and chemical aspects of the waste.
- C. the identification of aspects of the waste package which could adversely affect burial site performance, such as, reduced soil retention of radionuclides.
- D. (1) the review of the state-of-the art post-emplacement monitoring at burial sites; (2) recommendations for improvements in monitoring techniques (where necessary) to account for any chemical and/or radiological hazard that was identified.

#### PROPOSED 10 CFR PART 61.55 - WASTE CLASSIFICATION

Under proposed 10 CFR Part 61.55, radioactive wastes suitable for SLB are defined to fall within one of three classes, A, B, or C.

The characteristics of Class A segregated waste are that: (1) the radioisotopic concentration does not exceed the limit for Class A segregated waste; (2) it is segregated from other waste at the disposal site; and (3) it is disposed of with only minimum requirements on waste form and characteristics as set forth in proposed 10 CFR Part 61.56a.

The characteristics of Class B stable waste are that: (1) the radioisotopic concentration does not exceed the limit for Class B stable waste; (2) it must meet more rigorous requirements on waste form to assure stability after disposal; and (3) physical form and characteristics of the waste must meet the minimum and stability requirements as set forth in proposed 10 CFR Part 61.56a and proposed 10 CFR Part 61.56b.

The characteristics of Class C intruder waste are that: (1) the radioisotopic concentration does not exceed the limit for Class C intruder waste; (2) it must meet more rigorous requirements on waste form to assure stability and also requires special measures at the disposal facility to protect against inadvertent intrusion; and (3) physical form and characteristics of the waste must meet the minimum and stability requirements as set forth in proposed 10 CFR Part 61.56a and proposed 10 CFR Part 61.56b.

#### PROPOSED 10 CFR PART 61.56 - REQUIREMENTS FOR CLASS B STABLE WASTE

The minimum requirements for all classes of waste are detailed by proposed 10 CFR Part 61.56a. The evaluations done by BNL were on the

Class B stable waste packages of UCC and NEN. The stability requirements for Class B stable waste, as specified in proposed 10 CFR Part 61.56b, are:

- (1) Waste must have structural stability for at least 150 years. A structurally stable waste form will maintain its physical dimensions to within 5% under the expected disposal conditions of a compressive load of 50 psi and factors such as the presence of moisture, and microbial activity, and internal factors such as radiation effects and chemical changes.
- (2) Liquid wastes, or wastes containing liquid, must be converted into a form that contains as little free noncorrosive liquid as is reasonably achievable, but in no case shall the liquid exceed 1% of the volume of the waste.
- (3) Void spaces within the waste and between the waste and its package must be reduced to the extent practicable.

#### OBSERVATIONS AND CONCLUSIONS

During the characterization and evaluation of specific Class B waste packages, the following observations and conclusions were made:

#### CHARACTERIZATION

- (1) Non-fuel cycle waste packages are heterogeneous.
- (2) Packages can contain mixtures of:
  - (a) lab trash (e.g. cellulose, plastics, and glass);
  - (b) non-compressible metallic items (e.g. vacuum pumps);
  - (c) concrete;
  - (d) adsorbent (e.g. sand and clay);
  - (e) compacted metallics.
- (3) Waste can be in the form of contaminated solids, adsorbed or solidified liquid, or gas.
- (4) Chemical waste in the package can include organics, solvents, decontaminating agents, and sludges.
- (5) Radionuclides in the waste can be alpha, beta, and gamma emitters, with a wide range of half-lives.
- (6) Low-level radioactive waste does not necessarily mean low-activity.



## STABILITY AND CONTAINMENT

- (1) Some data is available on the corrosion behavior of buried metals. The review of pertinent corrosion data indicated that:
  - (a) A standard 55-gal carbon steel drum will not remain structurally stable for 150 yrs.
  - (b) 304 SS probably will not contain over a 150-yr period because of pitting.
  - (c) The extent of corrosion of aluminum and its alloys can vary from none to severe.
  - (d) Biogenic corrosion is a significant contributor to corrosion of buried metals. (However, the published data of corrosion in soils includes this factor, because the soils were not sterilized.)
  - (e) The corrosion of concrete by an aggressive agent (e.g. sulfate ion) depends on the presence of moisture.
  - (f) The corrosion of metal encased in concrete is an electrochemical process.
  - (g) Corrosive agents can be generated by biodegradative and radiolytic mechanisms.
  - (h) Short-term tests are not available to predict 150-yr behavior of metals or concrete buried in soil.
- (2) Radiolysis of water (in cement), organics, plastics, and cellulose can produce large quantities of gas. Biodegradation also has the potential to generate large quantities of gas. Some of the packages that were characterized were thought to be self-sterilizing.

Gas generation in a sealed container may cause overpressurization of the container. Gas pressure buildup will not occur in a standard 55-gal drum because the drum will leak.

## SHIPPING OF WASTE

- (1) Sample filled-in shipping forms were obtained. Information requested on the form was evaluated as to its usefulness for future documentation of the contents, both in terms of radionuclides and chemicals.
- (2) DOT regulations for package Quantity type and radionuclide Transport Groups were reviewed.

- (3) NRC regulations (e.g. Certificates of Compliance) were reviewed.
- (4) EPA regulations on toxic and hazardous waste were reviewed. Hazards that were considered included pyrophoricity, flammability, explosivity, reactivity, carcinogenicity, toxicity, and gas generation.
- (5) Recommendations for additional information to be included on a shipping form included: performance of quality assurance procedures; the range of the measured radionuclide values; and identification of organics by Chemical Abstracts Registry Number.

#### EFFECTS OF WASTE ON BURIAL SITE

- (1) Information was obtained on the commercial shallow-land burial sites as regards:
  - (a) geology;
  - (b) hydrology;
  - (c) climate;
  - (d) seismic considerations; and
  - (e) waste trench designs.
- (2) Estimates were made for the following:
  - (a) time to breach of containment and availability of radionuclides to the burial environment; and
  - (b) loss of gaseous contents by permeation through potential barriers inside packages.
- (3) Assuming that particular radionuclides are available for contact with the burial site environment, the following have to be considered:
  - (a) individual rates of radionuclide migration;
  - (b) waste-soil interactions;
  - (c) complexation characteristics elements; and
  - (d) current burial site monitoring practices.

#### WORK PLANS FY 1983

New generators are in the process of being selected. Their Class B stable waste packages will be characterized. These packages will be evaluated in terms of the finalized regulation, 10 CFR Part 61.

## REFERENCES

1. U.S. Nuclear Regulatory Commission, "Licensing Requirement for Land Disposal of Radioactive Waste," Federal Register, Vol. 46, No. 142, July 24, 1981, 38081-38105.
2. E. P. Gause, L. G. Stang, D. Dougherty, E. Veakis, and J. Smalley, Brookhaven National Laboratory, "Characterization of the Radioactive Large Quantity Waste Package of the Union Carbide Corporation," BNL-NUREG-30247R, July 1982.
3. E. P. Gause, E. Veakis, and J. Smalley, Brookhaven National Laboratory, "Characterization of the Class B Stable Radioactive Waste Packages of the New England Nuclear Corporation," BNL-NUREG-30623R, August 1982.

PROPERTIES OF RADIOACTIVE WASTES  
AND WASTE CONTAINERS\*

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*Day*

ABSTRACT

This program is sponsored by the Nuclear Regulatory Commission to address basic concerns in assessing the performance of solidified radwaste. Experiments were initiated to address these concerns. In particular, leachability of solidified radwastes and the physical stability of the ensuing waste forms were evaluated. In addition, leaching experiments designed to address the effects of alternating wet/dry cycles and of varying the length of these cycles on the leach behavior of waste forms were initiated.

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OBJECTIVES OF PROGRAM

The objectives of this program, sponsored by the Nuclear Regulatory Commission (NRC), are to provide a data base which will assist the NRC in evaluating the acceptability of solidified low level waste packages for disposal with regards to minimizing radionuclide migration after burial. The program will establish tests for the physical and chemical properties to evaluate the performance of solidified wastes in long-term storage or disposal and will establish procedures for testing containers with respect to conforming with regulatory guidelines. Test procedures and methodology are being evaluated to study: (1) long-term performance of waste forms based on short-term laboratory test results; (2) behavior of full size forms from tests on several size forms, (3) in situ leach behavior from laboratory tests; and (4) properties of actual reactor waste. The information generated in this program should be a useful source term for modeling of radionuclide migration in radwaste burial sites.

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\*Work performed under the auspices of the United States Nuclear Regulatory Commission.

## PROPERTIES OF SOLIDIFIED WASTE

Basic concerns in assessing the performance of solidified wastes (waste forms) are dimensional stability and radionuclide release in shallow land burial (SLB) or disposal environments.

The following parameters were addressed:

- Waste types:
1. Simulated waste - evaporator concentrate (boric acid wastes) and organic ion exchange resins (cation and anion resins).
  2. Actual waste - Acquired from a LWR (Prairie Island, in conjunction with NRC project FIN A-6359).

Solidification agents: Portland (I, II and III) and high alumina (Lumnite) cements and bitumen.

Radiotracers:  $^{137}\text{Cs}$ ,  $^{85}\text{Sr}$  and  $^{60}\text{Co}$ .

Tests used: Modified IAEA leach test<sup>(1)</sup> and ASTM-C 39-72 standard compressive strength test.

### BORIC ACID WASTE IN PORTLAND III CEMENT

A study designed to evaluate the leachability of  $^{137}\text{Cs}$ ,  $^{85}\text{Sr}$ , and  $^{60}\text{Co}$  from simulated boric acid waste in Portland III cement and to measure the compressive strength of the ensuing waste forms was performed.<sup>(2)</sup> The simulated waste forms were leached using a modified IAEA leaching procedure for periods extending to 229 days. The compressive strength of specimens was measured before and after immersion in the leachant for 352 days. The waste forms were made with 3%, 6%, and 12% boric acid solutions; the waste-to-cement ratios studied were 0.5 and 0.7.

Increasing the waste-to-cement ratio from 0.5 to 0.7 caused an increase in the leachability of  $^{137}\text{Cs}$  from the three boric acid/ cement composite formulations. This effect is not noticeable for the leachability of  $^{85}\text{Sr}$ . The extent of  $^{85}\text{Sr}$  release in a given time was approximately one-twentieth that of  $^{137}\text{Cs}$  from these composites. Cobalt-60, on the other hand, was below the detection limit in the leachate for all the composites ( $3.0 \times 10^{-2}$   $\mu\text{Ci}$  per 1.5 L samples).

For a waste/cement ratio of 0.5, 352 days of immersion caused a substantial decrease (approx. 50%) in the compressive strength of specimens. For a waste/cement ratio of 0.7, although initially the compressive strength of these specimens was approximately 40 to 50% lower than those for w/c ratio of 0.5, it did not decrease further after 352 days of leaching. The compressive strength of specimens for waste/cement

ratios of both 0.5 and 0.7 was approximately 20 to 38 times higher than the minimum acceptable limit (50psi) set forth in the proposed Code of Federal Regulations, 10 CFR Part 61.56.

#### ORGANIC ION EXCHANGE RESINS IN PORTLAND AND LUMNITE CEMENTS

An experiment was performed to determine the leachability of  $^{137}\text{Cs}$ ,  $^{85}\text{Sr}$ , and  $^{60}\text{Co}$  from organic ion exchange resin/cement composites.<sup>(2)</sup> Portland II and Lumnite (high alumina) cements were used as binders, with waste-to-cement ratios of 1.0 and 1.8 for both cements. In addition, the displacement of the three radiotracers from the ion exchange resins upon mixing the loaded resins with the two cements (before solidification) was measured.

Lumnite cement showed a lower leachability of  $^{137}\text{Cs}$  and  $^{85}\text{Sr}$  than Portland II cement. Cobalt-60 was below the detection limit in leachates for both types of cement. Strontium-85 was observed only in leachates for Portland II cement. These observations may be due to either an isotopic dilution of radiotracers with the cement components or to an actual chemical interaction of the radiotracers with the cement matrices.

The amounts of  $^{137}\text{Cs}$ ,  $^{85}\text{Sr}$ , and  $^{60}\text{Co}$  displaced from the ion exchange resins during the paste phase of the solidification process were comparable when mixed with Portland II cement (waste-to-cement ratio of 1.0 and 1.8), but not with Lumnite cement.

#### ORGANIC ION EXCHANGE RESINS/BITUMEN COMPOSITES

$\text{H}^+$ ,  $\text{Na}^+$ ,  $\text{Cs}^+$ , and  $\text{Sr}^{+2}$  form cationic resins, loaded with  $^{137}\text{Cs}$  and  $^{85}\text{Sr}$ , were mixed in varying ratios with anionic resins in  $\text{SO}_4^{-2}$  form, and subsequently solidified in bitumen. The leachability and physical integrity of the resulting composites were evaluated.<sup>(2)</sup>

The presence of anionic resins in the sulfate form (50% by weight of resins) increased the leachability of  $^{137}\text{Cs}$  from the waste forms by approximately two orders of magnitude and by approximately an order of magnitude for  $^{85}\text{Sr}$ . The physical integrity of the forms deteriorated during leaching, as the amount of  $\text{SO}_4^{-2}$  resin increased.

#### REACTOR WASTE

A sample of actual reactor waste from the Prairie Island reactor was acquired in conjunction with NRC Project A-6359. The waste was solidified according to optimized process parameters. After curing, the waste forms will be leached using the modified IAEA leaching procedure. Data from this study will be used in conjunction with data generated

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## OVERVIEW OF DOE LOW-LEVEL WASTE MANAGEMENT PROGRAM DISPOSAL TECHNOLOGY ACTIVITIES

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### INTRODUCTION

The DOE Low-Level Waste Management Program (LLWMP) utilizes a systems approach in performing technology development activities (Figure 1). The technology development portion of the program includes several waste technology and disposal technology elements. Waste generation, collection and segregation, treatment, packaging, and transportation activities are addressed elsewhere. The disposal technology elements are discussed here. Site selection, although a related process and shown here, is not treated as a separate disposal technology development activity by the LLWMP. The area of site selection technology covers the activities required to identify, characterize, and qualify sites for shallow land burial or greater confinement disposal facilities, and includes a site selection methodology and site performance prediction techniques. EG&G Idaho has prepared draft criteria for site selection, which are currently being applied to a recently selected low-level waste disposal site within Oak Ridge National Laboratory (ORNL) and also to statewide disposal site screening in Tennessee.

Low-level waste (LLW) disposal technology activities include the areas of improved shallow land burial (SLB) disposal, corrective measures (formerly called remedial action) technology, and greater confinement disposal (GCD) technology (Figure 2). These activities are focused on the development, demonstration, and documentation of the technology needed to open, operate, rehabilitate (if necessary), and close a safe, reliable facility for the shallow land burial or greater confinement disposal of low-level radioactive waste.

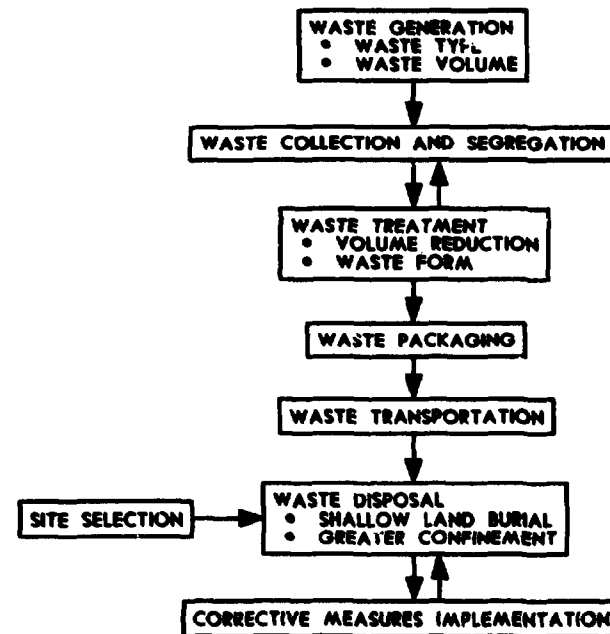
The principal goal of the LLWMP disposal technology development efforts is to develop and provide improved methods for the shallow land burial of LLW. Improved shallow land burial disposal technology includes the development, testing, and demonstration of improved disposal techniques for application to new shallow land burial sites. The improved SLB disposal technologies will be relevant to a wide range of disposal facility needs, which are: siting, design, construction, operation, and closure.



4th ANNUAL PARTICIPANTS' MEETING  
DOE LOW-LEVEL WASTE MGMT. PROGRAM  
DENVER, COLORADO - 8/31-9/2/82

WASTE DISPOSAL  
OVERVIEW

**THE WASTE MANAGEMENT SYSTEM MUST  
PROVIDE COST-EFFECTIVE PROTECTION  
OF HUMAN HEALTH AND SAFETY**



MSM 8/31/82

Figure 1.

4th ANNUAL PARTICIPANTS' MEETING  
DOE LOW-LEVEL WASTE MGMT. PRG. 2AM  
DENVER, COLORADO - 8/31-9/2/82

WASTE DISPOSAL  
OVERVIEW

**LOW-LEVEL WASTE DISPOSAL  
TECHNOLOGY ACTIVITIES INCLUDE  
THE RELATED AREAS OF:**

- IMPROVED SHALLOW LAND BURIAL  
DISPOSAL TECHNOLOGY
- CORRECTIVE MEASURES TECHNOLOGY
- GREATER CONFINEMENT DISPOSAL  
TECHNOLOGY

MSM 8/31/82

Figure 2.

Corrective measures (remedial action) technology is closely related to improved SLB technology, in that many areas of technical need identified for improved shallow land burial (such as those concerned with water management) are also applicable to corrective measures activities. Corrective measures technology development affects the rehabilitation of SLB facilities, and can be conducted during facility operation, closure, and post-closure. This element of the program is designed to ensure that radionuclide releases will not exceed performance standards at any particular site. Unplanned excessive releases will be contained, and/or leachate generation controlled or halted by employing the appropriate corrective measure(s).

Another primary goal of the LLWMP technology development work is to dispose of LLW "in such a manner as to provide greater containment of radiation, reduced potential for migration/dispersion of radionuclides, and greater protection from inadvertent human and biological intrusions in order to protect the health and safety of the public." Radionuclides will be afforded greater confinement than that achievable by shallow land burial. The greater confinement effort is predicated by the need to isolate certain higher activity or longer-lived radionuclides found in some LLW from the biosphere, and thereby reduce the health and safety risks to acceptable limits.

This overview covers the evolution of these three elements of the program, and the major technical issues that have been identified. Current work will be discussed and summarized with respect to the various contractor sites.

## DISPOSAL TECHNOLOGY OBJECTIVES AND MILESTONES

Several major programmatic objectives and milestones are associated with the three disposal technology elements. These are discussed below.

### IMPROVED SHALLOW LAND BURIAL TECHNOLOGY

The objective for improved shallow land burial is to develop and demonstrate the necessary technology (Figure 3). This objective will be accomplished by undertaking the required research, development and demonstration activities which will be documented in the Handbook for SLB disposal to be issued in March, 1984. This final SLB Handbook will be a technology transfer document for distribution to potential SLB facility developers and operators; it is the principal source of technical information on how to effectively conduct the shallow land burial of LLW. This manual will incorporate five major areas: site selection and characterization, design, operations, closure and post-closure. These sections of the Handbook will provide recommended

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WASTE DISPOSAL  
OVERVIEW

## **IMPROVED SHALLOW LAND BURIAL TECHNOLOGY**

- DEVELOP AND DEMONSTRATE THE TECHNOLOGY FOR SLB DISPOSAL
- ISSUE HANDBOOK FOR SLB DISPOSAL (3/84)

MSM 8/31/82

Figure 3.

practices, data requirements, data acquisition methodology, performance objectives, and performance assessment. The draft version of the Shallow Land Burial Handbook has just undergone review and revision; it will be issued shortly.

#### **CORRECTIVE MEASURES TECHNOLOGY**

The programmatic objective for corrective measures technology development efforts is to develop and demonstrate corrective measures technology for burial sites (Figure 4). It is anticipated that corrective measures will primarily be required at shallow land burial sites; however, these methodologies will also be applicable to greater confinement disposal sites as necessary. The major program milestone associated with this objective is to issue the final Corrective Measures Handbook in December, 1984. The Handbook will be the official technology transfer document for distribution to disposal facility operators, owners, and regulators. The manual will provide documentation of engineered techniques for implementing corrective and/or preventive action at disposal sites. Documentation will include fundamental engineered systems design and emplacement methodology, a review of current practices, and operational guidelines generic to LLW disposal sites.

#### **GREATER CONFINEMENT DISPOSAL TECHNOLOGY**

The greater confinement disposal technology element of the LLWMP includes all activities directed towards the objective "develop and demonstrate the technology for a greater confinement disposal facility" (Figure 5). The Greater Confinement Disposal Handbook, to be issued in FY 1986, will incorporate the results of the various tasks relative to this objective. This technology transfer document will be of interest primarily to potential greater confinement disposal facility developers and operators. The Handbook will be concerned with data analysis and evaluation of LLW containment in intermediate depth disposal environments, and procedures and guidelines for alternative greater confinement disposal methods.

Types of LLW requiring greater confinement disposal than shallow land burial will be explicitly identified, along with which technological option or options are the most appropriate means for achieving greater confinement disposal for specific wastes. The inventory and evaluation of alternative disposal technologies include technical and environmental acceptability, benefit/cost analyses, monitoring requirements and procedures, and disposal facility construction and emplacement techniques.

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## **CORRECTIVE MEASURES TECHNOLOGY**

- **DEVELOP AND DEMONSTRATE CORRECTIVE MEASURES TECHNOLOGY FOR BURIAL SITES**
- **ISSUE CORRECTIVE MEASURES HANDBOOK (12/84)**

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Figure 4.

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## **GREATER CONFINEMENT DISPOSAL TECHNOLOGY**

- DEVELOP AND DEMONSTRATE THE TECHNOLOGY FOR GREATER CONFINEMENT DISPOSAL
- ISSUE GREATER CONFINEMENT DISPOSAL HANDBOOK (FY 86)

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Figure 5.

## DISPOSAL TECHNOLOGY ISSUES AND ACTIVITIES

A number of technical issues related to each disposal technology element have been identified in the past. The current disposal technology activities address these issues and are based upon the recommendations of the Shallow Land Burial Steering Committee, the State-of-the-Art Review of Shallow Land Burial, annual program information meetings, and program planning meetings (Figure 6). The Shallow Land Burial Steering Committee existed from 1975 through 1977 to advise Headquarters management staff on the research requirements related to shallow land burial. The Committee was convened again in 1979 to identify and provide unresolved technical gaps and issues, which were then used to form the principal basis for the LLWMP plan and funding of activities in 1980.

The Evaluation Research Corporation prepared a state-of-the-art review of shallow land burial for the LLWMP in FY 1980. This document took the Shallow Land Burial Steering Committee's findings and used them to evaluate the current practice of SLB. Detailed technical information needed to support the Committee's conclusions was also developed.

Three annual participants' information meetings have been held which have served to bring the various contractors together to present and exchange information relative to LLWMP activities, as well as to provide direction by program and Headquarters management. Contractor participants have played an increasingly important role in program activity planning and task initiation or redirection. During, and subsequent to, these meetings, technical issues have been clarified and work restructured accordingly. Recently, these meetings have served as an interagency forum at which related research and development activities have been discussed and areas of need and overlap identified.

LLWMP planning meetings are held periodically to establish and evaluate program milestones and objectives. Two major LLWMP planning meetings have been convened. The first was held in Des Moines during FY 1980; it was during this meeting that top level program milestones were established to address overall issues. "Top-down" planning was subsequently used to develop supporting milestones.

A second major LLWMP planning session was held during FY 1982, in Salt Lake City. This meeting was devoted primarily to the determination of technology transfer needs and problems.

All of the technical issues that have been identified are being addressed by tasks currently funded at a number of contractor sites. These contractor sites are: Argonne National Laboratory (ANL), Brookhaven National Laboratory (BNL), Los Alamos National Laboratory (LANL), Oak Ridge National Laboratory (ORNL), the Nevada Test Site (NTS), Pacific Northwest Laboratory (PNL), and Savannah River Laboratory (SRL).



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## **THE CURRENT IMPROVED DISPOSAL TECHNOLOGY ACTIVITIES ARE BASED ON:**

- SLB STEERING COMMITTEE  
RECOMMENDATIONS (1979)
- STATE-OF-THE-ART REVIEW OF  
SLB (1980)
- ANNUAL INFORMATION MEETINGS
- PROGRAM PLANNING MEETINGS

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Figure 6.

One task funded by the LLWMP is being performed by another agency, the U. S. Geological Survey's Water Resources Division at the Denver (Lakewood) Federal Center.

#### IMPROVED SHALLOW LAND BURIAL

The principal technical issues related to the shallow land burial of LLW are: water management, radionuclide migration, performance monitoring and evaluation, predictive capabilities, and design, operation and management practices (Figure 7). An overall picture of the tasks underway at the various sites which address these issues is given in Figure 8.

##### Water Management

Water management includes the control of ground and surface water infiltration into buried waste, water erosion control, and the evaluation of the effects of these controls on the overall SLB system.

Three contractor sites currently are performing work related to water management: LANL, ORNL and SRL (Figure 9). The "Shallow Land Burial Technology - Arid" project underway at LANL responds to the technical issue of water management in that one aspect of the water management work consists of field testing the performance of wick systems for potential use in trench cap designs. Another portion of the water management - related effort is to evaluate laboratory, field and analytical methods for determining unsaturated hydraulic conductivities for use in modeling site performance and migration barrier selection.

The "Shallow Land Burial Technology - Humid" task at ORNL in part evaluates the performance of trench treatments designed to eliminate water intrusion problems at humid sites. An Engineered Test Facility consisting of small-scale monitored trenches has been constructed in the field. Two trench treatments are being tested; one is a liner and the other a waste fixative that encapsulates the waste from water. Status reports on the results to date of field tests will be released.

Savannah River Laboratory has a funded task that addresses water management indirectly. This task is "Migration in a Humid SLB Site."

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**THE PRINCIPAL TECHNICAL ISSUES  
RELATED TO THE SHALLOW LAND BURIAL  
OF LOW-LEVEL WASTE ARE:**

- WATER MANAGEMENT
- RADIONUCLIDE MIGRATION
- PERFORMANCE MONITORING AND EVALUATION
- PREDICTIVE CAPABILITIES
- DESIGN, OPERATION AND MANAGEMENT PRACTICES

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Figure 7.

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## RELATIONSHIP OF LLWMP TASKS TO SLB TECHNOLOGY ISSUES

| CONTRACTOR<br>SITE | WATER<br>MANAGEMENT | RADIONUCLIDE<br>MIGRATION | PERFORMANCE<br>MONITORING<br>AND<br>EVALUATION | PREDICTION<br>CAPABILITIES | DESIGN,<br>OPERATION<br>AND<br>MANAGEMENT<br>PRACTICES |
|--------------------|---------------------|---------------------------|------------------------------------------------|----------------------------|--------------------------------------------------------|
| ANL                |                     |                           | ●                                              |                            |                                                        |
| BNL                |                     | ●                         |                                                |                            |                                                        |
| LANL               | ●                   | ●                         |                                                | ●                          |                                                        |
| NTS                |                     |                           |                                                |                            |                                                        |
| ORNL               | ●                   | ●                         |                                                |                            | ●                                                      |
| PNL                |                     | ●                         | ●                                              | ●                          | ●                                                      |
| SRL                | ○                   | ●                         |                                                | ●                          |                                                        |
| USGS               |                     |                           | ●                                              |                            |                                                        |

● TASK UNDERWAY. SOME RESULTS AVAILABLE

○ TASK UNDERWAY

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Figure 8.

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## RELATIONSHIP OF LLWMP TASKS TO SLB TECHNOLOGY ISSUES

### WATER MANAGEMENT

| CONTRACTOR<br>TASK                             | ANL | BNL | LANL | NTS | ORNL | PNL | SRL | USGS |
|------------------------------------------------|-----|-----|------|-----|------|-----|-----|------|
| AL 3.5.4.0 AR<br>SLB TECHNOLOGY-ARID           |     |     | ●    |     |      |     |     |      |
| OR 3.5.4.4 AR<br>SLB TECHNOLOGY-HUMID          |     |     |      |     | ●    |     |     |      |
| SR 3.5.4.1 AR MIGRATION<br>IN A HUMID SLB SITE |     |     |      |     |      |     | ○   |      |

- TASK UNDERWAY. SOME RESULTS AVAILABLE  
○ TASK UNDERWAY

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Figure 9.

### Radionuclide Migration

Radionuclide migration includes transport by groundwater movement and dispersion. Also important are the mechanisms of radionuclide interactions in the ground including the effects of physicochemical conditions on the properties of specific radionuclides and radionuclide mobility. Site specific field and laboratory measurements pertaining to radionuclide migration that have generic applicability are included as part of this issue.

Five contractor sites are addressing the issue of radionuclide migration through six studies in progress (Figure 10). BNL is completing a task entitled "Microbial Effects at Shallow Land Burial Sites." The purpose of this project is "to determine the significance of the biodegradation of organic LLW and its effects on migration of radionuclides" from shallow land burial for both humid and arid sites. The significance of microbial degradation on the overall performance of SLB sites is being evaluated. Recommendations developed for minimizing the effect of degradation of radionuclide mobility will be available in a final document to be issued this fiscal year.

The LANL subtask, "Biointrusion Barrier Testing," of the "Shallow Land Burial Technology - Arid" task applies to radionuclide migration. Work conducted under this subtask has evaluated the effectiveness of several combinations of geologic materials in various thicknesses as biointrusion barriers.

The "Shallow Land Burial Technology - Humid" task at ORNL responds to the issue of radionuclide migration in part through the laboratory and field testing of barrier media. The two barrier materials now being evaluated in the field are Portland cement/bentonite grout and reinforced hypalon sheets.

PNL has a funded task in place which addresses radionuclide migration. "The Mobility of Organic Complexes" is investigating the effects of several complexants on the sorption properties of various radionuclides on soil materials.

The "Migration in a Humid Shallow Land Burial Site" task being conducted at the SRL has two subtasks involved with radionuclide migration. One of these, "Subsurface Monitoring," provides information on the migration of radionuclides at SRL. The "Soil-Water Chemistry" subtask is used to determine distribution coefficients using laboratory techniques and compares these results to actual measured field values. Agitation-suspension batch and soil column methods have been used to determine the  $K_d$  for strontium. Comparison to actual field data is currently underway. The "Special Waste Form Lysimeters" task also addresses radionuclide migration by the waste form leaching studies underway.

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## RELATIONSHIP OF LLWMP TASKS TO SLB TECHNOLOGY ISSUES RADIONUCLIDE MIGRATION

| CONTRACTOR<br>TASK                              | ANL | BNL | LANL | NTS | ORNL | PNL | SRL | USGS |
|-------------------------------------------------|-----|-----|------|-----|------|-----|-----|------|
| AL 3.5.4.0 AR<br>SLB TECHNOLOGY-ARID            |     |     | ●    |     |      |     |     |      |
| CH 3.5.3.0 AR MICROBIAL<br>EFFECTS AT SLB SITES |     | ●   |      |     |      |     |     |      |
| OR 3.5.4.4 AR<br>SLB TECHNOLOGY-HUMID           |     |     |      |     | ●    |     |     |      |
| RL 3.5.3.1 AR MOBILITY<br>ORGANIC COMPLEXES     |     |     |      |     |      | ●   |     |      |
| SR 3.5.3.2 AR SPECIAL<br>WASTE FORM LYSIMETERS  |     |     |      |     |      |     | ●   |      |
| SR 3.5.4.1 AR MIGRATION<br>IN A HUMID SLB SITE  |     |     |      |     |      |     | ○   |      |

- TASK UNDERWAY, SOME RESULTS AVAILABLE  
○ TASK UNDERWAY

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Figure 10.

### Performance Monitoring and Evaluation

The technical issue of performance monitoring and evaluation is being resolved by efforts including the conceptual design of monitoring systems, development of performance standards, and the development of monitoring technology and methodology. Contractor sites performing related work are shown in Figure 11.

ANL has compiled a handbook for generic disposal site monitoring and monitoring system evaluation. This "Generic Disposal Site Monitoring" task was initiated to develop an environmental monitoring program design for use at future LLW disposal sites. The handbook, which specifically identifies how to design and operate an environmental monitoring program, will be finalized in FY 1982.

PNL's "Well Logging Instrumentation" task supports performance monitoring and evaluation by investigating techniques for in-situ measurements of various radioisotopes. Neutron activation analysis, mass spectrometry, gas proportional counting and laser excitation experiments have been and are being performed as part of the work effort.

The U. S. Geological Survey's Water Resources Division is currently developing analytical and on-site techniques for evaluating potential LLW disposal sites and investigating radionuclide migration under the "USGS/Denver Borehole Geophysics" task. Researchers are evaluating borehole-wall imaging, acoustic determination of rock properties, gamma spectroscopy, flowmeter analysis of permeability distribution and well log resolution.

### Predictive Capabilities

Tasks directed towards improving predictive capabilities include the development, verification, and validation of comprehensive site performance models. These models will depict all significant pathways of radionuclide transport from LLW disposal sites and radionuclide transport via specific pathways. Program activities are focused on the application and modification of existing models rather than the development of new models. Contractor sites conducting prediction capabilities studies are shown in Figure 12.

Prediction capability activities are conducted at LANL under the "Model Verification" subtask of "Shallow Land Burial Technology - Arid". The LANL unsaturated, two-dimensional flow transport model has been evaluated for accuracy in predicting the performance of an arid SLB site. The model has also been applied to the Greater Confinement Disposal Facility (GCDF) at the Nevada Test Site.



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## RELATIONSHIP OF LLWMP TASKS TO SLB TECHNOLOGY ISSUES

### PERFORMANCE MONITORING AND EVALUATION

| CONTRACTOR<br>TASK                                | ANL | BNL | LANL | NTS | ORNL | PNL | SRL | USGS |
|---------------------------------------------------|-----|-----|------|-----|------|-----|-----|------|
| CH 3.5.3.1 AR GENERIC<br>DISPOSAL SITE MONITORING | ●   |     |      |     |      |     |     |      |
| ID 3.5.3.0 AR USGS/DENVER<br>BOREHOLE GEOPHYSICS  |     |     |      |     |      |     |     | ●    |
| RL 3.5.3.2 AR WELL<br>LOGGING INSTRUMENTATION     |     |     |      |     |      | ●   |     |      |

● TASK UNDERWAY, SOME RESULTS AVAILABLE

Figure 11.

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## RELATIONSHIP OF LLWMP TASKS TO SLB TECHNOLOGY ISSUES

### PREDICTION CAPABILITIES

| CONTRACTOR<br>TASK                             | ANL | BNL | LANL | NTS | ORNL | PNL | SRL | USGS |
|------------------------------------------------|-----|-----|------|-----|------|-----|-----|------|
| AL 3.5.4.0 AR<br>SLB TECHNOLOGY-ARID           |     |     | ●    |     |      |     |     |      |
| RL 3.5.3.4 AP TRANSPORT<br>ASSESSMENT-ARID     |     |     |      |     |      | ●   |     |      |
| RL 3.5.3.5 AR TRANSPORT<br>MODEL GUIDELINES    |     |     |      |     |      | ●   |     |      |
| SR 3.5.4.1 AR MIGRATION<br>IN A HUMID SLB SITE |     |     |      |     |      |     | ●   |      |

● TASK UNDERWAY. SOME RESULTS AVAILABLE

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Figure 12.

"Transport Assessment - Arid" is one of the PNL tasks concerned with prediction capabilities. The objective is to provide the technology necessary to characterize water and radionuclide transport through SLB sites. Specific project objectives include the characterization of unsaturated flow processes to enhance the reliability of transport rate estimation, and the verification of the unsaturated flow transport model using experimental data sets. PNL's other task involved with prediction capabilities is "Transport Model Guidelines." This task is aimed at developing the necessary guidelines for the documentation and evaluation of groundwater transport models relevant to SLB sites. The adequacy of the guidelines is being evaluated through comparison of experimental data test cases with LLW systems transport models.

SRL is employing the "Dose-to-Man" model under the "Migration in a Humid Shallow Land Burial Site" task. This model is being used to predict the performance of the LLW disposal sites after the period of institutional control has expired. It has been applied at the Savannah River Plant.

#### Design, Operation and Management Practices

The technology need related to design, operation and management practices is concerned with the development of guidelines to facilitate the implementation of optimal practices for all aspects of facility operation. This issue also includes the area of waste inventory records, which consists of developing waste characterization techniques at the source, the receiving site, and in-situ. Two sites are currently pursuing activities in support of this technical issue (Figure 13).

The "Shallow Land Burial Handbook" task being performed at ORNL is concerned with the development of guidelines to facilitate the implementation of good management practices for all aspects of disposal facility operation. This preliminary handbook, which will be issued in FY 1982, will serve as a guide for site developers, operators and regulators.

The Engineered Test Facility, part of ORNL's "Shallow Land Burial Technology - Humid" activities, has been instrumented with monitoring devices. Continuous records are being maintained for the evaluation of the hydrologic system of a typical humid eastern SLB site, and to determine monitoring and data collection needs for recommended management practices. Results documented in a progress report will be issued shortly.

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## RELATIONSHIP OF LLWMP TASKS TO SLB TECHNOLOGY ISSUES

### DESIGN, OPERATION AND MANAGEMENT PRACTICES

| CONTRACTOR<br>TASK                            | ANL | BNL | LANL | NTS | ORNL | PNL | SRL | USGS |
|-----------------------------------------------|-----|-----|------|-----|------|-----|-----|------|
| OR 3.5.4.3 AR<br>SLB HANDBOOK                 |     |     |      |     | ●    |     |     |      |
| OR 3.5.4.4 AR<br>SLB TECHNOLOGY-HUMID         |     |     |      |     | ○    |     |     |      |
| RL 3.5.4.0 AR VERIFICATION<br>INSTRUMENTATION |     |     |      |     |      | ●   |     |      |

- TASK UNDERWAY, SOME RESULTS AVAILABLE
- TASK UNDERWAY

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Figure 13.

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**THE PRINCIPAL TECHNICAL ISSUES RELATED  
TO CORRECTIVE MEASURES ARE  
CATEGORIZED AS FOLLOWS:**

- WATER MANAGEMENT
- SUBSIDENCE
- WIND EROSION
- BIOLOGICAL INTRUSION
- RADIONUCLIDE MIGRATION

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Figure 14.

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## RELATIONSHIP OF LLWMP TASKS TO CORRECTIVE MEASURES TECHNOLOGY ISSUES

| CONTRACTOR<br>SITE | WATER<br>MANAGEMENT | SUBSIDENCE | WIND<br>EROSION | BIOLOGICAL<br>INTRUSION | RADIOISOTOPE<br>MIGRATION |
|--------------------|---------------------|------------|-----------------|-------------------------|---------------------------|
| ANL                |                     |            |                 |                         |                           |
| BNL                |                     |            |                 |                         |                           |
| LANL               | ○                   | ○          |                 | ●                       |                           |
| NTS                |                     |            |                 |                         |                           |
| ORNL               | ●                   | ●          | ●               | ●                       | ●                         |
| PNL                |                     |            |                 |                         |                           |
| SRL                |                     |            |                 |                         |                           |
| USGS               |                     |            |                 |                         |                           |

● TASK UNDERWAY, SOME RESULTS AVAILABLE

○ TASK UNDERWAY

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Figure 15.

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## RELATIONSHIP OF LLWMP TASKS TO CORRECTIVE MEASURES TECHNOLOGY ISSUES

### WATER MANAGEMENT

| CONTRACTOR<br>TASK                                | ANL | BNL | LANL | NTS | ORNL | PNL | SRL | USGS |
|---------------------------------------------------|-----|-----|------|-----|------|-----|-----|------|
| AL 3.5.4.2 AR REMEDIAL<br>ACTION TECHNOLOGY-ARID  |     |     | ○    |     |      |     |     |      |
| OR 3.5.4.1 AR SLB<br>REMEDIAL ACTION MANUAL       |     |     |      |     | ●    |     |     |      |
| OR 3.5.4.2 AR REMEDIAL<br>ACTION TECHNOLOGY-HUMID |     |     |      |     | ○    |     |     |      |

- TASK UNDERWAY, SOME RESULTS AVAILABLE  
○ TASK UNDERWAY

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Figure 16.

PNL's task entitled "Verification Instrumentation" has the objective to organize, apply, and develop instrumentation for assaying radionuclide content of LLW packages received by disposal site operators. The physical, chemical and radiological parameters which are quantitatively important for recordkeeping are being established. Standard waste sources for use in instrumentation calibration and testing will be defined this fiscal year.

## CORRECTIVE MEASURES

The principal technical issues related to corrective measures technology development are: water management, subsidence, wind erosion, biological intrusion, and radionuclide phase migration (Figure 14). These issues are addressed by work underway at LANL and ORNL (Figure 15). Additional corrective measures work at the Maxey Flats site is being managed by EG&G Idaho. It should be noted that the "Remedial Action Manual" task at ORNL responds to all identified corrective measures issues. The purpose of the manual is to document corrective measures applicable to improved operational performance of LLW disposal sites in a readily accessible format.

### Water Management

Water management is as previously defined.

LANL is performing water management work as part of the "Remedial Action Technology - Arid" task (Figure 16). Both surface and groundwater management activities are included.

Trench cap studies are in progress at LANL, which include actions taken to reduce infiltration and promote runoff. Determination of the soil erodibility factors and vegetation cover management are being conducted for use in predicting surface water erosion potential. Drainage ditches and stabilizing riprap have been installed.

The major corrective measures effort at LANL taken to support groundwater management has been to determine the extent of contaminant transport to the surface due to moisture cycling. A summary of results is due out at the end of this calendar year.

ORNL's "Remedial Action Technology - Humid" task is also concerned with both surface water and groundwater management problems. The purpose of this task is to develop and demonstrate potential corrective measures for hydrologic contamination problems at humid LLW disposal sites. Related subtasks include the design and development of a passive subsurface drainage system, and grouting technology development and demonstration.

EG&G Idaho is managing a trench cap demonstration effort underway at the closed Maxey Flats commercial LLW site. This water management-related work utilized a synthetic cover to eliminate infiltration and halt the subsequent "bathtub effect" occurring in site trenches.



### Subsidence

Subsidence is caused by the settling and compaction of trench backfill material into voids and the decomposition of waste and waste packages. Subsidence is a significant technical issue since it can lead to ponding, increased infiltration through the breached trench surface, or even direct exposure of waste materials.

Corrective measures work at LANL involves the effects of subsidence on burial systems, and on trench cap performance in particular (Figure 17). Work elements include the determination and demonstration of the results of subsidence on biointrusion barrier performance, wick systems performance, and migration barriers and integrated systems.

The grouting technology studies ongoing at ORNL in part respond to the technical issue of subsidence. The encapsulating grout material will provide waste stability and prevent compaction (and subsidence) as well as isolating the buried wastes from water intrusion.

A portion of the EG&G Idaho-managed effort at Maxey Flats, Kentucky is a new initiative whereby grout will be emplaced into trenches. The grouting effort will in part eliminate subsidence by stabilizing the trench and its contents.

### Wind Erosion

Wind erosion poses a potential threat to the integrity of mounded trench caps in arid regions with sparse vegetation. Wind also serves as a potential transport mechanism for contaminated soil and wastes from both open working trenches and eroded closed trenches.

The issue of wind erosion is currently being addressed by efforts undertaken for the "Remedial Action Manual" at ORNL (Figure 18).

### Biological Intrusion

Biological intrusion by plants and animals may result in the uptake of radionuclides from buried LLW, or the direct exposure or removal of the waste. Vegetative uptake is a common route for the dispersion of radionuclides, and can occur through direct contact with the buried materials or with contaminated groundwater. Rodents and other small animals can burrow into waste trenches and spread contamination to the surface. Burrows also provide routes by which surface water can be channeled into the waste.

The technical issue of biological intrusion is being addressed at LANL by studies evaluating the effectiveness of gravel and cobble plant and animal barriers (Figure 19). Field testing of corrective measures using caissons is ongoing; further work is to be tested by application to a closed trench.

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## RELATIONSHIP OF LLWMP TASKS TO CORRECTIVE MEASURES TECHNOLOGY ISSUES

### SUBSIDENCE

| CONTRACTOR<br>TASK                                | ANL | BNL | LANL | NTS | ORNL | PNL | SRL | USGS |
|---------------------------------------------------|-----|-----|------|-----|------|-----|-----|------|
| AL 3.5.4.2 AR REMEDIAL<br>ACTION TECHNOLOGY-ARID  |     |     | ○    |     |      |     |     |      |
| OR 3.5.4.1 AR SLB<br>REMEDIAL ACTION MANUAL       |     |     |      |     | ●    |     |     |      |
| OR 3.5.4.2 AR REMEDIAL<br>ACTION TECHNOLOGY-HUMID |     |     |      |     | ○    |     |     |      |

- TASK UNDERWAY, SOME RESULTS AVAILABLE
- TASK UNDERWAY

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Figure 17.

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## RELATIONSHIP OF LLWMP TASKS TO CORRECTIVE MEASURES TECHNOLOGY ISSUES

### WIND EROSION

| CONTRACTOR<br>TASK                          | ANL | BNL | LANL | NTS | ORNL | PNL | SRL | USGS |
|---------------------------------------------|-----|-----|------|-----|------|-----|-----|------|
| OR 3.5.4.1 AR SLB<br>REMEDIAL ACTION MANUAL |     |     |      |     | ●    |     |     |      |

● TASK UNDERWAY, SOME RESULTS AVAILABLE

Figure 18.

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## RELATIONSHIP OF LLWMP TASKS TO CORRECTIVE MEASURES TECHNOLOGY ISSUES

### BIOLOGICAL INTRUSION

| CONTRACTOR<br>TASK                               | ANL | BNL | LANL | NTS | ORNL | PNL | SRL | USGS |
|--------------------------------------------------|-----|-----|------|-----|------|-----|-----|------|
| AL 3.5.4.2 AR REMEDIAL<br>ACTION TECHNOLOGY-ARID |     |     | ●    |     |      |     |     |      |
| OR 3.5.4.1 AR SLB<br>REMEDIAL ACTION MANUAL      |     |     |      |     | ●    |     |     |      |

● TASK UNDERWAY, SOME RESULTS AVAILABLE

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Figure 19.

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DOE LOW-LEVEL WASTE ADAPT. PROGRAM  
DENVER, COLORADO - 8/29-9/2/82

WASTE DISPOSAL  
OVERVIEW

## RELATIONSHIP OF LLWMP TASKS TO CORRECTIVE MEASURES TECHNOLOGY ISSUES

### RADIONUCLIDE MIGRATION

| CONTRACTOR<br>TASK                                | ANL | BNL | LANL | NTS | ORNL | PNL | SRL | USGS |
|---------------------------------------------------|-----|-----|------|-----|------|-----|-----|------|
| OR 3.5.4.1 AR SLB<br>REMEDIAL ACTION MANUAL       |     |     |      |     | ●    |     |     |      |
| OR 3.5.4.2 AR REMEDIAL<br>ACTION TECHNOLOGY-HUMID |     |     |      |     | ○    |     |     |      |

- TASK UNDERWAY, SOME RESULTS AVAILABLE
- TASK UNDERWAY

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Figure 20.

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DENVER, COLORADO - 8/31-9/2/82

WASTE DISPOSAL  
OVERVIEW

## **THE PRINCIPAL TECHNICAL ISSUES RELATED TO GREATER CONFINEMENT DISPOSAL ARE:**

- WATER MANAGEMENT
- RADIONUCLIDE MIGRATION
- PERFORMANCE MONITORING AND EVALUATION
- PREDICTION CAPABILITIES
- DESIGN, OPERATION AND MANAGEMENT PRACTICES

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Figure 21.

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WASTE DISPOSAL  
OVERVIEW

## RELATIONSHIP OF LLWMP TASKS TO GREATER CONFINEMENT DISPOSAL TECHNOLOGY ISSUES

| CONTRACTOR<br>SITE | WATER<br>MANAGEMENT | RADIONUCLIDE<br>MIGRATION | PERFORMANCE<br>MONITORING<br>AND<br>EVALUATION | PREDICTION<br>CAPABILITIES | DESIGN,<br>OPERATION<br>AND<br>MANAGEMENT<br>PRACTICES |
|--------------------|---------------------|---------------------------|------------------------------------------------|----------------------------|--------------------------------------------------------|
| ANL                |                     |                           |                                                |                            |                                                        |
| BNL                |                     |                           |                                                |                            |                                                        |
| LANL               | ○                   | ○                         | ○                                              | ●                          | ○                                                      |
| NTS                | ○                   | ○                         | ●                                              | ●                          | ●                                                      |
| ORNL               |                     |                           |                                                |                            |                                                        |
| PNL                |                     |                           |                                                |                            |                                                        |
| SRL                |                     |                           | ●                                              | ●                          | ●                                                      |
| USGS               |                     |                           |                                                |                            |                                                        |

● TASK UNDERWAY, SOME RESULTS AVAILABLE

○ TASK UNDERWAY

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Figure 22.

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WASTE DISPOSAL  
OVERVIEW

## RELATIONSHIP OF LLWMP TASKS TO GREATER CONFINEMENT DISPOSAL TECHNOLOGY ISSUES

### WATER MANAGEMENT

| CONTRACTOR<br>TASK                                        | ANL | BNL | LANL | NTS | ORNL | PNL | SRL | USGS |
|-----------------------------------------------------------|-----|-----|------|-----|------|-----|-----|------|
| AL 3.5.4.4 AR TECHNICAL<br>SUPPORT NON-SLB                |     |     | ○    |     |      |     |     |      |
| NV 3.5.4.0 AR GREATER<br>CONFINEMENT DISPOSAL<br>FACILITY |     |     |      | ○   |      |     |     |      |

○ TASK UNDERWAY

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Figure 23.



The trench grouting demonstration at Maxey Flats, Kentucky will result in the prevention of biological intrusion.

### Radionuclide Migration

Radionuclide migration results in the release of radionuclides to the environment. Gas phase migration is a primary transport mechanism. Radioactive methane, carbon dioxide and hydrogen sulfide can migrate through trench caps. Tritium can also diffuse through the soil or through open fractures as tritiated water vapor.

Two subtasks of the "Remedial Action Technology - Humid" task at ORNL are related to radionuclide migration (Figure 20). One of these concerns the chemical treatment of trenches; the other involves the chemical treatment of contaminated seep areas. For both of these studies a chemical solution is applied which increases the sorptive capacity of trench backfill or area soils for radionuclides.

The trench grouting demonstration at Maxey Flats, Kentucky will serve to show that grout emplacement inhibits radionuclide leaching and transport, as well as preventing subsidence and biological intrusion.

### GREATER CONFINEMENT DISPOSAL

The principal technical issues related to greater confinement disposal are the same as those given before for improved shallow land burial: water management, radionuclide migration, performance monitoring and evaluation, prediction capabilities, and design, operation and management practices (Figure 21).

The Nevada Test Site is the focus of most of the work currently underway related to the greater confinement disposal concept (Figure 22). The purpose of the Greater Confinement Disposal Facility (GCDF) at the NTS is to demonstrate the economics and environmental acceptability of greater confinement of low-level, higher activity or long-lived radioactive wastes. Activities undertaken at DOE/NV attention to all of the identified technical issues.

### Water Management

Water management efforts are being conducted by LANL and DOE/NV (Figure 23).

Water management-related activities are underway at the NTS, in support of hydrologic monitoring systems.

LANL is providing technical support to the GCDF at the NTS in the area of water management. One purpose of the "Technical Support Non-Shallow Land Burial" task is to evaluate water management systems for applicability to non-SLB facility sites.

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WASTE DISPOSAL  
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## RELATIONSHIP OF LLWMP TASKS TO GREATER CONFINEMENT DISPOSAL TECHNOLOGY ISSUES

### RADIONUCLIDE MIGRATION

| CONTRACTOR<br>TASK                                        | ANL | BNL | LANL | NTS | ORNL | PNL | SRL | USGS |
|-----------------------------------------------------------|-----|-----|------|-----|------|-----|-----|------|
| AL 3.5.4.4 AR TECHNICAL<br>SUPPORT NON-SLB                |     |     | ○    |     |      |     |     |      |
| NV 3.5.4.0 AR GREATER<br>CONFINEMENT DISPOSAL<br>FACILITY |     |     |      | ○   |      |     |     |      |

○ TASK UNDERWAY

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Figure 24.

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WASTE DISPOSAL  
OVERVIEW

## RELATIONSHIP OF LLWMP TASKS TO GREATER CONFINEMENT DISPOSAL TECHNOLOGY ISSUES

### PERFORMANCE MONITORING AND EVALUATION

| CONTRACTOR<br>TASK                                                    | ANL | BNL | LANL | NTS | ORNL | PNL | SRL | USGS |
|-----------------------------------------------------------------------|-----|-----|------|-----|------|-----|-----|------|
| AL 3.5.4.4 AR TECHNICAL<br>SUPPORT NON-SLB                            |     |     | ○    |     |      |     |     |      |
| NV 3.5.4.0 AR GREATER<br>CONFINEMENT DISPOSAL<br>FACILITY             |     |     |      | ●   |      |     |     |      |
| SR 3.5.4.0 AR GREATER<br>CONFINEMENT DISPOSAL<br>INVESTIGATIONS-HUMID |     |     |      |     |      |     | ●   |      |

● TASK UNDERWAY. SOME RESULTS AVAILABLE

○ TASK UNDERWAY

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Figure 25.

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WASTE DISPOSAL  
OVERVIEW

## RELATIONSHIP OF LLWMP TASKS TO GREATER CONFINEMENT DISPOSAL TECHNOLOGY ISSUES

### PREDICTION CAPABILITIES

| CONTRACTOR<br>TASK                                                    | ANL | BNL | LANL | NTS | ORNL | PNL | SRL | USGS |
|-----------------------------------------------------------------------|-----|-----|------|-----|------|-----|-----|------|
| AL 3.5.4.4 AR TECHNICAL<br>SUPPORT NON-SLB                            |     |     | ●    |     |      |     |     |      |
| NV 3.5.4.0 AR GREATER<br>CONFINEMENT DISPOSAL<br>FACILITY             |     |     |      | ●   |      |     |     |      |
| SR 3.5.4.0 AR GREATER<br>CONFINEMENT DISPOSAL<br>INVESTIGATIONS-HUMID |     |     |      |     |      |     | ●   |      |

● TASK UNDERWAY. SOME RESULTS AVAILABLE

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Figure 26.

### Radionuclide Migration

LANL and DOE/NV are also conducting studies related to radionuclide migration (Figure 24).

The issue of radionuclide migration is being addressed by DOE/NV under the "Greater Confinement Disposal Facility" task.

LANL is providing assistance in the area of radionuclide migration by evaluating SLB biointrusion barriers and migration barriers for use at non-SLB disposal sites. Field testing will be conducted at the NTS to obtain basic information on the mechanisms of radionuclide transport.

### Performance Monitoring and Evaluation

Three sites, LANL, DOE/NV, and SRL are conducting performance monitoring and evaluation-related activities (Figure 25).

The installation of monitoring instrumentation is being completed at the NTS, with routine monitoring to begin in FY 1983.

LANL is supporting the NTS effort by obtaining information on monitoring instrument placement and performance at the GCDF.

SRL is undertaking a task to develop specific methods for selecting GCD sites in humid regions. A portion of the "Greater Confinement Disposal Investigations - Humid" project will address the issue of performance monitoring and evaluation by monitoring a GCD site at the SRP. Transport of toxic nonradioactive contaminants as well as radionuclides will be monitored.

### Prediction Capabilities

LANL, DOE/NV, and SRL are also involved with studies addressing prediction capabilities (Figure 26).

Arid site performance modeling activities have been conducted by DOE/NV as part of the "Greater Confinement Disposal Facility" task.

One purpose of LANL's "Technical Support Non-Shallow Land Burial" task is to provide site screening and evaluation models. An experimental test plan for computer modeling of the GCDF performance at the NTS is under development.

SRL is utilizing computer models to ensure the design development of a planned near-zero release LLW GCD facility.

### Design, Operation and Management Practices

The GCD at the NTS will demonstrate greater confinement in an arid environment by disposal using the large diameter drilled hole concept (Figure 27). Final construction of surface facilities will be completed this fiscal year. Planning for other management practices such as waste transport and handling and in-situ waste inventory recordkeeping have been an FY 1982 activity.

Post-closure activities for non-SLB arid sites are being developed and evaluated by LANL under its "Technical Support Non-Shallow Land Burial" task.

The "Greater Confinement Disposal Investigations - Humid" task underway at SRL addresses general management practices. Work efforts include the testing of GCD facility site selection by selecting a site at the SRP. A major subtask is humid site GCD facility design development and demonstration. The demonstration will establish that greater depth burial combined with engineered barriers and improved waste form is cost-effective and environmentally acceptable for disposing of LLW nitrate salts.

### CONCLUSIONS

The current status of LLW technology development is as follows (Figure 28). First of all, the program has identified the significant technical issues related to disposal technology development and is addressing them by activities undertaken at various sites across the country. The program initially identified a set of research needs and then focused efforts and resources on resolving those needs. The technology development efforts of the LLWMP are primarily in the stage of development, demonstration and confirmation at present, with the bulk of basic research activities already conducted. At this time, program emphasis is beginning to shift from technology development to technology transfer; future work should reflect this concept.

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WASTE DISPOSAL  
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## RELATIONSHIP OF LLWMP TASKS TO GREATER CONFINEMENT DISPOSAL TECHNOLOGY ISSUES

### DESIGN, OPERATION AND MANAGEMENT PRACTICES

| CONTRACTOR<br>TASK                                                    | ANL | BNL | LANL | NTS | ORNL | PNL | SRL | USGS |
|-----------------------------------------------------------------------|-----|-----|------|-----|------|-----|-----|------|
| AL 3.5.4.4 AR TECHNICAL<br>SUPPORT NON-SLB                            |     |     | ○    |     |      |     |     |      |
| NV 3.5.4.0 AR GREATER<br>CONFINEMENT DISPOSAL<br>FACILITY             |     |     |      | ●   |      |     |     |      |
| SR 3.5.4.0 AR GREATER<br>CONFINEMENT DISPOSAL<br>INVESTIGATIONS-HUMID |     |     |      |     |      |     | ●   |      |

- TASK UNDERWAY. SOME RESULTS AVAILABLE
- TASK UNDERWAY

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Figure 27.

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WASTE DISPOSAL  
OVERVIEW

**THE CURRENT STATUS OF LLWMP  
WASTE DISPOSAL TECHNOLOGY  
ACTIVITIES IS AS FOLLOWS:**

- THE PROGRAM HAS IDENTIFIED A NUMBER OF ISSUES RELATED TO LLWMP WASTE DISPOSAL TECHNOLOGIES.
- ALL OF THESE ISSUES ARE CURRENTLY BEING ADDRESSED BY TASKS PERFORMED AT VARIOUS CONTRACTOR SITES.

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Figure 28.



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LOW-LEVEL WASTE CRITERIA DEVELOPMENT  
FOR DRAFT DOE ORDER 5820

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INTRODUCTION

EG&G Idaho, Inc., as lead contractor for the Department of Energy's National Low-Level Waste Management Program, is proposing criteria for the shallow land burial of low-level radioactive waste generated by Department of Energy efforts. This work was initiated in 1979 following a request by Department of Energy Headquarters to develop uniform guidelines for the management of its low-level radioactive waste. The criteria are to be incorporated into DOE Order 5820 - Radioactive Waste Management.

The criteria were divided into six areas; site selection, site design, site operations, waste acceptance, site closure /post-closure and corrective measures. The criteria are based on a concept of low-level waste management which recognizes the use of system performance objectives. The performance objectives for the shallow land burial of low-level radioactive waste are the limits of contaminant release to the accessible environment. Performance objectives are the ground rules for land disposal and the criteria are written to ensure that the performance objectives will be met. Where applicable, technical evaluation factors were developed. Technical evaluation factors are those items which must be examined to determine whether a criterion is being met.

Committees, composed of recognized experts in the criteria area of interest, were used to develop the shallow land burial criteria. An initial meeting was held, primarily as a brainstorming session, from which a preliminary draft of the criteria was generated. Following a period of internal review, a second meeting was held to finalize the draft criteria. The draft criteria were then delivered to Department of Energy - Idaho Operations Office for their review and comment and for transmittal to Field Offices for additional review and comment. The criteria were then modified in light of these comments and incorporated into the proposed draft DOE Order 5820.

PERFORMANCE OBJECTIVES

Limits on contaminant release to the accessible environment provide the performance objectives for land disposal of low-level radioactive waste. Performance objectives are given as the maximum dose, specified in time and space, an individual may receive from radioactive waste disposal.

They are specified for individuals within the site boundary during the time period in which access to the site is controlled and for the individual outside the site for the performance period.

Selecting time periods for performance objectives is key to determining what wastes are acceptable for land disposal based on the half-life or decay time associated with that waste. Thus, a concentration limit can be set for each radionuclide acceptable for burial at low-level waste disposal sites, such that all performance standards will be met.

The performance objectives proposed for draft DOE Order 5820 are:

1. The annual dose from radionuclides released from any shallow land burial site to persons outside the site boundary from all pathways shall not exceed the following:
  - 25 mrem/yr - - whole body
  - 75 mrem/yr - - thyroid
  - 25 mrem/yr - - any organ
2. Waste shall not be disposed of in a shallow land burial site that results in a waste inventory that poses a potential threat to health and safety after a performance period of 500 years has ended.
3. Every shallow land burial site shall be operated in conformance with Department of Energy occupational exposure standards.

#### SHALLOW LAND BURIAL CRITERIA

The following discussion concentrates on the pertinent points in each of the six criteria areas. Specific criterion are not given, as the proposed draft order is now undergoing review and revision by Department of Energy Headquarters.

#### SITE SELECTION CRITERIA

The site selection criteria require a new site to have sufficient area and depth to accommodate the projected volume of waste and a buffer zone completely surrounding the waste. The buffer zone must be of sufficient size to ensure that radioactivity at the site boundary are within performance objectives. Burial of the waste must be accomplished outside the transition zone between the unsaturated and saturated zone. The hydrogeology of a new site must be amenable to reliable predictions of site performance. Geologic hazards, erosion and flooding must not jeopardize site performance. Population distribution, land use, earth and water chemistry, transportation, and utilities should be considered in selecting any new site. Applicable federal laws and regulations must be considered and complied with.

## SITE DESIGN CRITERIA

The site design criteria, like the site selection criteria, are applicable to new shallow land burial sites. An assumption was made in developing the criteria that a new site subject to the design criteria would be selected consistent with the site selection criteria. In general the criteria require that engineering standard practices and codes be followed and that additional data would be obtained where needed to support the design of the facility. Facility design requirements are specified for the following site design element: excavations, backfills, and covers; utilities and support facilities including roads, decontamination areas and receiving areas; security systems, water management and erosion control systems; a waste locator system and monitoring systems.

## SITE OPERATING CRITERIA

Site operating criteria is intended to be applicable to both presently operating sites and new sites. The criteria specify that operating procedures be developed for normal operations including volume reduction, space utilization and waste inspection, contingencies, emergency response, training and surveillance. Criteria for monitoring programs are given that cover data requirements and reporting and record keeping requirements. Program requirements for equipment and facility maintenance are also covered in the criteria.

## WASTE ACCEPTANCE CRITERIA

Waste acceptance criteria are standard rules by which the acceptability of a low-level waste package at a shallow land burial facility can be judged. Both site specific and generic criteria are given. Generic criteria are applicable at all DOE sites as written. The site specific criteria recognize the natural and engineered variability of Department of Energy disposal sites. They require a determination to be made, based on the site performance characteristics, of what wastes are acceptable such that the criterion is met.

Generic criteria include criteria for criticality, thermal or radiative output, pathogenic, infectious or toxic materials, ignitable, reactive or corrosive materials, gaseous waste, free liquids, and surface contamination. Site specific criteria include criteria for establishing radionuclide concentration limits, chelating agents, structural stability, shipping papers, and waste testing.

## CLOSURE/POST-CLOSURE CRITERIA

The objective of the proposed closure and post-closure criteria is to serve as a basis for decisions pertaining to the closure/post-closure of shallow land burial sites. By satisfying the criteria, designs and

procedures will result that provide effective containment of the waste. The design and procedures used to close the site will be dependent upon site-specific characteristics and the length of time the waste must be confined before it no longer represents a hazard to the biosphere. The closure/post-closure criteria include requirements for a closure plan that covers the following elements: surface and subsurface water management, erosion control, site stabilization, decontamination, security, surface markers, monitoring and surveillance.

#### CORRECTIVE MEASURES CRITERIA

Corrective measures criteria were developed to provide standard rules for addressing potential problems which may be encountered with low-level radioactive waste disposed of by shallow land burial during the operational, closure and institutional control phase of the post-closure period. These criteria require that a corrective measures plan be developed that addresses graded levels of response for potential problems which may arise due to erosion, flooding, ponding of the site, exposure of waste in trenches, plant or animal intrusion, and chemical or biological reactions.

#### CONCLUSIONS

The shallow land burial of Department of Energy low-level waste must be accomplished in a manner that protects public health and safety. This can be attained by selecting, designing, operating, and closing shallow land burial sites in a manner such that radiation levels both within and outside the site boundary are within limits established to provide a significant margin of safety, i.e. the performance objectives. A system's concept of waste management, in which site selection, design, operation, and closure/post-closure are approached as a function of each other, is an effective means for optimizing safe, efficient and economical shallow land burial practices. This concept is the basis of the shallow land burial criteria developed for DOE Order 5820.

EPA RISK ASSESSMENT MODEL FOR SHALLOW LAND DISPOSAL  
OF LOW LEVEL RADIOACTIVE WASTES\*

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ABSTRACT

PRESTO (Prediction of Radiation Effects from Shallow Trench Operations) is a computer code developed under U.S. Environmental Protection Agency (EPA) funding to evaluate possible health effects from shallow land burial of low-level radioactive wastes. The model is intended to assess radionuclide transport, ensuing exposure, and health impact to a static local population for a period (nominally 1000 years) following the end of burial operations. Human exposure scenarios that may be evaluated include normal releases (including leaching and operational spillage), human intrusion, and near site farming. Pathways and processes of transit from the trench to an individual or population include groundwater transport, overland flow, erosion, surface water dilution, resuspension, atmospheric transport, deposition, inhalation, and ingestion of contaminated beef, milk, crops, and water off-site. Population and individual doses and risks may be calculated as well as doses and risks to the intruder and farmer. Collective health effects in terms of deaths from cancer are calculated for the population over the assessment period using a life-table approach. Data have been compiled for three extant shallow land burial sites: Barnwell, South Carolina; Beatty, Nevada; and West Valley, New York.

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\*This study sponsored by the U.S. Environmental Protection Agency under interagency agreement EPA-D-89-F-000-60 with U.S. Department of Energy under contract W-7405-eng-26 with Union Carbide Corp. Nuclear Division.

## INTRODUCTION

The U.S. Environmental Protection Agency (EPA) is responsible for developing a generally applicable environmental standard for disposal of low-level radioactive waste (LLW). PRESTO (Prediction of Radiation Effects from Shallow Trench Operations) is a computer code developed under EPA funding to evaluate possible health effects from shallow trenches for 1000 years following the end of LLW burial operations. PRESTO is being documented (1); readers interested in more of the operational details than are given herein should consult that document.

Although designed for use by EPA, the PRESTO code may be of interest to others. Agencies or persons comparing various energy technologies from a risk assessment perspective may have potential applications for the code.

PRESTO allows numerous human exposure scenarios depending on the structure of the input data set. Pathways and processes of transit from the trench to an individual or population include: groundwater transport, overland flow, erosion, surface water (stream) dilution, resuspension, atmospheric dispersion, deposition onto crops and surfaces, inhalation, and ingestion of contaminated beef, milk, crops, and water. Pathways of transport by air or water, respectively, are shown in Figs. 1 and 2. Cancer risks are calculated for individuals or populations over the assessment period by utilizing a life-table approach, the DARTAB subroutine (2).

The model is simple and input data requirements are relatively small. The code construction is modular to allow for more detailed submodels or subroutines to be substituted at later dates. The code may be suited for use at other locations if data are available. Several of the submodels included in PRESTO were developed for other types of assessments, and we have adapted them for use in assessing health effects from shallow land burial.

Important guidelines during development of the code include:

- + The radionuclide inventory is the amount of activity found in the trenches at the end of the burial operation.
- + Waste materials in the trench are considered to be uncanistered and effectively homogeneous both in terms of radionuclide distribution and type of material.
- + Ingrowth of daughter products of radiological decay is not calculated by the model.
- + Partitioning of radionuclides between soil and water and retardation of radionuclide transport velocity is described by use of the distribution coefficient,  $K_d$ .

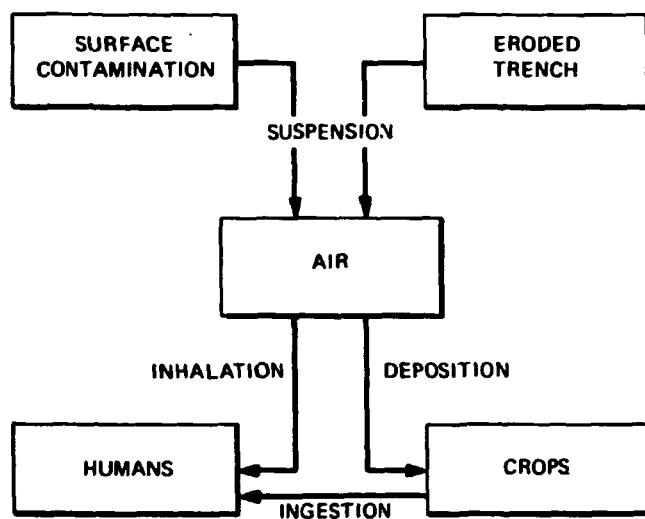


Fig. 1. Pathways of environmental transport in PRESTO that are associated with air.

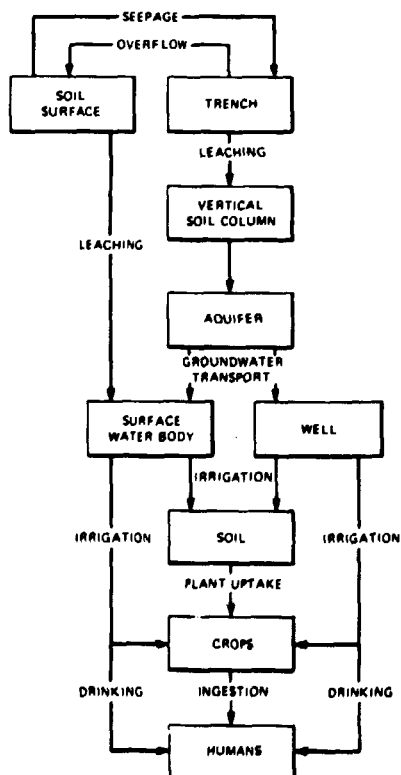


Fig. 2. Pathways of environmental transport in PRESTO that are associated with water.



- + Below the trench, conditions are assumed to be saturated; transport of radionuclides is one-dimensional both vertically from the trench bottom to the aquifer and horizontally through the aquifer to an outcrop or well.
- + Mass of radionuclide inventory is conserved throughout the model.

PRESTO considers only one scenario per computer run. The user may describe the scenario to be simulated by changing input values of such parameters as population size and location, distance to well, percent of cap failure, resuspension rate, etc. By thoughtfully changing the many site description parameters, the user may design numerous scenarios that may be of particular interest. Although the code has been run numerous times for various combinations of parameters for each of the three sites for which data have been compiled, results are not discussed in this paper, but will be published elsewhere. Potential users should bear in mind that the code is relatively new and may yet contain some errors of implementation. In addition, because of uncertainties in the input data values, incomplete knowledge of the site being simulated, or misapplication of the model, imprecise or inaccurate results may be expected in some cases.

#### MODEL DESCRIPTION

The PRESTO code may be operated to calculate doses and risks to an individual or population averaged over the total assessment period, or for some shorter time frame. The model has purposefully been kept relatively simple so that input data requirements are kept relatively small. General assumptions that guided the model development will be discussed later in this section.

PRESTO is modular in construction to allow for more sophisticated submodels or subroutines to be substituted at later dates. The code has been designed with three specific shallow land burial operations in mind: Barnwell, South Carolina; Beatty, Nevada; and West Valley, New York. Example data input decks have been compiled for each of these three locations. However, the code may be suited for use at many other locations if data are available. Some of the submodels included in PRESTO were developed for other types of assessments, and we have adapted them for use in assessing health effects from shallow land burial.

The code user may change the length of simulation to any number of years desired, although the model was designed to estimate releases and subsequent health effects over a period of 1000 years following closure of a burial ground. The code time step is one year and parameters in the model are annual average values. The model also assumes that the radionuclide inventory entered as input data is the

amount of activity found in the trenches at the end of burial operation.

Waste materials in the trench are considered to be effectively homogeneous both in terms of radionuclide distribution and type of material. This homogeneous matter is also considered to be uncontained by canisters, a conservative assumption with regard to radiological dose and potential health effects.

The ingrowth of daughter products of radiological decay is not calculated by the model. In cases where the major dose contribution is from external exposure to a short-lived progeny in equilibrium with a parent radionuclide in the trench inventory, the user should also include the daughter in the inventory.

No more detailed description of chemical reactions than that provided by  $K_d$  is included in the model. The distribution coefficient,  $K_d$  (3), is used to simulate partitioning between water and soil or rock for each element. Values of  $K_d$  are also used to compute the retardation of radionuclides relative to water during transport in the saturated zone. Separate values of  $K_d$  are input for soil, trench material, subtrench soil, and aquifer to allow for changing conditions.

Below the trench, conditions are assumed to be saturated. Transport of radionuclides is one-dimensional both vertically from the trench bottom to the aquifer and then horizontally through the aquifer to an outcrop or well. Each element is transported at a retarded velocity determined by user input values of water velocity, porosity,  $K_d$ , and material density. A correction factor (adapted from reference 4) is calculated and applied to account for the combined effects of radioactive decay and dispersion in the groundwater. The model conserves mass of the input inventory throughout.

Three types of submodels are used in the code: unit response submodels, bookkeeping submodels, and exposure/risk submodels. Unit response submodels calculate the annual response of a given process. For example, submodel INFIL calculates the amount of liquid water available for infiltration through the trench cap each year. The amount of water available annually is placed into the system each year and apportioned by the various bookkeeping submodels. Other unit response models calculate the annual average atmospheric dispersion coefficient and erosion from the trench cap.

Bookkeeping submodels utilize results of unit response submodels plus user control options to account for materials within the model system. For instance, one submodel (TRENCH) maintains a water balance in the trench, calculates the maximum level of standing water in the trench, and estimates the volumes of water leaving annually via the trench bottom or overflow. Other bookkeeping subroutines keep track of leaching from the trench, annual resuspension contribution to the atmospheric source term, etc. The bookkeeping submodels are enclosed in a program loop that iterates once each year during the simulation.

Average concentrations of each radionuclide in important environmental media (e.g., well water) over the assessment period as calculated by the bookkeeping submodels are used by the exposure and risk submodels. Equations from an EPA-funded computer program, AIRDOS-EPA (5) are used to calculate radionuclide concentrations in foodstuffs from atmospheric deposition and irrigation. Foodstuff information and human ingestion and breathing rates are utilized to calculate the annual average radionuclide intake per individual in the population by ingestion and inhalation. These intake data are then converted into estimates of dose rate and health risk by the DARTAB subroutine.

The following assumptions regarding the impacted population are important to note. The population age distribution, spatial distribution, and size is held constant over the simulation period (nominally 1000 y). The health risk methodology assumes, for metabolic parameters, no age or sex distribution; i.e., each member of the population is adult. The total population is assumed to reside within the same 22.5 degree sector. Each member of the population, breathes at the same rate and will inhale the same annual quantity of a given radionuclide. Each member of the population is assumed to eat the same quantities of food, grown on the same fields, and spray irrigated with the same contaminated water that they also drink and feed to livestock. These assumptions while greatly simplifying, seem reasonable considering the uncertainties in predicting individual mobility, population demography, changing agricultural practices, and climatic shifts that might occur during the thousand-year period the model was designed to simulate.

User-input parameters specify the distance and fraction of year that an atmospheric plume blows towards the sector of interest. To allow coordination with more detailed atmospheric dispersion codes, PRESTO has another option. If desired, the user may calculate a value of  $X/Q$  the annual average atmospheric dispersion coefficient, for the site and population in question. Use of such a value will override the atmospheric transport-population assumptions just discussed. In such an instance, the atmospheric source term calculated by subroutine AIRTRM will be distributed to the population as described by the input value of  $X/Q$ .

Several scenarios of human exposure to radiation from the buried waste may be considered by changing various input parameters. These include: normal burial ground function with operational spillage; human intrusion into the trench; site reclamation or farming; an eroded trench cap with subsequent atmospheric contamination via suspension. PRESTO has been kept flexible with regard to structure and definition of exposure scenarios. By thoughtful change of various site description parameters, the user may design numerous scenarios that may be of specific interest.

For the normal scenario, one may wish to calculate collective doses and risks to populations with population centers for each site

characterized by geographic centroid location and number of persons. Operational spillage, defined as the residual radioactivity on the soil surface at the close of burial operations from routine occasional leakage from incoming shipments, may be specified to assess the potential additional risk from such occurrences. Annual average meteorology for the modeled site is input. As noted above, all members of the population utilize common sources of drinking and irrigation water contaminated from trench seepage or overflow. It would be reasonable to calculate individual doses and risks for atmospheric releases under the normal scenario to a person at the site boundary in the prevailing downwind direction. Individual doses due to water-borne effluents may also be calculated at the site boundary.

Under an intrusion scenario, individual calculations may assume that the intruder (a) receives an internal dose from drinking water from a nearby the trench, and (b) receives an external dose from residing in a basement on the trench location; the time of onset and duration of exposure and basement residency are user-specified.

To simulate a farming scenario, an individual may grow and eat his or her own vegetables, beef, and milk produced on land irrigated from the well which may be placed any distance from the trench. The land may also be contaminated by overflow of the waste burial trench onto nearby soil. The farmer also may inhale suspended contaminated soil from the residual operational spillage. Population dose and risk calculations under the farming scenario may assume that the food products grown on-site are ingested by the general population rather than the farmer alone. A limitation to farming scenario calculations in PRESTO is that the code does not include the possibility of roots entering the trench proper. Rather, crops and livestock produced may become contaminated by virtue of overflow from the trench or by irrigation from the contaminated well only.

Given certain inputs, the code may calculate that the trench cap has become completely removed by either erosion or other mechanical means. Thereafter, the absence of a trench cap may provide for suspension and atmospheric transport of trench contents and resulting inhalation by an individual or the specified population. Populations may receive an external dose from immersion in the suspended plume.

The code is structured to consider only one scenario per computer run. The scenario to be simulated may be designed by the user by changing input values of such parameters as population size and location, distance to well, percent cap failure, resuspension rate, etc.

## PRESTO INPUT DATA

The four groups of data needed to execute PRESTO include: (a) site specific data used in transport section of the code for calculating radionuclide concentrations in environmental media, (b) data used specifically by the DARTAB (2) subroutine for creating tabular output, (c) data necessary for subroutine INFIL to estimate infiltration through the trench cap, and (d) dosimetric and health effects data (created by RADRISK, reference 6) also used by the DARTAB subroutine. Preparation of data will be discussed briefly in the following paragraphs.

Site specific environmental and nuclide data have been compiled for each of three sites (Barnwell, Beatty, and West Valley). Some of the data for each site were readily available, and others were estimated from other sites. Easily available data include site location (latitude), precipitation rate, mean temperature, mean barometric pressure, soil porosity, soil bulk density, depth of aquifer, trench dimensions, depth of overburden, location and flow of surface water bodies and wells, topography, average meteorology, location of population centers, etc. Some of the site-descriptive data used to simulate a site such as Barnwell are included in Table 1.

Most uncertain data include: groundwater flow velocity both vertically and horizontally, element-specific retardation as a function of the distribution coefficients ( $K_d$ ) and the activity inventory by nuclide of each site. Estimates of the aquifer flow velocity are available from USGS investigations or other sources such as site operators (7-9). A recent survey of  $K_d$  values for agricultural soils (3) is a prominent source of distribution coefficient data. It is doubtful that site-specific values of  $K_d$  will ever be available for use in the code. The activity inventory of the sites may be the least accessible of any of the necessary data. For the Barnwell facility, incoming radionuclide activity receipts for the life of the burial facility are relatively well known. The West Valley facility radionuclide inventory was described by EPA in 1977 (9). However, the inventory was described without reference to specific radionuclides in most cases. Virtually no data on activity inventory exist for the Beatty site; inventory input for that site were estimated by merging the inventories from other facilities. Assumed values of inventory and  $K_d$  for Barnwell are shown in Table 2.

The necessary data for the food chain, irrigation, human exposure, dose, and risk portions of PRESTO are generally generic in nature. Transfer coefficients for crops, meat, and milk, and human ingestion and inhalation rates are taken from AIRDOS-EPA (5) and from other recent assessment literature (10). Site-specific food chain data, including productivity of pasture grass and produce, irrigation rates, and amount of land irrigated, were considered whenever available.

Table 1. Values of some site descriptive parameters used  
in PRESTO runs to simulate Barnwell site.

| Parameters                                            | Value                |
|-------------------------------------------------------|----------------------|
| Collective at-risk population                         | 7000                 |
| Annual precipitation                                  | 1.18 m               |
| Trench area                                           | 3000 m <sup>2</sup>  |
| Trench depth                                          | 6.7 m                |
| Soil porosity                                         | 0.40                 |
| Soil bulk density                                     | 1.6g/cm <sup>3</sup> |
| Watershed permeability                                | 43 m/y               |
| Trench cap permeability                               | 0.02 m/y             |
| Trench to aquifer depth                               | 4 m                  |
| Trench to well distance                               | 914 m                |
| Groundwater velocity                                  | 2.16 m/y             |
| Wind velocity                                         | 0.4 m/s              |
| Source to population distance, surface                | 100 m                |
| Source to stream distance                             | 914 m                |
| Fraction of initial inventory<br>spilled onto surface | 10 <sup>-8</sup>     |

Table 2. Assumed inventory and  $K_d$  for Barnwell burial ground simulations.

| Radionuclide Activity, Ci Fraction of total |         |        | $K_d$  |
|---------------------------------------------|---------|--------|--------|
| H-3                                         | 34,499  | 3.8E-2 | .01    |
| C-14                                        | 9.5     | 1.1E-5 | .01    |
| Mn-54                                       | 9737    | 1.1E-2 | 150    |
| Fe-55                                       | 65,672  | 7.3E-2 | 55     |
| Co-60                                       | 432,755 | 4.8E-1 | 55     |
| Ni-63                                       | 11,029  | 1.2E-2 | 150    |
| Zn-65                                       | 7,613   | 8.5E-3 | 16     |
| Kr-85                                       | 1,682   | 1.9E-3 | 0      |
| Sr-90                                       | 2,617   | 2.9E-3 | 27     |
| Tc-99                                       | 61      | 6.8E-5 | .033   |
| Cd-109                                      | 16      | 1.8E-5 | 6.7    |
| Sb-125                                      | 2.7     | 3.9E-6 | 45     |
| Cs-134                                      | 37,069  | 4.1E-2 | 1100   |
| Cs-137                                      | 289,403 | 3.2E-1 | 1100   |
| Ce-141                                      | 547     | 6.1E-4 | 1100   |
| Ce-144                                      | 2160    | 2.4E-3 | 1100   |
| Pm-147                                      | 179     | 2.0E-4 | 61     |
| Re-187                                      | 1.5     | 1.7E-6 | 7.5    |
| Pb-210                                      | 22      | 2.4E-5 | 99     |
| Ra-226                                      | 0.6     | 6.7E-7 | 220    |
| Th-232                                      | 1.3     | 1.4E-6 | 60,000 |
| U-234                                       | 3.0     | 3.3E-6 | 45     |
| U-235                                       | 3.5     | 3.9E-6 | 45     |
| U-236                                       | 0.5     | 5.6E-7 | 45     |
| U-238                                       | 3063    | 3.4E-3 | 45     |
| Pu-238                                      | 0.2     | 2.2E-7 | 1800   |
| Pu-239                                      | 0.1     | 1.1E-7 | 1800   |

The input to subroutine DARTAB (2) provides the user with options for processing the exposure data, dosimetric data, and of tabulations of output. The data are read using the special form of the IBM FORTRAN NAMELIST format. The order of data items in a NAMELIST group is not important, but names, such as those for radionuclides, cancers and organs must agree with those expected by the RADRISK data file discussed later. The data described in this section must follow directly after the radionuclide-specific environmental transport data discussed above. Data entered in the namelist section include information necessary to control the types of tables output by DARTAB. Summary tables or very detailed tables that describe dose rates or cancer risks may be printed. Tables may be structured to indicate impacts to either individuals or the population. Dose rates and risks may be categorized by affected organ, pathway of exposure, and type of cancer expected. Up to twenty organs and/or twenty types of cancer may be explicitly designated. As with the rest of PRESTO, up to forty radionuclides may be considered.

Inputs to subroutine INFIL (which was adapted from reference 11) are of four types: hydrologic and trench description characteristics; maximum day length in hours by month; daily mean temperatures by month; and hourly precipitation for non-zero precipitation events. Descriptive parameters include trench width and slope, permeability of the trench cap, porosities for the gravity water and pellicular water zones, equivalent upward diffusivity and hydraulic conductivity, and infiltration capacities for the pellicular and gravity water zones. Also entered are the monthly mean maximum length of day for the site's latitude. Daily mean temperatures by month are also required for INFIL operation. The final block of data read by INFIL consists of hourly precipitation data for days when rain occurred.

Dosimetric and health effects data are read in an unformatted form from a magnetic tape which contains information about many radionuclides. The data on the tape were created by executing the RADRISK code (6) and should be sufficient to meet most users' needs. For a further description see the DARTAB user's guide (2) and the RADRISK documentation.

## CONCLUSIONS

A computer code has been developed for use by the USEPA in support of development of a generally applicable environmental standard for low-level waste disposal. The code, named PRESTO, is designed to simulate environmental transport and ensuing risks to populations or individuals following closure of a shallow land disposal trench. The code is flexible, easy to parameterize, relatively quick-running, and easily modified. This paper briefly describes the code structure and some of the assumptions guiding its development. Simulations have



been performed for several release scenarios at three specific sites having different waste inventories, geophysical characteristics, and population distributions.

We hope to evaluate the uncertainties associated with predictions of the PRESTO mode as functions of the precision with which input variables are known. Determination of the sensitivity of model results to variations in model input values would indicate where efforts to gather data would be most valuable in determining where research dollars should be spent.

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## GROUNDWATER TRANSPORT MODEL DOCUMENTATION AND EVALUATION GUIDELINES

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### ABSTRACT

Documentation and evaluation guidelines are being developed to test and verify computer simulation models of contaminant transport in the saturated and unsaturated ground-water zones. The guidelines will assist managers with selecting predictive models appropriate for specific low-level radioactive waste burial sites. A selection process will be facilitated by ensuring that transport models evaluated with the guidelines will adequately describe the relevant physical phenomena for an arid or humid site.

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### INTRODUCTION

It is perhaps self evident that computer simulation models will be needed in order to organize and utilize shallow-land burial site characterization information for purposes of making decisions about site operational safety. Models are required in order to focus the multitude of complex technical considerations involved in a decision making process for low-level waste management. As scientific tools, models should be reliable and credible representations of burial sites and their physical behavior. In as much as ground water has been identified as a major environmental pathway for possible escape of radionuclides (1), documentation and evaluation guidelines are being developed exclusively for ground-water transport models in this project. The purpose of the guidelines is to assist with development and selection of appropriate and credible ground-water transport models for application to shallow-land burial site analysis. This task is in support of Milestone C of the LLWMP: Develop Technology and Documentation Required to Open a Shallow-Land Burial Site.

To manage low-level wastes, systems models will be needed to perform initial screening of possible sites, to assist in the design of burial schemes and trenches, to assess the containment performance of specific sites, and to aid in the design of site monitoring programs and data analysis. This diverse range of model applications will call for many different kinds of systems simulation models of

similarly diverse theoretical complexity. The variety of systems models that may be necessary makes development of a single set of guidelines for selection purposes a difficult problem. In particular, transport models that neglect certain physical phenomena of no relevance to a specific site should not be rejected for possible application. On the other hand, highly detailed models, including nearly all known physical mechanisms, should not necessarily be selected when site characterization is insufficient to operate the models.

Nevertheless, the guidelines will attempt to provide criteria for identifying and selecting appropriate and credible transport models. The guidelines are being formulated in two main chapters. Documentation guidelines (I) will remind model developers of their scientific obligations for providing a complete and adequate report that makes a model understandable and easily transferred to a user. The evaluation guidelines (II) will treat certain ground-water transport modeling technical issues, by providing suggested reference case solutions and experimental data sets representing fundamental physical processes. Details of the proposed guideline contents, which are only briefly reviewed here, were recently submitted as a letter report.

#### DOCUMENTATION GUIDELINES

These guidelines will establish documentation standards and format for the adequate description of a model's capabilities, limitations, computer implementation, and applicability. This part will indicate the necessary content of a useful and adequate computer model document. The documentation guidelines are being selected and adapted from those proposed by the U. S. NRC (2). Guideline contents are shown in Table 1 with the meaning of each item given in parentheses. By addressing completely the ten aspects in Table 1, a model developer can assure creation of an understandable and useful model. The final guidelines will recommend detailed guidance for each aspect. Essentially, a well-documented model reports all assumptions and the methods employed to solve a particular problem and provides a working computer code along with consistent example simulations. Fulfillment of all ten aspects, however, is usually quite difficult to achieve, although the list is deceptively simple.

#### EVALUATION GUIDELINES

The purpose of these guidelines is to help select adequate models for a particular site application and to assure reasonable simulation accuracy.

Table 2 contains the proposed contents. The major part of the guidelines will reside in the evaluation test cases, Section 4. The detailed content of the evaluation test cases is shown in Table 3. These test case are being arranged according to fundamental physical

TABLE 1. A List of Contents for an Adequate  
Computer Model Documentation

I. Documentation Guidelines

- 1) Purpose of model (reason for development and physical phenomena simulated)
- 2) Fundamental physical processes described (basic mathematical equations solved and assumed conditions)
- 3) Mathematical methods and computer algorithms employed (solution of basic equations)
- 4) Site specific data required to control model (simulation control parameters)
- 5) Calibration of model parameters (adjustment of model to input data).
- 6) Verification and validation evaluations performed (application of evaluation test cases)
- 7) Example simulation results (a site application of the model and demonstration of test cases)
- 8) Sensitivity study and prediction uncertainty (accuracy and credibility of model)
- 9) User's manual and commented computer codes (numerical implementation of model)
- 10) Methods for display and interpretation of computer output (simulation results and analysis)

**TABLE 2. Contents of Evaluation Guidelines****II. Evaluation Guidelines**

- 1) Philosophy of Systems Modeling
- 2) Review of Transport Physical Theory
  - A. Saturated flow
  - B. Partially saturated flow
  - C. Transport of solutes
- 3) Solution of Equations: Numerical Methods
- 4) Evaluations: Verification and Validation Test Cases
- 5) Calibration and Sensitivity: Prediction Uncertainty

**TABLE 3. Evaluations: Verification and Validation Test Cases**

- A) Water Flow
 

Aquifers, infiltration, drainage, redistribution, evaporation, runoff, soil layer effects, hysteresis, and multiphase air and water vapor flow
- B) Evapotranspiration
 

Non-isothermal moisture flow, plant root water extraction, and climatic influences
- C) Coupled Solute and Water Transport
 

Convective-dispersive transport, radioactive chain decay, and chemical-media interactions

processes likely to be relevant for simulation of a shallow burial site in either humid or arid environments. Each case will represent at least one accurate numerical example of a particular fundamental process. No attempt will be made, however, to review or demonstrate every numerical procedure applicable for solving a fundamental process, since this would produce an excessively large list of cases. Such test cases are not meant either to represent an absolutely exhaustive list. In view of the multitude of kinds of models that may apply to a shallow burial site, a model developer may find it necessary to introduce a different test case with a similar intent of verification.

Effects of discretization, time step size, boundary conditions, representation dimensionality, and parameter calibration will be discussed for each physical process in Table 3.

An ideal test case would consist of an analytical solution to a fundamental equation and the associated experimental data verifying the accuracy. However, that ideal may not always be attainable. Other applicable numerical methods will then be used instead to obtain an example solution.

Briefly, the content of each section will be the following:

PHILOSOPHY OF SYSTEMS MODELING will discuss how models should be constructed based on theoretical concepts and available data. Use of models as scientific tools for organizing information and predicting systems behavior will also be discussed.

REVIEW OF TRANSPORT PHYSICAL THEORY will summarize the mathematical equations which describe water flow and solute transport in porous media. A discussion of initial conditions, boundary conditions, and sources and sinks will also be included.

SOLUTION OF EQUATIONS will briefly review mathematical methods used to obtain numerical solutions to the fundamental equations.

EVALUATIONS: VERIFICATION AND VALIDATION TEST CASES will provide a collection of analytical, numerical, and experimental data sets which exemplify proper modeling of the fundamental physical process likely to be important to simulating a shallow burial site.

CALIBRATION AND SENSITIVITY: PREDICTION UNCERTAINTY will discuss sensitivity of modeling predictions based on model design and parameter estimation methods.

The evaluation test cases can be used to identify an appropriate model by simply selecting one that has compared favorably for the physical processes relevant to a particular problem or shallow burial



system. This would constitute proof that a particular model simulates certain physical processes adequately. Secondly, a selected model would be demonstrated for the same relevant processes but with parameters consistent with a particular set of site specific data. This two-stage test approach would thus verify proper model function relative to established cases and then demonstrate similar model behavior relative to site specific data.

### AN EXAMPLE TEST CASE

The evaluation test cases will constitute the main technical contribution of this project. The strategy of test case selection will be to identify well-established examples in the ground-water literature, which can be re-simulated to determine reproducibility and the effect of certain physical assumptions. Thus, as much as possible, test cases will be doubly verified. For instance, the Hillel (3) examples of one-dimensional, vertical unsaturated moisture flow in different soil types were selected as ideal reference cases. The examples were originally simulated using explicit finite difference methods implemented with the CSMP computer language. Those examples were re-simulated using a numerically more stable implicit finite difference scheme coded in the Fortran language. Results were similar to Hillel's original simulations, but still display certain technical discrepancies. Each case, representing the physical processes of drainage, evaporation, and infiltration, are typical of unsaturated flow behavior and can be used to verify the accuracy of simulations performed with any other similar model.

As test cases, the Hillel examples, however, are not exactly ideal, since each basic physical process does not have an exact analytical solution. Indeed, analytical solutions for vertical, transient unsaturated moisture flow have never been found for realistic soil hydraulic properties. Therefore, reference cases for purposes of solution verification must be based on numerical approximation. Validation of the basic physical processes is implicit in an acceptance of Darcy's Law and Richard's Equation as fundamental and proven theory. Of course, in an actual field situation, various other phenomena can interfere to cause deviation from that theory. And that is a principal reason for requiring a further comparison with site specific data. That is, validation of basic theory is not required for each model application; but the adequacy of the representation for a specific site may need comparison and evaluation with actual data, which is the purpose of field-experimental test cases.

Graphs of the hypothetical, but typical, hydraulic properties are shown in Figures 1 and 2. The curves are described by polynomial expressions fitted to Hillel's original data, indicated by circle points. The polynomials are universally transferable to any other unsaturated flow simulation model and constitute a method for providing hydraulic data which is free of interpolation and interpretation error.

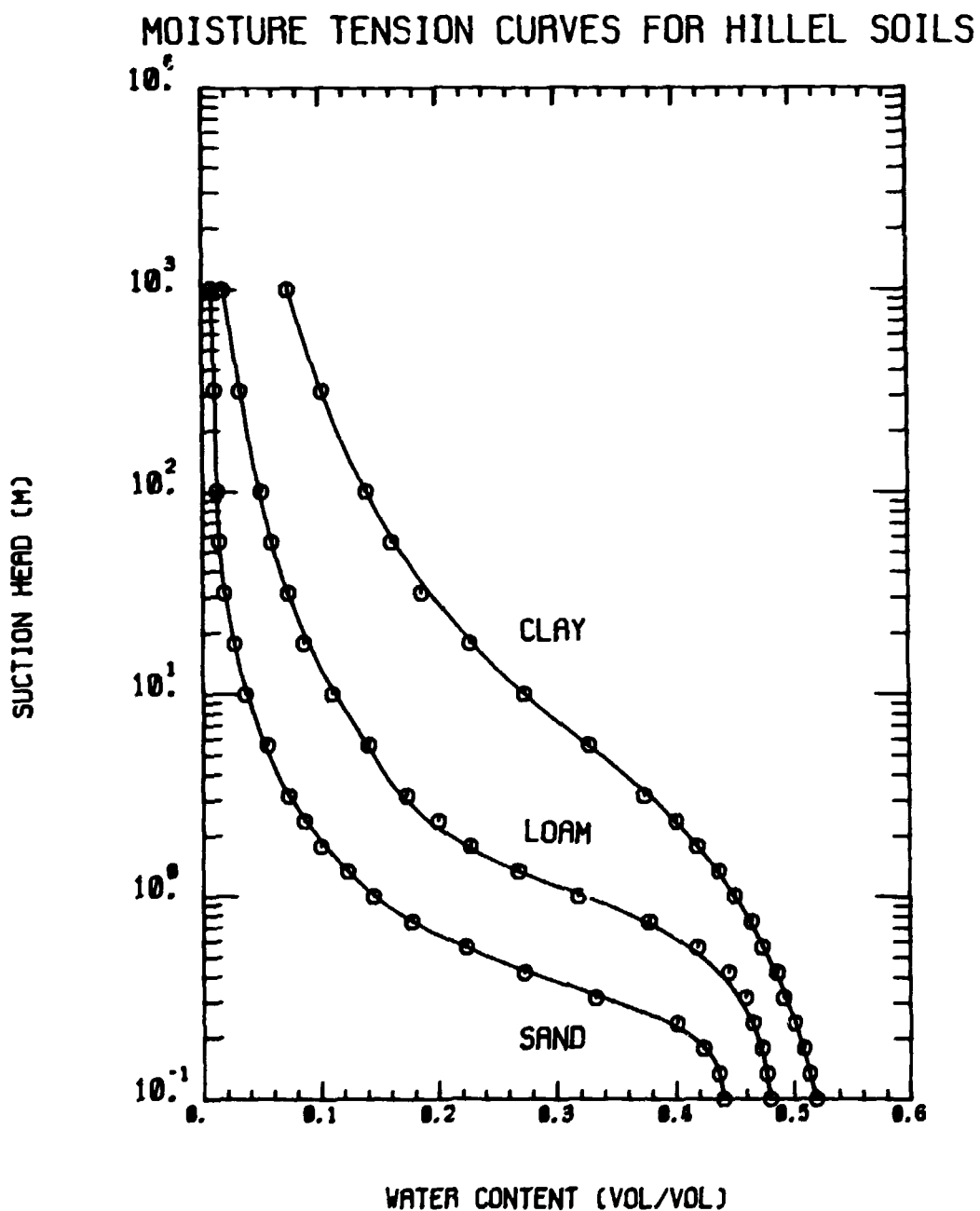


**FIGURE 1.** Soil Moisture Characteristic Curves for Sand, Loam, and Clay. Circles are Hillel's (4) original data, and the curves are polynomial representations.

## HYDRAULIC CONDUCTIVITY CURVES FOR HILLEL SOILS

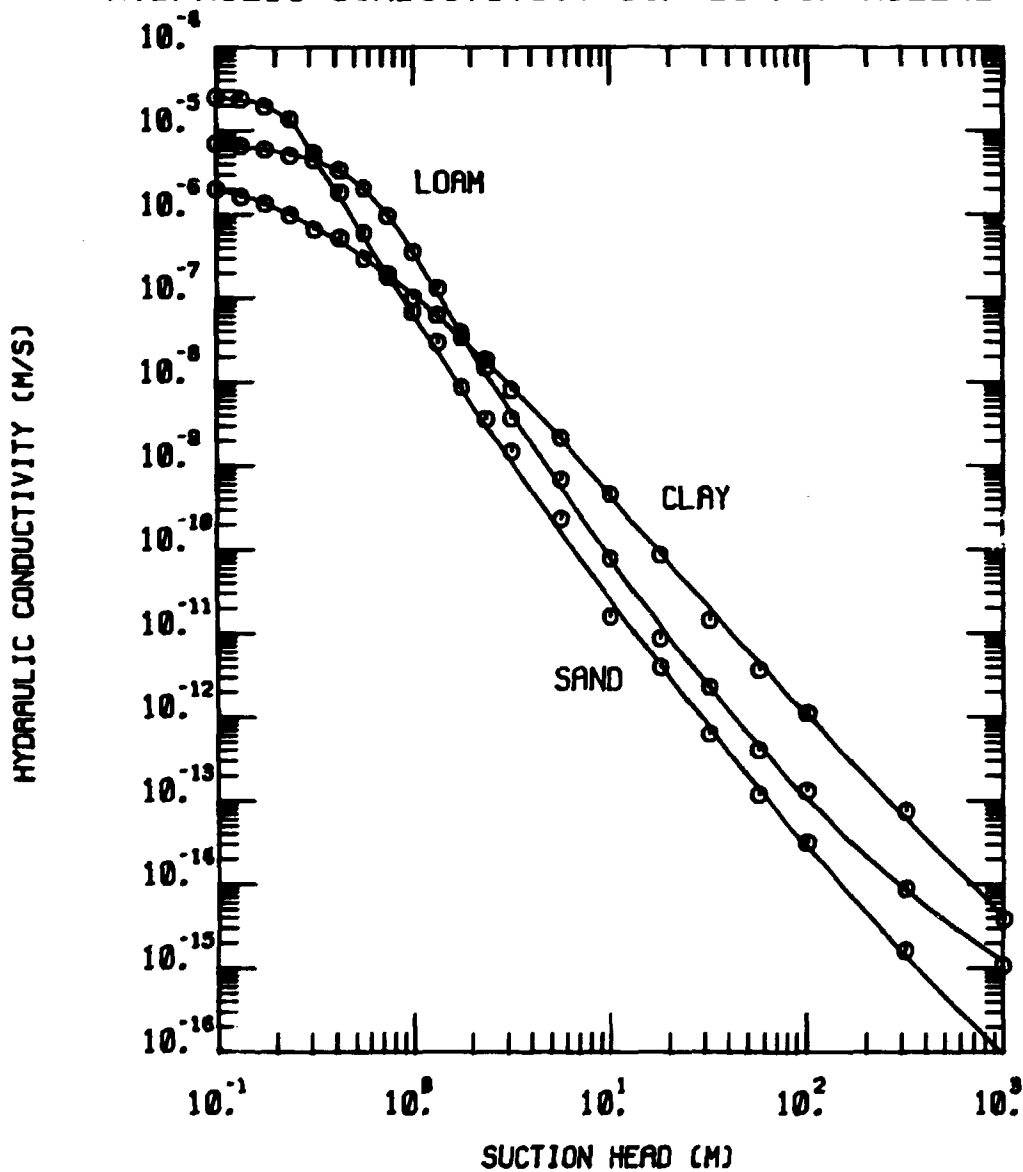


**FIGURE 2.** Hydraulic Conductivity As a Function of Suction Head for Hillel's Three Soil Types. Circles are Hillel's (4) original data derived from the Millington-Quirk method using the soil moisture characteristic curves. The curves are polynomial representations used in the simulations.

The Hillel (3) example for a 72 mm rainfall lasting six hours on uniform soil profiles of clay, loam, and sand followed by a cyclic pattern of diurnal evaporation is shown in Figure 3. The water content profiles for various times during and following the rainfall event are the re-simulations of the originals. Simultaneous process of infiltration, redistribution, drainage, and evaporation are demonstrated by the profiles. Differing amounts of water flux passing through such soils could be available for dissolving and transporting shallow buried and exposed radioactive wastes. An actual shallow-land burial site, however, would not necessarily have the same soil types or profile depth, but similar moisture flow processes would be relevant. As a general rule, any unsaturated flow model to be employed for assessing possible contaminant migration should be capable of accurately simulating such a level of physical detail. On the other hand, the average downward flow produced by many such detailed events might suffice to describe the long-term leaching of a very slowly escaping contaminant. Thus, in that situation, a transport model of a more simple design might be just as valid for problems of greater time scale. That is, verification of a particular physical model does not guarantee a uniquely valid description of transport processes for all applications.

## PROJECT PLANS

The approach taken in this project has been to examine various kinds of available ground-water transport models and to assess their approach relative to current technical understanding. No attempt, however, is being made to criticize or identify inadequacies of specific models. The composite of many diverse ground-water models seems to indicate naturally what constitutes a well-documented and adequately tested model. It is our belief that by satisfying every aspect listed in Table 1, existing models can achieve considerably improved credibility and usefulness.

Evaluation test cases will be based as much as possible on simplified analytical solutions of fundamental equations and on actual experimental data. However, when analytical solutions do not exist, approximate numerical schemes will be applied. A single collection of evaluation test cases applicable to all the various kinds of transport models is probably nearly impossible to devise without allowing the possible rejection of certain useful models. However, the evaluation test cases will attempt to identify most modeling technical issues which, if not properly addressed, can lead to incorrect results. Model developers and users, however, can introduce other test cases when necessary to satisfy Table 1.

A letter report on the selected analytical solutions will be submitted at the end of FY82 and final guidelines are to be completed in FY83. Applicable experimental data sets for validation purposes will be identified in FY83. A peer review of the guidelines is also planned.

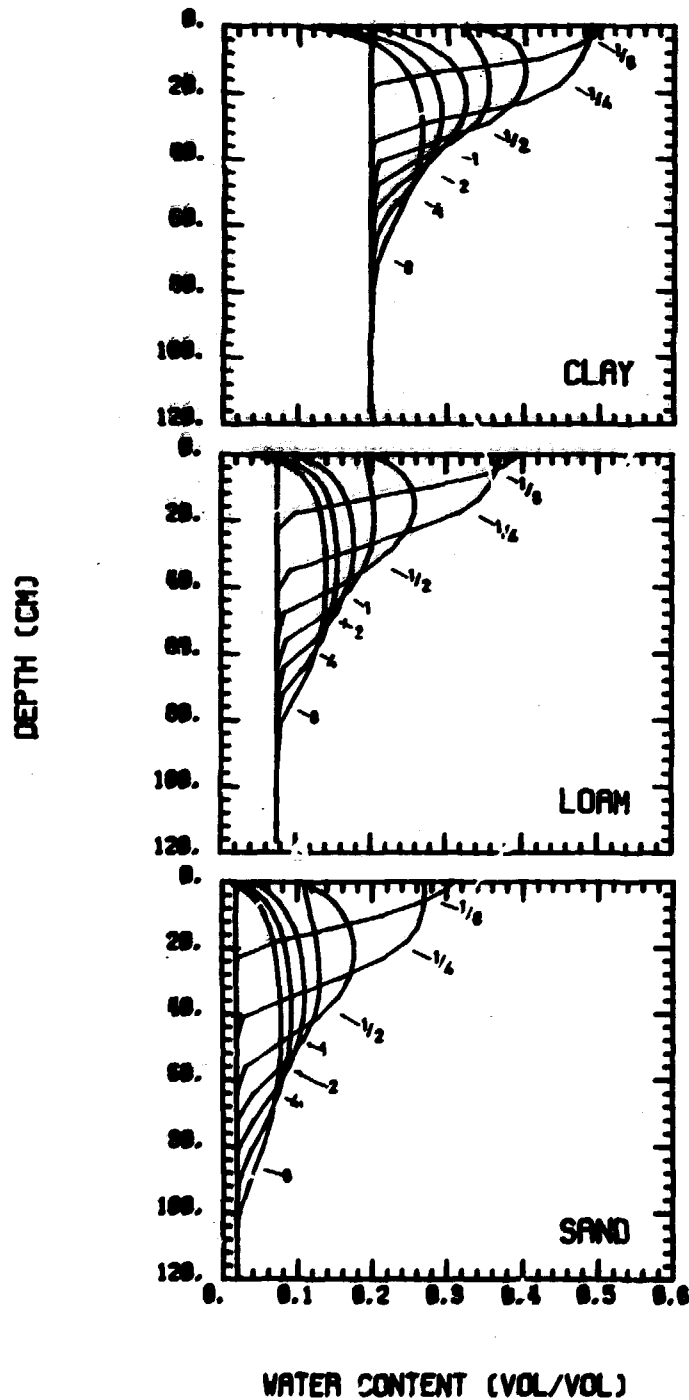


**FIGURE 3.** Soil Moisture Distributions During Infiltration and Following Diurnal Cyclic Evaporation in Uniform Profiles of Sand, Loam, and Clay. The solid lines are for a single rainstorm of 72 mm lasting 6 hr, and numbers indicate days following the onset of the rain. Simulations are based on an implicit finite difference solution of Richard's equation for unsaturated moisture movement in soils characterized by Figures 1 and 2.

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**SHALLOW LAND BURIAL HANDBOOK**

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**INTRODUCTION**

Additional disposal facilities are needed to accommodate the low-level radioactive wastes being produced by both commercial and defense activities. DOE's Low-Level Waste Management Program is to "Develop the technology and documentation required to open a shallow land burial site." The Shallow Land Burial Handbook supports this goal by transmitting state-of-the-art information about shallow land burial technology to the user community. The handbook is structured to provide an overview and initial guidance to developers and state/local planners concerning development phases and activities required for establishing new facilities for near-surface disposal of low level radioactive waste. Activities include defining and scoping the problem, establishing objectives, compiling information, considering alternatives, and establishing and implementing plans. These activities are implemented during the preliminary analysis, site selection, facility design and construction, facility operation, and facility closure/post-closure development phases.

**PRELIMINARY ANALYSIS**

The preliminary analysis phase of developing a new disposal site provides the data and assurance needed to proceed. Preliminary analyses define and scope the problem, evaluate regulatory and licensing requirements, establish basic objectives and goals required for facility development, and include preliminary design and economic studies. The objectives and goals addressed in the Shallow Land Burial Handbook are derived from basic scientific principles and engineering practices, DOE guidelines, NRC regulations and position papers, and regulatory requirements imposed by other federal and state agencies. The developer may choose to add other constraints such as waste acceptance criteria or desired facility capacity.

The compilation of information provides a basis for preliminary marketing, feasibility, and economic studies as well as for understanding site development strategies, alternatives, and interfaces. Sufficient information is required to prepare a logic diagram highlighting major requirements and decision points. All phases of development are interdependent.

## SITE SELECTION

Successful site selection can simplify licensing, reduce site development costs, and simplify post-closure surveillance and maintenance. Site characteristics important to shallow land burial include predictability, stability, hydrogeology favorable for avoiding water, no exploitable resources, and public acceptance. The primary goal of site selection is to identify a site that can be suitably characterized, modelled, analyzed and monitored to show compliance with performance objectives. Site selection is completed in two stages - site screening and site characterization. Site screening analyses utilize reconnaissance level data and start with a region of interest to which area screening criteria are applied to identify a few candidate areas. Site screening criteria are then applied to the candidate areas to identify a slate of candidate sites. After data collection and evaluation, a preferred site can be identified from the slate of candidate sites. The logic is outlined in Figure 1.

Site characterization follows site screening and concentrates on evaluating the preferred site to confirm that the selected site meets performance objectives and collecting baseline site data. Detailed field studies and laboratory analyses are initiated and a preliminary pathways analysis is completed. If the preliminary pathways analysis complies with performance requirements a more detailed analysis is completed and site development proceeds. Site flaws identified in the preliminary pathways analysis will be addressed by adding design features, operating constraints, or waste form constraints to be considered in conjunction with site characteristics. The detailed pathways analysis will then be completed to either demonstrate compliance and proceed with site development or indicate that a new site must be considered. The site characterization logic is summarized in Figure 2. Input from parallel engineering studies is required during site characterization.

The handbook identifies the type of information required, sources of information, and limitations on its availability and use. Parameters important to both site screening and characterization are organized in the handbook according to the disciplines of geology, hydrology, meteorology, ecology, and socioeconomics.

## FACILITY DESIGN AND CONSTRUCTION

Design requirements and preliminary design studies are important inputs to the previously described steps of preliminary analysis and site selection. Similarly, development of design strategies and alternatives are dependent on site operations and site closure/post-closure requirements.

The disposal unit is the portion of the facility where waste is placed for disposal. The important design objectives for disposal units are minimizing water contact with the waste, avoiding subsidence and



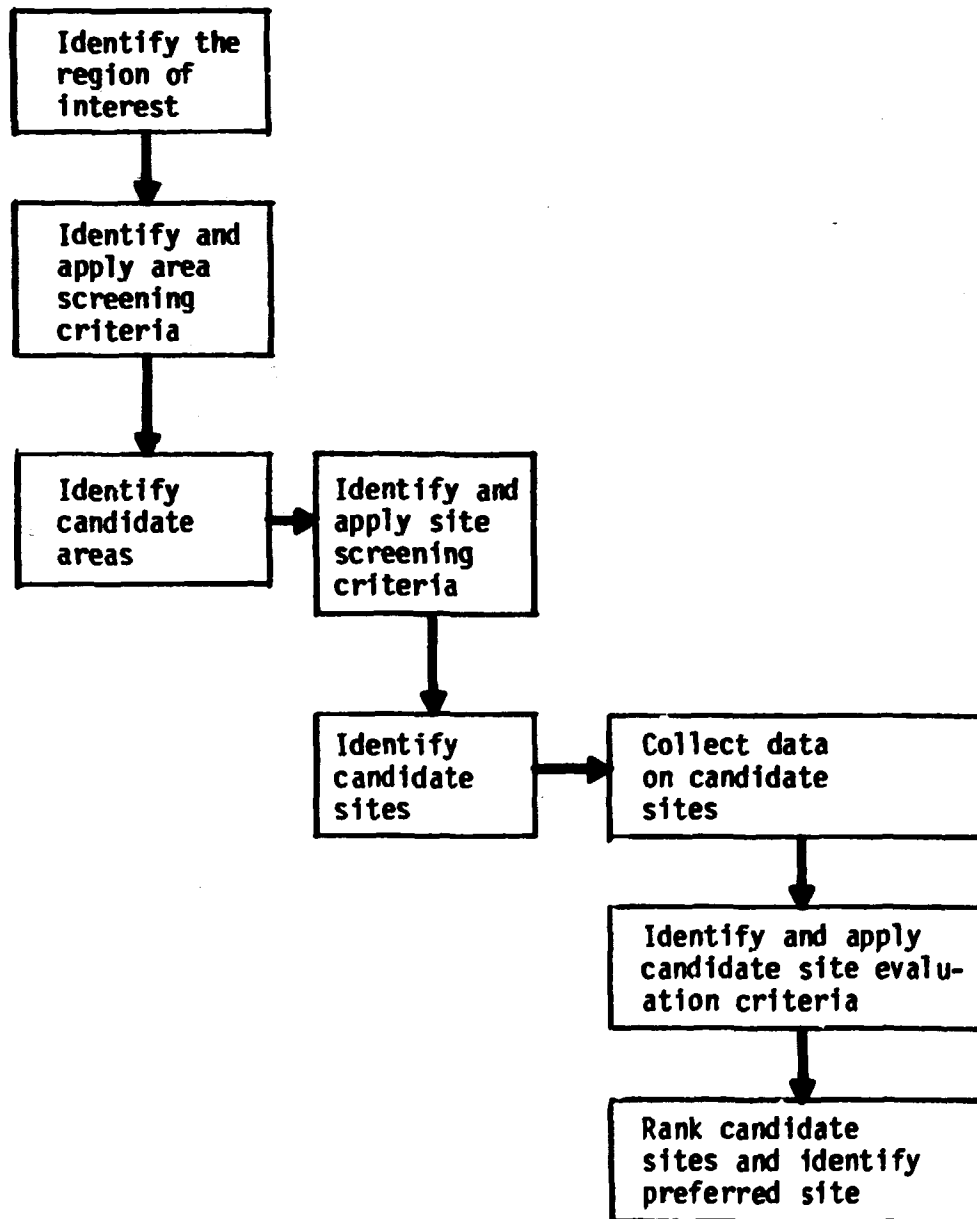


Figure 1. Site screening logic

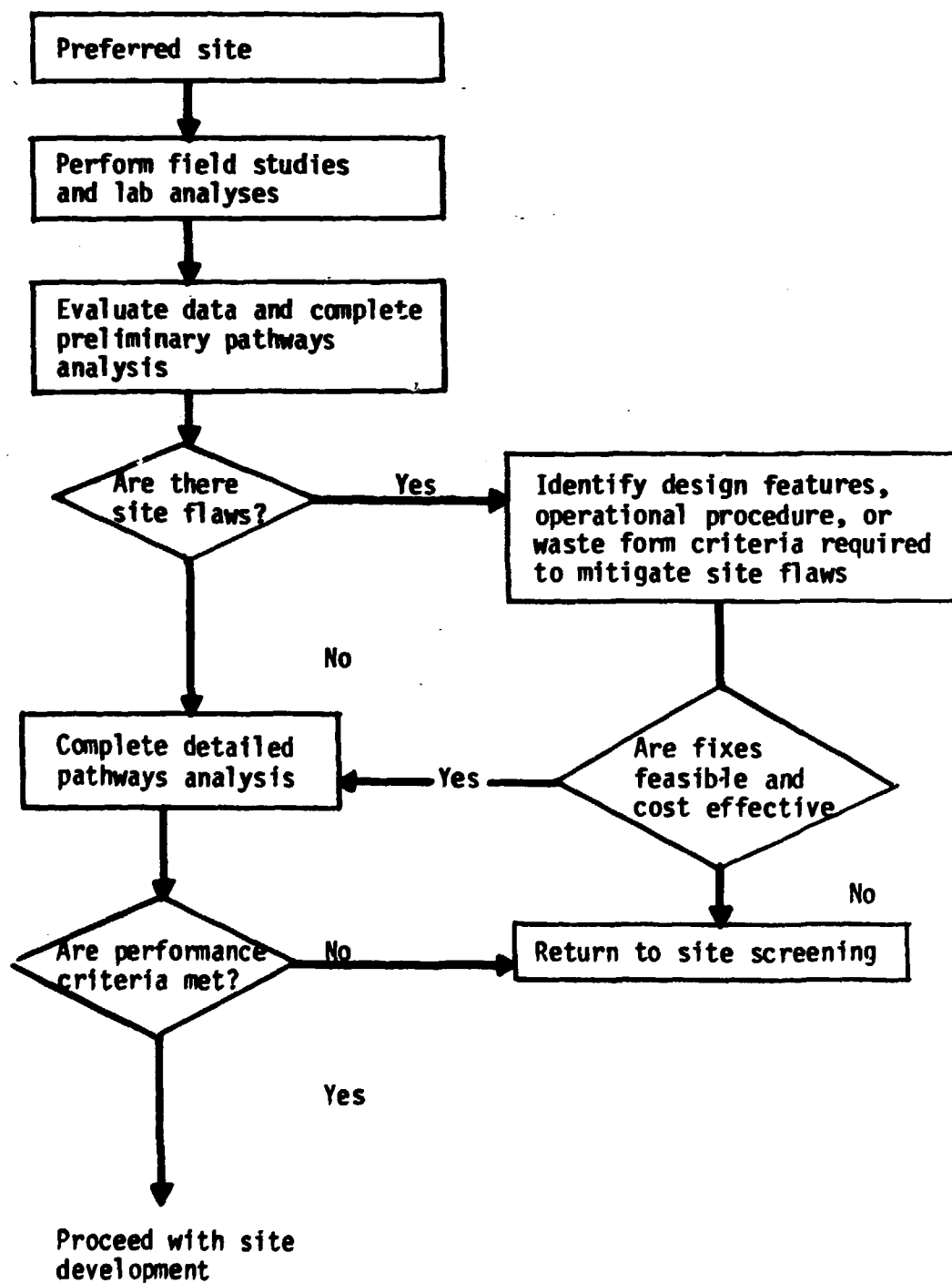


Figure 2. Site characterization logic

optimizing land use. Site and operations considerations influence these objectives. Typically, a variation of the trench concept is used in near-surface disposal. Options include vertical walled trenches, sloped trenches, broad trenches, or slit trenches. The site may use a combination of trenches depending on waste characteristics, soil stability, depth to groundwater, and available disposal area. Applications, advantages, and disadvantages are summarized in the handbook. A number of design features are considered for minimizing water contact, reducing subsidence, or optimizing land utilization; including sizing, drainage, liners, installing leachate detection systems, and interim closing with impermeable caps. Site development, assessments and permits (i.e., environmental, safety and quality assurance), and actual construction are also addressed.

### FACILITY OPERATIONS

The primary objective of site operations is to avoid water contact with the waste and minimize subsidence. Facility operations and procedures applicable to low level waste disposal include waste handling, waste placement in trenches, interim closure and stabilization, personnel monitoring, and recordkeeping. These procedures must consider the nature of the waste handled and the regulatory requirements. Waste handling encompasses transport, receiving and storage. Waste placement techniques considered in the handbook are the ones currently used: remote mechanical placement using cranes at the trench edge, forklift placement from the edge of the trench or from the bottom of the trench, direct dumping into the trench. Reducing active areas and stabilizing by backfill and compaction are considered to be important actions in interim closure and stabilization. Personnel monitoring and recordkeeping requirements are addressed.

Other operational considerations such as environmental monitoring, operating equipment, security, staffing, contingency planning, and required assessment and audits are addressed in the handbook.

### FACILITY CLOSURE/POST-CLOSURE

Strategies for site closure and post-closure affect site selection, design, and operation. Protection of the public, protection of the individual and site stability are major goals. A final site closure plan which must be prepared prior to the end of facility operations, includes decontamination, dismantling, removing, and disposing of all facilities not necessary for post-closure or institutional controls and may require recontouring or revegetation of the entire site. Post-closure monitoring and maintenance and corrective actions are addressed. The institutional control period follows post-closure, with active control measures as necessary.

An important element in developing a commercial low-level waste disposal site is the requirement by NRC for financial assurance arrangements that finance final site closure, post-closure, and institutional care. Financial assurance must be fully funded prior to startup of site operations.

#### CONCLUSIONS

The facility development phases (preliminary analysis, site selection, facility design and construction, facility operation, and facility closure/post-closure) are systematically integrated into a logical plan for developing near surface disposal plans. The Shallow Land Burial Handbook provides initial guidance and concepts for understanding the magnitude and the complexity of developing new low-level radioactive waste disposal facilities.

**GEOTECHNICAL QUALITY CONTROL: LOW-LEVEL RADIOACTIVE  
AND URANIUM MILL TAILINGS DISPOSAL FACILITIES**

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**ABSTRACT**

Current general regulations for low-level radioactive waste (LLW) do not contain any nationally accepted geotechnical standards or criteria that apply to the disposal of this type of waste. The Nuclear Regulatory Commission (NRC) through its rulemaking authority is charged with the responsibility of developing necessary standards and technical criteria for LLW disposal.

Through an interagency agreement with the NRC, the US Army Corps of Engineers, Waterways Experiment Station will provide to the NRC technical assistance and guidance to develop a geotechnical quality control program for use during construction, operation, and closure of LLW and uranium mill tailings disposal facilities. This paper presents the work to be conducted in accomplishing this task and work that has been accomplished to date.

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**INTRODUCTION**

Current general regulations for low-level radioactive waste (LLW) do not contain any nationally accepted geotechnical standards or criteria that apply to the disposal of this type of waste. However, the need for such comprehensive, national standards and technical criteria is well documented. The Nuclear Regulatory Commission (NRC) through its rulemaking authority is charged with the responsibility of developing necessary standards and technical criteria for LLW disposal.

On July 24, 1981, the proposed rules "Licensing Requirements for Land Disposal of Radioactive Waste" (10 CFR 61) were published in the Federal Register, Vol 46, No. 142. The purpose is to establish technical criteria and procedures for the land disposal of radioactive waste. "Subpart B-Licenses" of 10 CFR 61 defines the specific technical information to be included in a license application.

The specific technical information for the license application must include (10 CFR 61.12) a description of (a) the natural site characteristics, (b) the facility design features, (c) the principal design criteria, (d) the design basis natural events, (e) the codes and standards used, (f) the methods of construction and operation, (g) the site closure plan, (h) the site natural resources, (i) the radioactive material to be disposed of, (j) the quality assurance program for site characterization and for design, construction, and operation of the facility, (k) the radiation safety program, (l) the environmental monitoring program, and (m) the administrative procedures to be used. A number of reports are in various stages of preparation by the NRC on these subjects. The documents are intended to provide technical background of use to the developers and operators of LLW disposal facilities and to the agencies responsible for licensing in the preparation, review, and issuance of licenses for low-level radioactive waste disposal facilities.

#### WORK TO BE ACCOMPLISHED

Through an interagency agreement with the NRC the U. S. Army Corps of Engineers, Waterways Experiment Station will provide to the NRC technical assistance and guidance to develop geotechnical quality control in support of the requirement of item (j) above.

The work to be accomplished for the NRC will be developed and published in three reports. The objective of the first report will be to identify those site geotechnical parameters which should be tested, observed, and documented during construction, operation, and closure of a site. For LLW disposal sites, these aspects include site grading, locations of trenches or other structures, materials encountered on the sides and floors of the trenches, and types and engineering properties of fill, including the effects of compaction. For uranium recovery waste retention systems, these aspects include site grading, foundation preparation, liners, leak detection systems, embankment construction, and erosion protection. Groundwater testing or monitoring is not included except in areas where it will directly affect other parameters.

The second report will identify recommended laboratory and field testing and observation techniques needed for measurement of the parameters established in the first report. These techniques will be referenced to published standards or to appropriate technical publications to the extent practicable. Where such references are not available, state-of-the-art procedure will be described for that particular technique. In the event that no state-of-the-art or standard procedure exists, identification of needed research will be made to determine suitability of the parameter in question. Included in this report will be an explanation as to the reasons certain tests and field methods, equipment, etc., were selected over other available options.

The third or final report will identify a recommended quality control program and documentation to be used during site construction, operation and closure of low-level waste disposal facilities and during construction of waste retention systems such as uranium recovery facilities. Specifications for the type and frequency of testing will be included along with specifications for appropriate observations and documentations.

#### WORK ACCOMPLISHED TO DATE

The first report under work to be accomplished is in final draft form. It is entitled "Geotechnical Parameters for Use in Construction Quality Control of Low-Level Radioactive Waste Disposal Sites". The scope of the report reads:

"During the construction, operation, and closure of a LLW disposal site, and during construction of a uranium recovery waste retention system, the licensee is required to test and/or observe and to document:

- a. The verification of geologic media site characterization and soil engineering properties design assumptions following exposure of trench or impoundment walls and bottoms.
- b. The quality control of the construction group's geotechnical materials and workmanship.
- c. Acceptance testing and inspection for compliance with specification requirements.

This report identifies the various soils tests and observations that should be made for a, b, and c above. A division between design verification and construction quality control is made. The criteria for selection of a test/observation are given.

The final section of this report lists the various processes involved in the construction, operation, and closure of a LLW disposal site and presents those geotechnical tests and/or observations that should be made during each process. Emphasis throughout is on parameters that can be documented, either as a number or a simple, easily understood descriptive term.

This report is limited to consideration only of geotechnical observations and tests, i.e., involving physical properties of soils. Geochemical tests are not included. It is not common in the earthwork construction industry to perform geochemical tests as part of the quality control of construction. However, if such tests are necessary during construction, they should be performed on an as-needed basis, e.g., for the verification of an assumption made during the design stage."

Specifically, the report is outlined as shown by Table 1, the content page. Part I discusses the purpose, previous studies and scope. Part II, Tests and Observations, presents a rationalization for the geotechnical observations and tests that should be performed and documented for quality assurance/quality control (QA/QC) during construction operations. The unique relationship between the licensee and the regulatory agency (licensor) and the effect it has on the quality assurance program is given. Design parameter tests are presented and their unsuitability for quality control of construction processes is discussed. The criteria for a quality control type of observation or test are given and a group of soil index properties tests commonly used in quality control is presented.

Part III, Geotechnical Parameter Identification, presents those geotechnical observations and/or tests that should be made and documented during the construction, operation, and closure of a LLW disposal site or the construction of a uranium recovery waste impoundment. The presentation is done in the context of the various unit processes involved. The observations/tests and documentation will be the same for the same unit process, whether for a LLW disposal site or a uranium tailings impoundment.

Only those unit processes, or portions of processes, whose quality of materials and workmanship directly affect the ability of the site to meet mandated performance standards are presented. The emphasis in the listing given for each process is on a quality assurance/quality control program that focuses on catching mistakes before or as they occur, correcting them, testing for specification compliance, and documenting the entire process to provide the necessary confidence in the QA procedure. During the conduct of each unit process, there must be continuous, cooperative interaction between the groups within the licensee's forces responsible for the Design function, the QA/QC function, and the Construction and Operations function to secure the required verification of design assumptions and quality of work, along with the effective, efficient, and profitable use of men, machinery, and materials.

In Appendix A, media verification studies for confirming the geologic site characterization made during the planning stage are discussed.



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## TABLE 1

Table of contents, "Geotechnical Parameters For Use In Construction Quality Control of Low-Level Radioactive Waste Disposal Sites".

## SHALLOW LAND BURIAL CRITERIA FOR SODIUM-CONTAMINATED WASTE

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The background material for this program was presented in the November 1981 meeting in New Orleans. To ensure continuity for the audience, a summary of the previous material will be discussed before the FY-82 progress is presented.

Radioactive sodium-bearing waste from LMFBR programs is being stored since present shallow land burial criteria prohibit elemental sodium. The present criterion is "no elemental sodium." Any sodium removal process such as water, alcohol, steam-moist nitrogen, and evaporation will probably leave some amount of elemental sodium in cracks or crevices. Therefore, "no sodium" requires further definition for disposal of this type of waste. Although evaporation may approach the criterion of "no sodium" and is probably more amenable to remote handling for high level type waste, other processes require testing to determine if the residual sodium, after treatment, does not present a safety hazard. A subcontract was issued to Atomics International-Rockwell International Energy Systems Group (AI) to assist ANL in the development of shallow land burial criteria for sodium-bearing waste.

There are three phases in this program - (1) gather information on previous and present disposal criteria; (2) determine the effect of tritium on the environment; and (3) perform the necessary testing to develop shallow land burial criteria. The conclusions from the first phase were that burial sites interpret the present rules as "no sodium" and the burden of what constitutes a safe quantity of residual sodium rests with the waste generator. Past experience indicates no reported incidents when sodium-bearing components were buried. Data were not available on residual sodium after cleaning components except for some results from evaporation. As to phase 2, there does not seem to be any significant environmental problem from tritium releases from a water-sodium reaction from disposed waste. No further work is anticipated for these two phases.

As to phase 3, testing was initiated in FY-81 in a closed container (Fig. 1) with small quantities of elemental sodium and with water being added at controlled rates. The quantities of sodium (0.275 g to 7.5 g) used in these tests were assumed to be equivalent to that amount remaining after cleaning. The rate of water between 1-20 cc per min was equivalent to a rate which could occur when there is a 5 mil hole in a waste cylinder. The results of the tests are presented in Tables 1 and 2. There were two types of tests - one with bulk sodium and the other with sodium dispersed over a surface. In the six

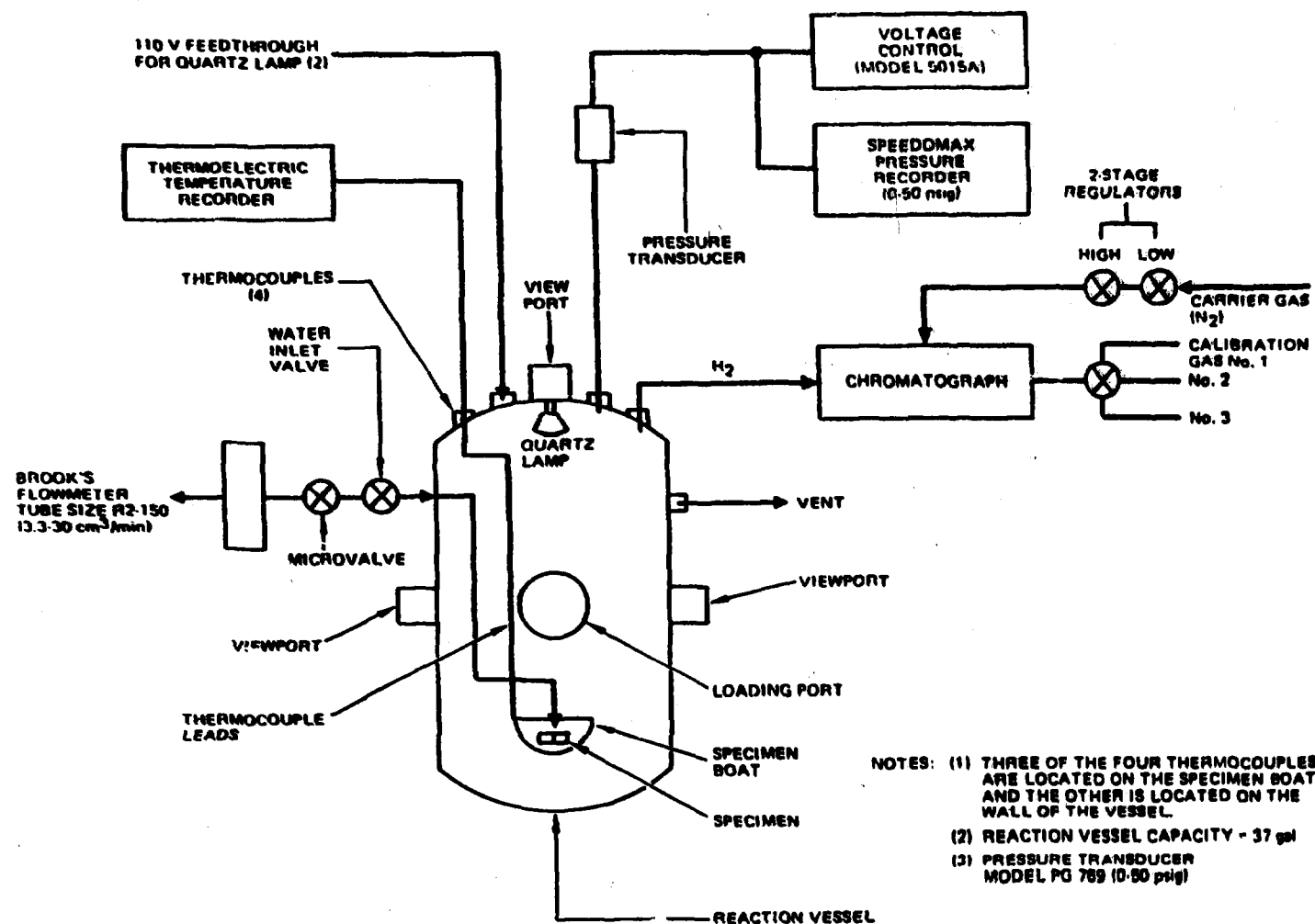


Figure 1: Sodium-Water Reactions Test Schematic

TABLE 1  
RESULTS OF DIRECT SODIUM-WATER REACTIONS

| Test No. | Amt. of Sodium (gm) | Water Add'n Rate (Range) (cc/min) | Total Water (cc) | Pressure (Max) (psig) | Temp (°F)* |     |     |     | Conc. H <sub>2</sub> (ppm) |        |        |
|----------|---------------------|-----------------------------------|------------------|-----------------------|------------|-----|-----|-----|----------------------------|--------|--------|
|          |                     |                                   |                  |                       | 1          | 2   | 3   | 4   | Max                        | Min    | Avg    |
| 1        | 3.75                | 5.0                               | 100              | 1.2                   | 90         | 140 | 100 | 320 | 160                        | 10     | 60     |
| 2        | 3.8                 | 10.0-15.0                         | 100              | 2.4                   | 90         | 140 | 160 | 206 | 18,000                     | 10     | 16,000 |
| 3        | 5.0                 | 10.0                              | 100              | 0.7                   | 100        | 140 | 160 | 260 | 74,000                     | 14,000 | 27,000 |
| 4        | 7.5                 | 2.5-3.0                           | 100              | 1.2                   | 90         | 620 | 110 | 260 | 12,000                     | 70     | 2,000  |
| 5        | 1.6                 | 1.0-5.0                           | 100              | 0.82                  | 90         | 470 | 90  | 120 | 4,000                      | 270    | 2,300  |
| 6        | 1.5                 | 1.6-2.5                           | 100              | 0.60                  | 100        | 390 | 110 | 110 | 2,900                      | 820    | 1,800  |

Comments

Test No. 1 - Vessel opened before test completed; H<sub>2</sub> data suspect.

Test No. 2 - Repeat of Test #1 with diaphragm pump; Na in pieces; small explosions heard.

Test No. 3 - Sodium in one piece, fire and small explosions.

Test No. 4 - Loud explosion; water dump tube bent due to explosion.

Test No. 5 - No explosions; fire and smoke.

Test No. 6 - No explosions; fire and smoke. (Repeat of Test #5)

\*Thermocouple readings at following locations:

1. Ambient, vessel environment
2. Bottom of boat, in proximity of Na/specimen
- 3,4. Sides of specimen boat

TABLE 2  
RESULTS OF REACTIONS OF SODIUM DISTRIBUTED ON SPECIMENS, WITH WATER

| Test No. | Specimen Weight (gm) | Amount of Sodium (gm) | Water Add'n Rate (Range) (cc/min) | Total Water (cc) | Pressure (Max) (psig) | Temp (°F)* |     |     |     | Conc. H <sub>2</sub> (ppm) |      |      |
|----------|----------------------|-----------------------|-----------------------------------|------------------|-----------------------|------------|-----|-----|-----|----------------------------|------|------|
|          |                      |                       |                                   |                  |                       | 1          | 2   | 3   | 4   | Max                        | Min  | Avg  |
| 7        | 993.6                | 3.8                   | 10.0-20.0                         | 280              | 2.0                   | 100        | 120 | 160 | 120 | 1200                       | 46   | 1200 |
| 8        | 149.7                | 1.0                   | 4.0-6.0                           | 180              | 0.42                  | 100        | 100 | 120 | 120 | 600                        | 2600 | 3100 |
| 9        | 1069.2               | 0.4                   | 20.0-3.0                          | 140              | 0.25                  | 110        | 130 | 130 | 130 | 460                        | 48   | 400  |
| 10       | 824.4                | 0.275                 | 1.0-2.0                           | 200              | **                    | 110        | 120 | 120 | 120 | 2500                       | 220  | 2400 |
| 11       | 640                  | 0.853                 | 4.0-5.0                           | 200              | **                    | 100        | 120 | 120 | 140 | 4600                       | 40   | 3200 |

Comments

Test No. 7 - Support plate for Few Tube Steam Generator, supported by bottom plate; most sodium at bottom of 0.625 inch hole; loud explosion heard.

Test No. 8 - Stainless steel bolt with sleeve - H<sub>2</sub> sample line near specimen boat; specimen not on water path, reactions occurred as water level rose. Bubbles, foam seen followed by flames.

Test No. 9 - Square steel blocks fastened with bolt; Na squeezed in between; joint line on water paths; only bubbles seen - visibility clear throughout; residual sodium ~0.16 gm after test.

Test No. 10- Bolted angle irons. No fumes or flames; only bubbles, pressure monitored by water manometer; visibility clear throughout test.

Test No. 11- Three bellows seal valves assembled. Sodium squeezed into valve seal areas prior to assembly. Only bubbles seen. Visibility clear, manometer used.

\*Thermocouple readings at following locations: 1. Ambient vessel environment  
2. Bottom of boat  
3,4. Sides of specimen boat

\*\*Measured with water manometer Test #10: ~2 inches of water  
Test #11: ~0.1 inches of water

tests (1.5 g to 7.5 g) with bulk sodium there were some explosions and fire and smoke. In the five tests with dispersed sodium (0.275 g - 3.8 g), there was an explosion with the 3.8 g and some flames with the 1 g. There were no unusual reactions with the remaining three. Preliminary conclusions are (1) bulk sodium is not desirable for shallow land burial even though container integrity was maintained during the testing and (2) dispersed sodium may be a safer method for burial depending upon the results from future testing.

The fiscal year 1982 objectives are as follows:

- (1) Fabricate test specimens
- (2) Conduct experiments of water and moisture reactions with sodium-wetted specimens (base line measurements)
- (3) Conduct experiments of water and moisture reactions with sodium-wetted and evaporation cleaned specimens.

Five pairs of metal crevice specimens were designed and fabricated. They are made of Type 304 stainless steel. Three of the specimens are cylindrical in shape with crevice dimensions of 1, 6 and 13 mils, respectively. The fourth is a rectangular crevice specimen with crevice dimensions of 3.3 in. x 0.39 in. x 0.039 in. (39 mil crevice). Finally, the fifth specimen consists of three rectangular blocks mated and bolted together, forming an inner circular crevice between the blocks and the bolt resulting in crevices of 40 mil and less than 1 mil. Figure 2 presents a picture of the specimens.

The specimens have a range of crevice volumes, surface areas and weights so as to subsequently provide a range of amounts of sodium to accumulate in the crevices and surfaces when they are sodium wetted. They are weighed before and after wetting in a sensitive top loading electronic balance to obtain the amounts of sodium retained upon wetting and draining.

Sodium wetting of the specimens was carried out in the ESG High Temperature Sodium Loop. This was accomplished by suspending the specimens from the top of a perforated stainless steel bucket and loading the bucket into a test pot section of the loop. Care was taken to avoid contact between the specimens. Wetting was performed at ~1000°F for a period of 48 hours. The specimens were subsequently removed in an argon atmosphere and transferred to a glove box.

One set of five specimens were used for the baseline tests of water reactions. These tests are conducted for comparison purposes since they are not cleaned after wetting. Therefore, they are used to compare results from various types of cleaning. Initially, water addition rates of 3 cc/hr were used during the first test. The test was continued until the specimen was completely submerged in water. Based on the data obtained in this test, it was determined that reaction rates were quite slow, and doubling the water addition rate would



**METAL CREVICE SPECIMENS**

**figure 2**

reduce the test duration by half without hampering data collection. Hence, the remaining four tests were conducted with a water addition rate of 6 cc/hr.

Typically, the tests were run with water addition continuing until the specimen was submerged in water, and allowing measurements of temperatures, pressure and hydrogen concentrations to continue until their values reached a steady state. Periodic visual observations through the view ports indicated that the reactions were slow; no fires, fumes, or explosions occurred.

At the end of each test, the specimen was retrieved, dried in argon, and weighed to obtain the weight of unreacted sodium (and any reaction residues); the specimen was cleaned of the unreacted sodium, dried, and weighed again to obtain the weights of the clean specimen.

Data obtained during the five water reaction tests are shown in Table 3, including the maximum temperatures, pressures, and hydrogen concentrations observed. In general, the maximum hydrogen concentrations observed during the tests increase with the amount of sodium reacted.

The second set of five specimens were reacted with moisture during five tests, each lasting about one week in duration. The specimens have identical geometries as those for which water reaction data were obtained. Slight variations in dimensions, total weights, and amounts of sodium deposited during wetting were noted.

For these moisture tests, the test apparatus was modified to collect moisture-reaction data by inserting a calibrated hygrometer probe through one of the available ports. The hygrometer and a chart recorder were connected to the probe outside the vessel. Moisture was introduced into the vessel by connecting a water column to the return line from the chromatograph, and by bubbling the returning air/hydrogen mixture through the column. It was possible to manually control the moisture level in the vessel by appropriate valving of the reentering gas mixture into the water column. No direct water additions to the sample boat took place during these tests. Moisture levels during each of the tests was maintained approximately in the 50-100% RH range. Analysis of the data obtained during the five tests indicates the following:

- The range of sodium available for reactions in these tests was 1.66 to 3.76 g which compares against the 0.68 to 3.38 g sodium available for the sodium-water reaction tests.
- Reactions were slow and no transient pressure, hydrogen concentration or temperature peaks were observed in all five tests. This is indicated by the absence of early, pressure/temperature peaks that were present during the water reaction tests.



TABLE 3  
RESIDUAL SODIUM-WATER REACTION TEST DATA

| Specimen<br>Description                                                                                                                                                       | Specimen<br>wt. (g) | Amt. of Na<br>Reacted (g) | Amt. of Na<br>Unreacted (g) | Test<br>Duration (h) | Water<br>Addition<br>Rate (cc/h) | Max. Temp (°F) | Max. Press.<br>(psig) | Max. H <sub>2</sub><br>Conc. (ppm) |
|-------------------------------------------------------------------------------------------------------------------------------------------------------------------------------|---------------------|---------------------------|-----------------------------|----------------------|----------------------------------|----------------|-----------------------|------------------------------------|
| Rectangular crevice specimen; two rectangular flat plates bolted on four corners; crevice grooved into thicker plate part way.                                                | 512.12              | 1.35                      | 0.15                        | 281                  | 3                                | 64             | 0.32                  | 3850                               |
| Rectangular block specimen; 3 rect. blocks of the same sizes bolted together. Annular crevice in center block. (Countershank)                                                 | 1533.00             | 0.68                      | 1.08                        | 53                   | 6                                | 88             | 0.2                   | 2300                               |
| Cylindrical crevice specimen; crevice is formed by a 13 mil clearance between cylindrical housing and rod assembly; weld buildup on rod to maintain concentricity of crevice. | 988.23              | 3.38                      | 0.0                         | 88                   | 6                                | 98             | 0.120                 | 11500                              |
| Same as above; 6 mil clearance                                                                                                                                                | 1004.80             | 3.03                      | 0.0                         | 45                   | 6                                | 106            | 0.08                  | 10000                              |
| Same as above; 1 mil clearance                                                                                                                                                | 1011.16             | 1.91                      | 0.0                         | 49.5                 | 6                                | 99             | 0.04                  | 7000                               |

- Larger quantities of unreacted metallic sodium were observed in all five tests. This is substantiated by observation of continued hydrogen gas release during post-test cleaning.

All the ten specimens used in the base line water and moisture reaction tests were cleaned, weighed in the glove box, and loaded into the high temperature sodium loop test station for wetting in sodium, so that data on evaporation-cleaned specimens could be collected. The wetting resulted in up to 3.2 g of sodium deposited which compared favorably with the amounts used during the base line data tests. One set of five specimens were kept for control and the other set was evaporation cleaned using MEDEC parameters of 900°F and  $10^{-3}$  torr for 20 hours. The cleaned specimens were reweighed to determine the amount of residue remaining. The 1 and 6 mil crevice specimens had no discernable sodium remaining; however, the 13 mil specimen had 1.4 g of the original 3.2 g remaining.

This 13 mil specimen was then placed in the test chamber for a water reaction test. Although the reaction appeared to be benign as evidenced by no bubbles, fumes or flames, some reaction was indicated by the presence of 0.36% hydrogen in the test chamber. Stoichiometric calculations of the amount of hydrogen that would be generated from 1.4 g of sodium agreed quite closely with the measured hydrogen generation thus indicating that all the residue (1.4g) was elemental sodium. The identical control specimen without evaporation was then tested and resulted in 1.1% hydrogen generated. The two specimens were then disassembled and visual observation indicated that all sodium had been reacted during the water tests. This appears to indicate that for intermediate size crevices (around 13 mil) water will penetrate and react with sodium not removed by evaporation cleaning. The reaction is slow and benign.

Two additional specimens were prepared with tapped 0.5 in. diameter holes. With a bolt threaded into the hole this specimen represents a pool of sodium with a small crevice outlet. These specimens were sodium wetted with one and two grams of sodium respectively. The two gram specimen was placed in the test chamber for a water reaction test. Although the reaction was more vigorous as evidenced by the hydrogen evaluation, no bubbles, fumes or flames were observed indicating that small pockets of sodium connected by a small crevice may not represent a significant hazard to shallow land burial. The second specimen was evaporation cleaned using MEDEC parameters and all sodium was removed.

During FY83, water reaction tests will continue using the same specimens. The specimens will be sodium-wetted and then cleaned using the water vapor-nitrogen cleaning method and the alcohol cleaning method. A peer review will be conducted in midyear.

On the basis of the data obtained, criteria will be developed for the safe, shallow land burial of low level, radioactive sodium wastes. In addition, quality control and verification procedures based on the endpoints of sodium removal methods will be established.

The purpose for developing these procedures is to provide assurance to the burial sites that processed wastes shipped to burial sites meet the criteria developed. The procedures will describe (1) the estimated initial amounts of sodium in a given package, (2) the method(s) by which the waste was processed, and the endpoints/estimates of sodium remaining on the wastes, (3) results of passivation, if and where applicable, and (4) container and packaging information. The procedures and data would be supplied to the burial site operator with each shipment.

The end results of this task will be to provide the elements of the quality control and verification steps that the waste generator may adopt in order to assure the burial sites of the safety measures taken in processing and packaging the sodium wastes.

### BACKGROUND

- RADIOACTIVE SODIUM-BEARING WASTE FROM LMFBR PROGRAMS IS STORED.
- DOE-ID ORIGINALLY STATED: SHALLOW LAND BURIAL WILL BE CONSIDERED ON A CASE BASIS.
- DOE-ID/FG&G ISSUED WASTE PACKAGING CRITERIA PROHIBITING ELEMENTAL SODIUM.
- DOE AND COMMERCIAL BURIAL GROUNDS NOW PROHIBIT ELEMENTAL SODIUM.

- ONE PURPOSE WAS TO DEVELOP METHOD OF REMOVING SODIUM FROM WASTE TO MEET "NO" CRITERION.
- SELECTED MEDEC (MELT/EVAPORATION) PROCESS TO DEMONSTRATE SINCE IT SHOULD MOST CLOSELY MEET THE "NO" CRITERION AND WOULD BE MORE AMENABLE TO REMOTE HANDLING TECHNIQUES.
- IDENTIFIED NEED FOR DEVELOPMENT OF SHALLOW LAND BURIAL CRITERIA FOR SODIUM-BEARING WASTES.

- SHALLOW LAND BURIAL OF SODIUM-BEARING WASTE PRESENTS ONLY TWO BASIC DIFFERENCES FROM NORMAL LOW LEVEL WASTE:
  1. SAFETY ASPECT OF SODIUM-WATER REACTION.
  2. TRITIUM RELEASE DURING THE SODIUM-WATER REACTION.

### THREE TASKS FOR 1981

1. BACKGROUND DATA GATHERING FROM FOUR GENERATIONS AND SIX DISPOSAL SITES.
2. TESTS OF TRITIUM DIFFUSION THROUGH SOIL (TRITIUM).
3. RESIDUAL SODIUM - WATER REACTION TESTS IN A CLOSED CONTAINER (SAFETY).

### TRITIUM SOIL PERMEATION TESTS

- CONCERN WAS THAT  $^3\text{H}$  RELEASED UNDERGROUND WOULD BE OXIDIZED TO TRITIATED WATER RESULTING IN A POTENTIAL LEACHING PROBLEM.
- TESTING CONSISTING OF PASSING  $\text{H}_2$  THROUGH A 15 FOOT SOIL COLUMN USING BOTH CALIFORNIA AND IDAHO SOILS.
- RESULTS INDICATED:  
  
LITTLE OR NO CONVERSION TO TRITIATED WATER.  
  
TRITIUM IS TRAPPED, AS A GAS, IN THE SOIL.



- TESTING TERMINATED BASED PRIMARILY ON RESULTS OF  
SODIUM-WATER REACTION TESTS WHICH SHOWED TRITIUM  
NOT TO BE A PROBLEM.

## SODIUM-WATER REACTION TESTS

1981

PURPOSE: DETERMINE CHANGES IN PRESSURE, TEMPERATURE  
AND  $H_2$  CONCENTRATION WHEN REACTION TAKES  
PLACE IN A CLOSED CONTAINER.

### WORK INVOLVED:

- TESTS WITH SMALL AMOUNTS OF SODIUM READILY ACCESSIBLE  
IN A SAMPLE BOAT (BULK SODIUM).
- TESTS WITH SODIUM DISTRIBUTED OVER THE SURFACE OF A  
SPECIMEN.

### SODIUM-WATER REACTION TESTS (CONT'D)

- AMOUNTS OF SODIUM FROM 0.275 TO 7.50 GMS.
- VARYING WATER FLOW RATES INTO THE CLOSED CONTAINER.

#### RESULTS:

- FLAMES AND SOMETIMES SMALL EXPLOSIONS OCCURRED WITH BULK SODIUM TESTS.
- MOST TESTS SHOWED NO FLAMES OR LITTLE VISIBLE REACTION DURING THE DISPERSED SODIUM TESTS.

#### PRELIMINARY CONCLUSIONS:

- READILY AVAILABLE, "BULK" SODIUM IS NOT DESIRABLE AS WASTE FOR SHALLOW LAND BURIAL.

SODIUM WATER REACTION TESTS (CONT'D)

- DISPERSED SODIUM MAY PROBABLY BE SAFELY BURIED, FURTHER TESTING REQUIRED.
- BECAUSE OF THE SLOW REACTION, IF ANY, OF THE DISPERSED SODIUM, TRITIUM WOULD NOT BE A PROBLEM AND THE TRITIUM TESTING SHOULD BE TERMINATED.

### 1982 GOALS

- FABRICATE FIVE PAIRS OF CREVICE SPECIMENS.
- CONDUCT BASELINE MEASUREMENT EXPERIMENTS WITH WATER AND MOISTURE REACTIONS WITH SODIUM-WETTED SPECIMENS.
- CONDUCT EXPERIMENTS WITH SODIUM-WETTED AND EVAPORATION CLEANED SPECIMENS.

1982

STAINLESS STEEL CREVICE SPECIMENS FABRICATED:

- THREE CYLINDRICAL CREVICES (1 MIL, 6 MIL, 13 MIL)
- ONE RETANGULAR CREVICE 39 MIL,
- ONE SERIES OF BLOCKS BOLTED TOGETHER RESULTING IN AN  
INNER CIRCULAR CREVICE OF ABOUT 40 MIL.

1982

WATER REACTION BASELINE TESTS

- ONE OF EACH PAIR OF SPECIMENS WERE TESTED
- REACTIONS WERE SLOW; NO FIRES, FUMES, OR EXPLOSIONS
- HYDROGEN GENERATED RANGED 0.23% TO 1.15%

1982

EVAPORATION CLEANING TESTS

- ALL TEN SPECIMENS WERE WETTED, ONE OF EACH PAIR KEPT FOR CONTROL.
- REMAINING SET OF FIVE WERE EVAPORATION CLEANED AT 900°F AND  $10^{-3}$  TORR.
- THE 13 MIL SPECIMEN HAD LARGEST AMOUNT OF RESIDUAL SODIUM (1.4 g).
- THE 13 MIL SPECIMEN WAS TESTED FOR WATER REACTION:

REACTION WAS SLOW

MAXIMUM H<sub>2</sub> GENERATION WAS 0.36%

ALL RESIDUAL SODIUM REACTED WITH WATER



1982

MOISTURE REACTION TESTS

- SECOND SET OF FIVE WERE TESTED.
- AMOUNTS OF SODIUM INITIALLY DEPOSITED SIMILAR TO THAT DEPOSITED ON WATER REACTION TEST SPECIMENS.
- REACTIONS WERE MUCH SLOWER.
- AFTER TESTS ALL FIVE SPECIMENS HAD LONGER AMOUNTS OF UNREACTED SODIUM REMAINING THAN WATER REACTION TEST SPECIMENS.

- TWO ADDITIONAL SPECIMENS FABRICATED TO REPRESENT A POCKET OF SODIUM WITH SMALL CREVICE OUTLET.
- ONE SPECIMEN WAS EVAPORATION CLEANED RESULTING IN ALL SODIUM REMOVED.
- THE OTHER WAS WATER REACTION TESTED WITHOUT CLEANING. ALTHOUGH THE REACTION WAS MORE VIGOROUS THAN OTHER BASELINE TESTS, NO FIRES, FUMES, OR BUBBLES WERE OBSERVED.

### PRELIMINARY 1982 CONCLUSIONS

- WATER WILL PENETRATE SOME CREVICES, EVAPORATION CLEANING MAY NOT REACH.
- EVAPORATION WILL CLEAN SOME CREVICES WATER WILL NOT PENETRATE.
- SMALL AMOUNTS OF SODIUM DISPERSED AND IN CREVICES MAY NOT BE A PROBLEM FOR SAFE SHALLOW LAND BURIAL.
- MATERIAL WHICH HAS BEEN CLEANED BY WATER OR EVAPORATION MAYBE SAFE FOR SHALLOW LAND BURIAL.
- POCKETS OF SODIUM WITH SMALL CREVICE OUTLETS SHOULD BE SAFE FOR SHALLOW LAND BURIAL.

### 1983 OBJECTIVES

- CONTINUE TESTING USING ALCOHOL CLEANING AND WATER VAPOR-NITROGEN CLEANING.
- DEVELOP SHALLOW LAND BURIAL CRITERIA.
- CONDUCT PEER REVIEW OF PROPOSED CRITERIA.
- DEVELOP QA AND VERIFICATION PROCEDURES FOR GENERATORS TO USE TO ASSURE BURIAL SITE OPERATORS THAT SHIPMENT IS SAFE.

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## TECHNOLOGY DEVELOPMENT

RFP-3436

## TO MEET

## THE LLW - NITRATE CHALLENGE

AL 3.5.1 AR

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August 4, 1982

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U.S. Department of Energy  
Contract DE-AC04-76DPD3533

## ABSTRACT

Work performed in fiscal year 1982 to test technologies for destroying nitrates in low-level radioactive waste is presented. Three technologies were tested at the laboratory scale for destroying nitrates in Defense Waste Process Facility LLW. The results of these tests are summarized.

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INTRODUCTION

Waste treatment operations at defense and commercial nuclear facilities generate large quantities of low-level radioactive waste nitrates. Increased pressure from the Nuclear Regulatory Commission (NRC) and Environmental Protection Agency (EPA) to remove these materials from nuclear plant effluents, and to protect low-level waste repository ground water from nitrate contamination, requires that waste management practices be changed. The task at Rocky Flats deals with the application of waste treatment technologies to effect that change.

An investigation of the source of nitrate wastes in the commercial fuel cycle and defense nuclear operations was completed in fiscal year

1981.<sup>1</sup> The combined inventory from these two waste sources is estimated to be in excess of  $1.2 \times 10^8$  Kg. These quantities and their composition (both in storage and being generated) emphasize the need for continuing research into processes which will render the material environmentally acceptable.

An insight into the regulations, both proposed and in effect, which influence present and future waste management practices for waste nitrate is important. The proposed NRC Rule 10 CFR 61, establishes minimum waste form requirements which current waste nitrates will not meet. This rule is supplemented by EPA Rule 40 CFR 141, the National Primary Drinking Water Quality Criteria (DWQC). This rule limits all plant effluents, as well as ground water seepage from burial sites, to less than 10 mg/l elemental N (45 mg/l  $\text{NO}_3^-$ ).

This task supports alpha milestone "B" and the Advanced Treatment Milestone of the National Low-Level Waste Management Program (NLLWMP). Technologies are being developed which will provide alternative solutions for nitrate disposal so nuclear facility operators can meet the challenge of stricter regulations.

#### TASK DESCRIPTION

##### OBJECTIVE

The objective of this task is to develop and demonstrate a process(es) to eliminate or reduce the amount of nitrates in LLW streams to an acceptable level without the generation of objectionable oxides of nitrogen. The products of the process are expected to be nitrogen and carbonates or hydroxides, which are environmentally acceptable as well as being amenable to a wide spectrum of immobilization options to meet future transportation and disposal criteria.

##### BUDGET

The fiscal year 1982 budget is \$100K. Funding was delayed until the beginning of the second quarter because the task was not included in the continuing resolution from Congress.

The budget is allocated to support one technical person for the remaining three quarters of the fiscal year. Funds are also allocated to provide administrative and matrix management support. There is \$20K allocated for contract testing in equipment not available at Rocky Flats.

## SCHEDULE

The major task for fiscal year 1982 involves process evaluation studies. The goal is to select one or more processes for bench scale process development and subsequent pilot scale demonstration. Figure 1 describes the 3-year task schedule. There are four lead-office monitored milestones for fiscal year 1982. Work is on schedule to complete these milestones.

## TECHNOLOGICAL INVESTIGATIONS

Based on the results of fiscal year 1981 investigations,<sup>1</sup> nitrate bearing waste streams fall into three basic categories: dilute liquid (<10,000 ppm nitrate); concentrated liquid (up to 35 wt %); and artificially or naturally dried solids.

Liquid waste streams with nitrate concentrations below 10,000 ppm are successfully being treated with a biological denitrification process developed at the Oak Ridge National Laboratory (ORNL).<sup>2</sup> The ORNL process employs an anaerobic bacteria to reduce nitrate or nitrite to gaseous molecular nitrogen in a fluidized bed reactor. The process has been developed through the pilot stage and is ready for industrial application. Since development is essentially complete, biodenitrification is not being pursued as part of this task. However, the work will be summarized in the final report.

Three technologies for destroying nitrates in the two remaining categories of waste were selected for laboratory scale testing. They use a chemical reductant in an aqueous medium, in a molten salt medium, and direct thermal decomposition of the nitrate itself.

### Chemical Reduction

Chemical reduction in an aqueous medium has been utilized in the United States and foreign countries. In this technique, a reducing agent is added to a saturated liquid nitrate waste prior to a drying procedure. This process is discussed in the patent literature but there is no data on large-scale process applications.

The reducing agents most mentioned are urea, formic acid, and formaldehyde. A small amount of work was done with urea at RFP several years ago with promising results. The concept is desirable because (1) it could easily be implemented prior to the drying step in existing treatment processes, (2) capital costs would be small, (3) it would be applicable to waste streams with either high or low concentrations of nitrate, and (4) nearly all plant waste streams pass through an aqueous phase. The end product is expected to be molecular nitrogen, ammonia gas, and metal hydroxides.



FIGURE 1: MILESTONE SCHEDULE

| Task No. | Tasks                                                                                                                                                                                                      | FY-1982 |   |   |   | FY-1983 |   |   |   | FY-1984 |    |    |    |
|----------|------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------|---------|---|---|---|---------|---|---|---|---------|----|----|----|
|          |                                                                                                                                                                                                            | 1       | 2 | 3 | 4 | 1       | 2 | 3 | 4 | 1       | 2  | 3  | 4  |
| 1.1      | PROCESS EVALUATION STUDIES<br>1. Process test contract awarded<br>2. Process testing completed<br>3. Status report issued                                                                                  |         | 1 | 2 | 3 |         |   |   |   |         |    |    |    |
| 1.2      | CANDIDATE PROCESS DEVELOPMENT<br>4. Equipment purchased and/or fabricated<br>5. Equipment installed<br>6. Process optimization completed<br>7. Laboratory development completed<br>8. Status report issued |         |   |   |   | 4       | 5 |   | 8 |         |    |    |    |
| 1.3      | PILOT-SCALE PROCESS DEMONSTRATION<br>9. Installation completed<br>10. Demonstration completed<br>11. Report issued                                                                                         |         |   |   |   |         | 6 | 7 |   | 9       |    | 10 |    |
| 1.4      | PROCESS APPLICATION STUDY<br>12. Producer update study completed<br>13. Report issued (supports LL082.2)                                                                                                   |         |   |   |   |         |   |   |   | 12      | 13 |    |    |
| 1.5      | PRODUCT DISPOSITION STUDY*<br>14. Reuse and immobilization tests completed<br>15. Report issued                                                                                                            |         |   |   |   |         |   |   |   |         |    | 14 | 15 |
| 1.6      | PROGRAM MANAGEMENT<br><br>* Increment funding only                                                                                                                                                         |         |   |   |   |         |   |   |   |         |    |    |    |

Chemical reduction of solid nitrates in a molten salt medium can be achieved using carbon as the reducing agent. Preliminary testing of this concept had been conducted at RFP. The results were promising enough to indicate process feasibility. The end product is sodium carbonate, which is nontoxic and amenable to immobilization techniques. Another advantage is that the reaction results in a ~35 percent weight loss. The carbon/nitrate waste mixture must be heated to 400°C to initiate the reaction. The reaction itself is exothermic. Initial results indicated that only very low concentrations of  $\text{NO}_x$  were generated but additional work is required to justify this. The process is applicable to solid wastes with a high concentration of nitrates such as Savannah River Plant, Defense Waste Process Facility, low-level waste (SRP-DWPF-LLW) and RFP salts.

### Thermal Decomposition

Thermal decomposition of nitrates occurs between 500° and 1000°C, and is accompanied by the release of  $\text{NO}_x$ ,  $\text{N}_2$ , and  $\text{O}_2$ . A weight loss of about 60-65 percent occurs. This process could be conducted with liquids, but it would appear to be more applicable to wastes that have already been solidified in a previous treatment step. The  $\text{NO}_x$  product could possibly be converted to  $\text{HNO}_3$  in subsequent process steps, depending on needs.

Because of  $\text{NO}_x$  generation, the only thermal decomposition process given serious consideration was the Thagard high temperature fluid wall reactor. This reactor is capable of heating materials to ~2200°C in microseconds and, consequently, disassociate compounds to their elemental forms almost instantly. Under these conditions, the formulation of  $\text{NO}_x$  may be eliminated, providing an attractive nitrate decomposition process.

## RESULTS

The results reported here are preliminary because not all of the analytical results are available.

### CHEMICAL REDUCTION

#### Aqueous Medium

Urea was selected as the first choice reducing agent, with formaldehyde the second. A test plan was developed for urea with three series of tests: evaporation, evaporation to dryness, and a solid nitrate-urea reaction in molten urea. The parameters varied were evaporation rate, urea-nitrate mole ratio, and temperature. Criteria for success is <500 ppm  $\text{NO}_x$  in the offgas and >90 percent decomposition of the nitrate in the sample.

Three formulations were tested: industrial grade  $\text{NaNO}_3$ , simulated SRP-DWPF-LLW, and Rocky Flats spray dried nitrate waste. Formulations are shown in Table 1.

Destruction efficiency was measured by determining the amount of nitrate in dissolved residue using a specific ion electrode. Effluent emissions were measured for  $\text{NO}_x$  using a calibrated analyzer.

The initial residue analytical results are shown in Table 2 and emissions in Table 3. There doesn't appear to be a significant reduction in volume.

### Molten Salt Medium

This concept was tested initially at Rocky Flats by injecting reagent grade  $\text{NaNO}_3$  into a fluidized bed of sand and charcoal. The product is  $\text{Na}_2\text{CO}_3$  and nitrogen gas.  $\text{NO}_x$  emissions were low and the process promising. However, heat from the exothermic reaction caused the unreacted  $\text{NaNO}_3$  and the  $\text{Na}_2\text{CO}_3$  to melt. This resulted in agglomeration of the sand fluidizing medium and subsequent loss of fluidization.

A molten salt median was then investigated. The Environmental and Energy Systems Division (EESD) of Rockwell International's Energy Systems Group has a proprietary molten salt reactor which utilize  $\text{Na}_2\text{CO}_3$  (the reaction product) as the molten salt medium. The equipment has been demonstrated to successfully destroy many forms of nuclear waste, as well as toxic and hazardous wastes.

A test plan was developed and carried out at EESD. Two series of tests were run using the simulated SRP-DWPF-LLW in Table 1. The carbon source was a refinery by-product called delayed coke. Parameters varied were temperature and purge gas. An excess of carbon was maintained throughout the experiment. Criteria for success was  $<500$  ppm  $\text{NO}_x$  in the offgas and  $>90$  percent destruction of nitrate in melt residues. The offgas was monitored continuously for  $\text{CO}$ ,  $\text{CO}_2$ , and  $\text{NO}_x$ . The initial results of the residue analysis are shown in Table 2 and the emissions in Table 3.

### THERMAL DECOMPOSITION

Thermal decomposition occurs between  $500^\circ$  and  $1000^\circ\text{C}$ . Decomposition products are metal oxides and  $\text{NO}_x$ . The Thaggard Research (Irvine, CA) high temperature fluid wall reactor (HTFW) operates at  $2200^\circ\text{C}$ . It was postulated that  $\text{NO}_x$  would not be formed, but rather that  $\text{N}_2$ ,  $\text{O}_2$ , and metal oxides. It was further postulated that the metal oxides could be combined with  $\text{SiO}_2$  and  $\text{CaO}$  to form a vitreous slag, and added carbon would combine with the oxygen, thus reducing electrical power consumption. The proposed products from this process would be flareable gases and low leachrate glasses.

Table 1

Nitrate Waste Formulations Tested  
In Weight Percent

| Compound                 | Simulated<br>SRP-DWPF-LLW | RFP-LLW | Industrial<br>Grade $\text{NaNO}_3$ |
|--------------------------|---------------------------|---------|-------------------------------------|
| $\text{NaNO}_3$          | 40                        | 49      | 97                                  |
| $\text{NaNO}_2$          | 14                        | --      |                                     |
| $\text{KNO}_3$           | --                        | 44      |                                     |
| $\text{NaOH}$            | 19                        | --      |                                     |
| $\text{NaAlO}_2$         | 9                         | --      |                                     |
| $\text{Na}_2\text{CO}_3$ | 9                         | --      |                                     |
| $\text{Na}_2\text{SO}_4$ | 8                         | 3       |                                     |
| $\text{NaCl}$            | 0.2                       | 3       |                                     |
| Misc.                    | bal                       | bal     |                                     |

Table 2  
Initial Results - Residues  
Simulated SRP-DWPF-LLW

| Technology Tested                        | Temp °C | Wt. Percent     |                 |                 |     | Soluble Residues, Percent |
|------------------------------------------|---------|-----------------|-----------------|-----------------|-----|---------------------------|
|                                          |         | NO <sub>3</sub> | NO <sub>2</sub> | SO <sub>4</sub> | Cl  |                           |
| Chemical Reduction<br>Aqueous Medium     | 100-200 | >10             | x               | x               | x   | >90                       |
| Chemical Reduction<br>Molten Salt Medium | 1000    | 0.7             | 0               | 0.3             | 1.8 | 98                        |
| Thermal Decomposition                    | 2200    | <1.0            | 0.3             | 1.9             | 0.3 | 14                        |

x Analysis incomplete

Table 3  
Initial Results - Gases  
Simulated SRP-DWPF-LLW

| Technology Tested                        | Temp °C | Cover Gas      | PPM             |     |                 |
|------------------------------------------|---------|----------------|-----------------|-----|-----------------|
|                                          |         |                | NO <sub>x</sub> | CO  | CO <sub>2</sub> |
| Chemical Reduction<br>Aqueous Medium     | 120     | air            | 10              | x   | x               |
| Chemical Reduction<br>Molten Salt Medium | 1000    | N <sub>2</sub> | 40              | 15* | 3*              |
| Thermal Decomposition                    | 2200    | N <sub>2</sub> | <10             | 7*  | <1*             |

\* Volume Percent

x Analysis incomplete

A contract valued at \$10K was negotiated with Thaggard to test the simulated SRP-DWPF-LLW in Table 1 in their 7.62-cm reactor. A series of three tests were planned in a matrix experiment. Criteria for success was <500 ppm NO<sub>x</sub> in the offgas and >90 percent destruction of nitrate in the residue. Parameters varied were carbon mole percent, CaO, and SiO<sub>2</sub> concentrations. The initial test was run without additives.

The offgases were analyzed using gas chromatography/mass spectroscopy. Slag residues were submitted to Rocky Flats laboratories for analysis. The initial results of the residue analysis are shown in Table 2 and emissions in Table 3. Scanning electron microscopy results of the slag residues indicate they are calcium silicate.

### CONCLUSIONS

The task is on schedule and within budget. Results are as yet incomplete. Sufficient information from the tests will be available by the end of fiscal year 1982 to select the processes for bench scale testing in fiscal year 1983. A summary report will be written as planned to complete the end of the fiscal year milestone.

### FUTURE WORK

The task is budgeted at \$250K for fiscal year 1983. This funding level will support two technical people and a Program Manager. Effort will be directed toward bench scale testing of selected processes. Bench scale testing will provide the economic and engineering information necessary to design and build pilot scale systems for testing in fiscal year 1984.

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ULTRAFILTRATION/REVERSE OSMOSIS  
(LIQUID TREATMENT SYSTEMS)

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ABSTRACT

This project is involved with the development of low-level liquid waste treatment processes. Reverse osmosis (RO) has been found to be an effective decontamination process with a large number of radioisotopes. Fiscal year 1982 saw the conclusion of reverse osmosis development with a plant design for a 40 gal/min RO unit. Background and engineering data were included to enable an applicability assessment and rough cost estimate to be made.

The removal of cobalt from aqueous solutions with adsorbents was investigated. It was found that solution pH, ammonium, and sulfite ions affect cobalt removal. The effect of these factors was measured and defined in a set of prediction equations.

A design for an adsorbent pilot plant was also produced which included a process schematic, solution criteria, recommended isotope-adsorbent combinations, and cost estimates.

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INTRODUCTION

This project directly benefits the Low Level Waste Management Program (LLWMP), milestone B: "Develop and document technology for improved waste treatment, handling, and packaging for shallow land burial site disposal" and milestone G: "Develop and document technology for waste treatment, handling, and packaging for greater confinement than shallow land burial." Specifically, this project is involved with the development of liquid waste treatment processes. During the life of this project, the processes have included ultrafiltration, reverse osmosis, and both organic and inorganic adsorbents which are applicable to the decontamination of aqueous waste streams originating in a large number of laboratories and power plants. Table 1 lists some typical applications of ultrafiltration and reverse osmosis.

During fiscal year 1982, the adsorbents and reverse osmosis projects were completed.

\*Mound is operated by Monsanto Research Corporation for the U.S. Department of Energy under Contract No. DE-AC04-76-DP00053.



Table 1 - TYPICAL APPLICATIONS OF ULTRAFILTRATION AND REVERSE OSMOSIS PROCESSES

| Plant (Name & Location)                          | Unit Size                       | Membrane Type and Configuration               | Total Solids Level (mg/L) | Source of Waste               | Volume Reduction | Reference |
|--------------------------------------------------|---------------------------------|-----------------------------------------------|---------------------------|-------------------------------|------------------|-----------|
| Tsuruga Station<br>Japan Atomic Power Comp.      | UF Pilot Plant<br>UF Full Scale | Noncellulosic<br>Tubular Membranes            | 10-15                     | Equipment<br>Drains           | 75-150           | 7         |
| Soviet Test Cells<br>Russia                      | RO Laboratory<br>Test Cells     | Cellulose-Acetate<br>Plant-Frame Membrane     | 500-700                   | Industrial<br>Radwaste        | 400              | 6         |
| Point Beach Nuclear Plant<br>Wisconsin - USA     | RO Pilot<br>Plant               | Polyamide Hollow<br>Fiber Membranes           | 90                        | Radwaste                      | 2-10             | 7         |
| H. B. Robsin Plant<br>S. Carolina - USA          | RO Pilot<br>Plant               | Cellulose-Acetate<br>Outside-in ceramic tubes | 7.1                       | Radwaste                      | 2                | 7         |
| Mound Laboratory<br>Ohio - USA                   | UF<br>Full Scale                | Noncellulose tubular<br>membranes             | 30-70                     | Fuel Reprocessing<br>Radwaste | 200              | 8         |
| Rocky Flats Plant<br>Colorado - USA              | RO Pilot Plant<br>& Full Scale  | Cellulose Acetate<br>Spiral Wound Membranes   | 221                       | Caustic Radioactive<br>Water  | 100-200          | 19        |
| Westinghouse Elec. Corp.<br>Pennsylvania - USA   | RO Laboratory<br>Scale          | Cellulose Acetate<br>Tubular Membranes        | -                         | Steam Generator<br>Blow Down  | 10               | 20        |
| Brunswick Steam Elec. Plant<br>N. Carolina - USA | RO<br>Full Scale                | Cellulose Acetate<br>Spiral Wound Membrane    | 5450-7400                 | Floor Drains                  | 3-5              | 7         |
| Hitachi Research Lab<br>Ibaraki - Japan          | RO<br>Pilot Plant               | Cellulose Acetate<br>Tubular Membrane         | 10-20                     | Laundry                       | 1000             | 21        |
| R. E. Ginna Station<br>New York, USA             | RO<br>Full Scale                | Cellulose Acetate<br>Tubular Membrane         | 2200-6100                 | Floor Drain<br>and Laundry    | 400              | 7         |
| Chalk River Nuclear Lab<br>Ontario - Canada      | RO<br>Pilot Plant               | Cellulose Acetate<br>Tubular Membrane         | 100-3000                  | Hot Shower<br>and Laundry     | 150-200          | 7         |

## MEMBRANE PLANT DESIGN

## REVERSE OSMOSIS

Reverse osmosis, as the title implies, is the reverse process of natural osmosis. Natural osmosis transports solvent from a dilute solution to a more concentrated solution through a semipermeable membrane. The driving force for this process is called osmotic pressure. Reverse osmosis is the movement of solvent from a concentrated solution to a more dilute solution. Pressure exceeding the osmotic pressure must be applied to the concentrated solution to effect this process. Figures 1 and 2 illustrate these mechanisms.

## THE MECHANISMS OF REVERSE OSMOSIS

There are two fundamental mechanisms of rejection at work in the reverse osmosis process. One is the mechanism of salt rejection. Salt rejection occurs because of the repulsion of the salt ions from the surface of the membrane and the absorption of water to the membrane surface. Because of the physical and chemical properties of cellulose acetate which was used as the membrane material in these experiments, a pure water layer about two molecules thick ( $\sim 10$  Å thick) develops at the surface of the membrane, with the higher valence ions being repelled to a greater distance. If ions of two different valences of a salt (e.g.,  $\text{Ca}^{+2}$  and  $2\text{Cl}^{-1}$ ) are in solution, the  $\text{Ca}^{+2}$  tends to hold the  $\text{Cl}^{-1}$  with it, and both ions are repelled from the membrane as if both had a valence of two. The highest valence ion of a salt is usually the dictating species in predicting membrane rejection performance. The mechanism of salt rejection with a cellulose acetate reverse osmosis membrane is illustrated in Figure 3. The second mechanism is that of organic rejection, in which the dissolved organics are rejected by a screening or sieving mechanism. The rejection of any given organic molecule is a function of membrane pore size, size of the molecule, and the geometry of the molecule. Ions complexed with organics of large molecular weight will usually be rejected at approximately the same rate as are the organics with which they are complexed.

## MEMBRANE MODULE CONFIGURATIONS

There are three basic types of membrane module configurations: tubular, spiral wound, and hollow fiber. Membranes in the form of a tube are inserted into tubular casings so that the membranes line the casings. The tubular casings, which have porous walls, serve both as the pressure vessel and as the support for the membranes. Solvent permeates the membranes and then seeps through the porous walls of the casings. Spiral-wound reverse osmosis modules are made with planar membranes. They are

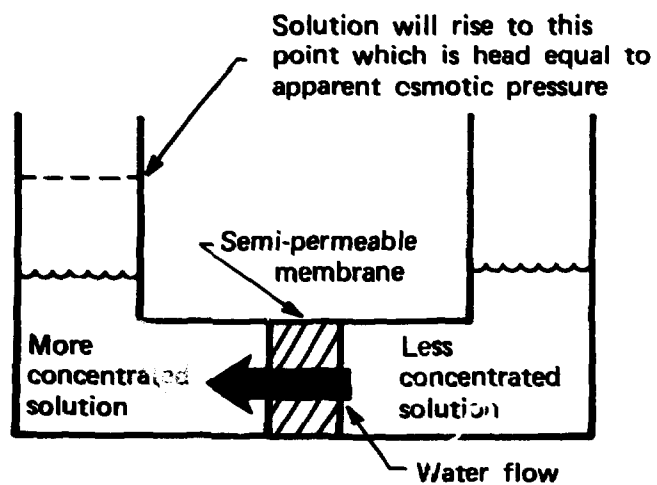


FIGURE 1 - Osmosis.

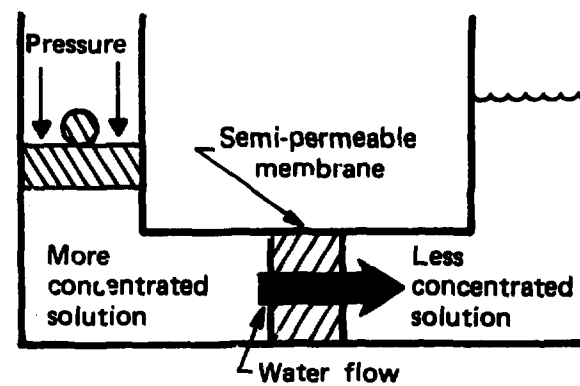


FIGURE 2 - Reverse osmosis.

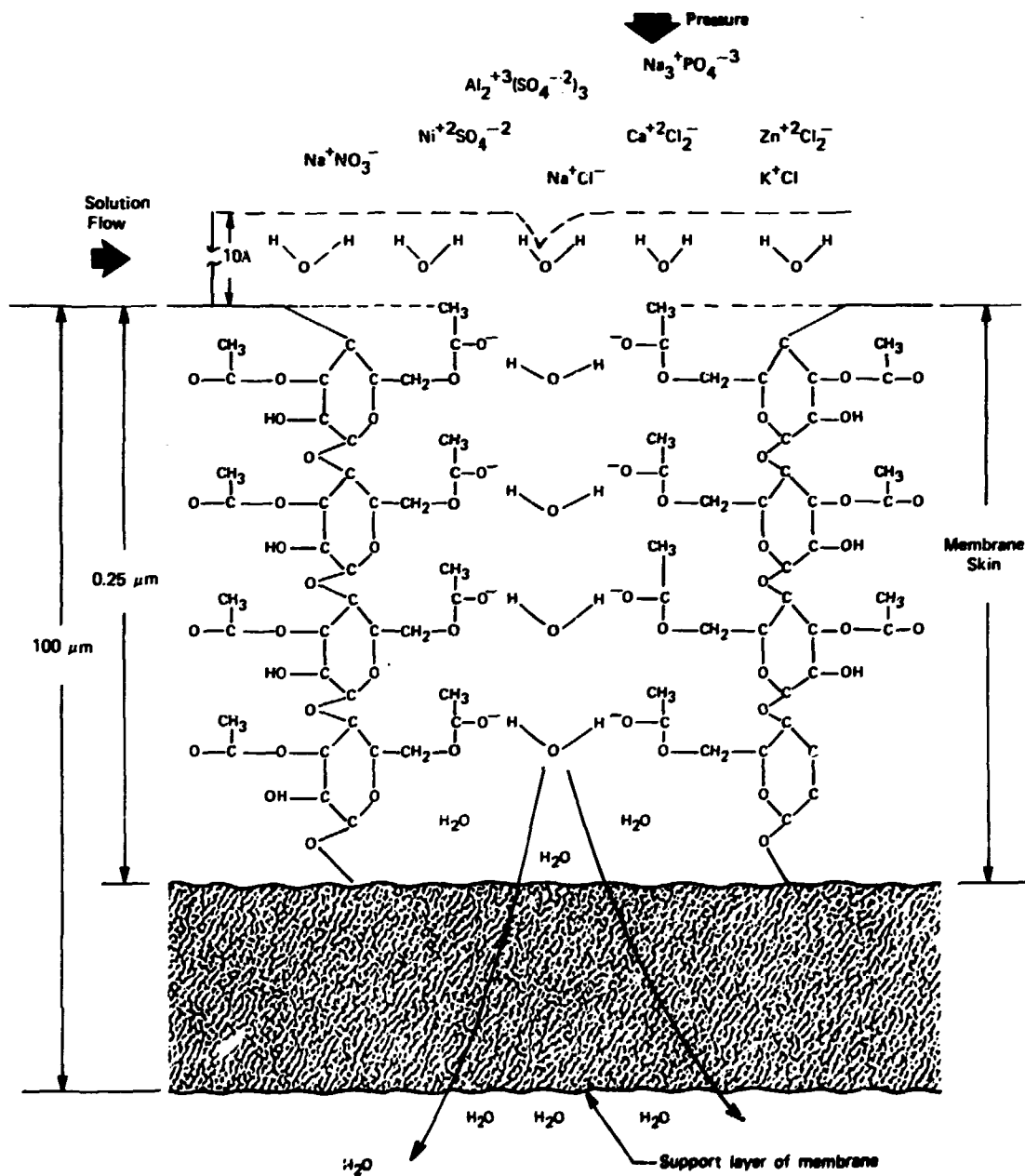


FIGURE 3 - Conception of salt rejection with a "97%" rejection membrane cellulose acetate.

fabricated by sandwiching a porous supporting material between the planar reverse osmosis membranes; sealing the edges of the membranes to each other and to a central tube; and then wrapping membranes, porous support, and a mesh feed-side spacer into a spiral around the central tube. The spiral of membranes, spacer, and porous support is slipped into a cylindrical casing capable of withstanding high pressures. The pressurized feed solution is fed into the casing so that it flows through the mesh feed-side spacers and along the surfaces of the membranes. Part of the solvent permeates the membrane and flows through the porous support material into the central tube, from which it is collected.

In hollow-fiber reverse-osmosis units, the membranes are spun into hollow fibers. These are almost as fine as a human hair (100 to 200  $\mu\text{m}$ ) but are hollow, with walls about 25  $\mu\text{m}$  thick. Cylinders this small need no supporting structure to withstand large pressures without collapsing. A bundle of long fibers is arranged in a U-shape, and the open ends of the U-shaped bundle are potted in plastic. The bundle with its potted ends is arranged in a cylindrical pressure shell. Feed solution is introduced into the middle of the fibers through a porous tube at one end of the cylinder, and the concentrated product leaves from an effluent tube at the same end. The permeate transfers through the fibers and flows through each fiber to the potted end, where it leaves the cylinder.

Some of the preceding types of membrane module configurations are more suitable for certain applications than are others. The features of the various types are compared in Table 2.

## MEMBRANE POROSITY

Reverse osmosis membranes are classified by NaCl rejection. There are several types of reverse osmosis membranes with different NaCl rejections. Table 3 lists some general specifications for five types of spiral-wound membranes manufactured by Osmonics Inc. Membranes with NaCl rejection of 0, 50, and 97 (manufactured by Osmonics Inc.) were tested at Mound for rejection of the following isotopes: plutonium-239, uranium-233, americium-241, neptunium-237, cesium-137, iodine-125, and cobalt-60. The 0 membrane, it should be pointed out, is an ultrafiltration membrane, not a reverse osmosis membrane. A reverse osmosis membrane is capable of rejection, or selective retention, of ionic impurities. The ionic impurities are repelled by the membrane and restricted from passing through the membrane pore. The size of the ionic impurity is normally smaller than the membrane pore. Reverse osmosis has become generally accepted as the removal of ionic impurities from water by means of a membrane. The rejection, or selective retention, of nonionic impurities such as organics and emulsified materials has become known as ultrafiltration (UF). Membranes that remove impurities based on size are classified as ultrafiltration membranes. In general, a UF membrane has become commonly accepted as a membrane that does not reject ionic materials. The 0 membrane, because ionic rejection was attempted at operating pressures associated with reverse osmosis, will be referred to as an RO membrane.

Table 2 - FEATURES OF DIFFERENT MEMBRANE CONFIGURATIONS

| Commercially Available Configuration | Advantages                                                                                                                                                                              | Disadvantages                                                                                                                                                                                                  |
|--------------------------------------|-----------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------|----------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------|
| Tubular                              | <ol style="list-style-type: none"> <li>1. Easily cleaned chemically or mechanically if membranes become fouled</li> <li>2. Can process dirty feeds with minimal pretreatment</li> </ol> | <ol style="list-style-type: none"> <li>1. High holdup per unit membrane area</li> <li>2. Relatively expensive</li> <li>3. Requires moderately large floor space per square foot of membrane surface</li> </ol> |
| Spiral Wound                         | <ol style="list-style-type: none"> <li>1. Low in cost</li> <li>2. Compact, low floor space per square foot of membrane surface</li> <li>3. Low holdup per unit membrane area</li> </ol> | <ol style="list-style-type: none"> <li>1. Susceptible to particle plugging</li> <li>2. Badly fouled membranes difficult to clean</li> </ol>                                                                    |
| Hollow-Fiber                         | <ol style="list-style-type: none"> <li>1. Compact</li> <li>2. Inexpensive</li> </ol>                                                                                                    | <ol style="list-style-type: none"> <li>1. Very susceptible to particle plugging</li> <li>2. Badly fouled membrane modules nearly impossible to clean</li> </ol>                                                |

Table 3 - GENERAL SPECIFICATION FOR RO/UF MEMBRANE MODULES

|                                           | PERMEATION RATES (gal/hr) |       |         |       |       |
|-------------------------------------------|---------------------------|-------|---------|-------|-------|
|                                           | 200 psi                   |       | 400 psi |       |       |
|                                           | "0"                       | "50"  | "89"    | "92"  | "97"  |
| OSMO-52                                   | 9.4                       | 4.7   | 5.0     | 4.2   | 2.8   |
| OSMO-112                                  | 21                        | 10.8  | 10.8    | 9.0   | 6.0   |
| OSMO-192                                  | 31                        | 15.8  | 15.8    | 13.1  | 8.8   |
| OSMO-334                                  | 78                        | 35    | 35      | 28    | 19    |
| OSMO-554                                  | 134                       | 60    | 60      | 48    | 33    |
| NaCl %<br>Rejection                       | 0-10                      | 40-70 | 85-90   | 90-94 | 94-98 |
| Organic Cut-<br>Off Molecu-<br>lar Weight | 1000                      | 600   | 400     | 300   | 200   |

NOTES

1. Permeation rates assume 1000 ppm NaCl feed, 77°F at 10% recovery. 0 and 50 modules at 200 psig; 89, 92, and 97 modules at 400 psig.
2. Permeation rates are proportional to pressure.
3. Suggested operating temperature is 32-85°F. Temperatures over 85°F will reduce membrane life.

The fact that the 0 membrane is fabricated of polysulfone, which is tolerant to solutions ranging in pH from 2.5 to 12.5 and is resistant to temperatures in excess of 180°F, makes this a potentially useful membrane. The maximum suggested operating pressure is 200 psig. The manufacturer's suggested operating pressure is 100 psig, although polysulfone membranes tend to compact at pressures above 50 psig. The 0 membrane has an average pore size of 15 Å and a molecular weight cutoff of 1000 for organics. The 0 membrane is not effective in salt separation that is dependent on ionization. In the 0 membrane studies performed at Mound, there was a great deal of variation in the results, but this was expected because of the low salt rejection of the membrane and dependence on particle size for rejection (Table 4).

Both the 50 and 97 membranes are constructed of cellulose acetate and should be used only in the pH range from 3 to 6. The cellulose acetate membranes vary in maximum operating pressure yet, unlike the polysulfone membrane, the cellulose acetate membranes resist compaction at high pressure. The maximum suggested long-term operating pressure is 300 psig, and the recommended operating pressure is 200 psig for the 50 membrane. The 50 membrane has a molecular weight cutoff of 600 for organics and an average pore size of 11 Å. For the 97 membrane, the maximum suggested operating pressure is 800 psig, and the suggested operating pressure for low compaction is 400 psig. The 97 membrane has a molecular weight cutoff of 200 for organics and an average pore size of 5 Å.

The 50 membrane performed better than anticipated with the actinides and about as expected with the fission products. The 50 membrane demonstrated better than 95% rejections with actinides which are of higher valence and larger size than the fission products which averaged around 50% rejection (Table 4).

Table 4 - RESULTS OF RO MEMBRANE TESTS

| Isotope           | 0-Membrane<br>Rejection<br>(%) | 50-Membrane<br>Rejection<br>(%) | 97-Membrane<br>Rejection<br>(%) |
|-------------------|--------------------------------|---------------------------------|---------------------------------|
| <sup>239</sup> Pu | 64.0 ± 35.8                    | 99.4 ± 0.7                      | 98.9 ± 1.5                      |
| <sup>237</sup> Np | 35.5 ± 29.5                    | 96.8 ± 1.7                      | 99.0 ± 1.0                      |
| <sup>241</sup> Am | 21.8 ± 28.3                    | 98.6 ± 1.2                      | 98.9 ± 0.1                      |
| <sup>233</sup> U  | 14.3 ± 1.4                     | 98.8 ± 0.1                      | 99.1 ± 0.2                      |
| <sup>137</sup> Cs | 1.1 ± 2.5                      | 42.2 ± 1.5                      | 97.1 ± 0.2                      |
| <sup>125</sup> I  | 2.9 ± 0.6                      | 49.8 ± 1.9                      | 92.9 ± 0.3                      |
| <sup>60</sup> Co  | 4.9 ± 4.7                      | 95.4 ± 0.1                      | 97.5 ± 0.9                      |



The 97 membrane demonstrated the best performance of the membranes tested. The 97 membrane had high rejections with all the isotopes tested (Table 4).

Both the 50 and 97 membranes demonstrated excellent potential for the volume reduction of actinide-contaminated waste streams, but the 97 membrane should be used for fission products.

## COST ESTIMATES

Table 5 lists some typical capital costs of reverse osmosis systems operating at 90% recovery. Figure 4 is a graph of the capital cost as a function of the permeation rate. The operating costs range from about \$0.75 to \$2.00 per 1000 gal of water; 10% of the cost is for electricity, 75% for membrane replacement, 5% for chemicals, and 10% for maintenance.

The operating cost of reverse osmosis remains constant up to a feed-water concentration of about 2000 ppm TDS and increases only slightly above 2000 ppm. The operating cost per 1000 gal decreases as unit permeation flow rates increase. The capital cost of a membrane system is increased by 10% if chlorine levels in the feed stream are greater than 6000 ppm, because the 304 stainless steel in the RO unit must be replaced by 316 stainless steel.

To obtain a recovery of greater than 95% with solutions high in salts, a two-stage system is necessary. This would increase the capital cost by 50%. Recovery can be converted to volume reduction as follows:

| <u>Recovery (%)</u> | <u>Volume Reduction</u> |
|---------------------|-------------------------|
| 67                  | 3X                      |
| 90                  | 10X                     |
| 97                  | 30X                     |
| 98                  | 50X                     |
| 99                  | 100X                    |

The primary purpose of incorporating reverse osmosis as a means of decontaminating radioactive streams is to produce a purified stream concurrent with reducing the volume of radioactively contaminated water. This approach assumes necessary volume reductions of 10 to 400, or water recoveries in excess of 90%. This level of water recovery is possible only if the primary constituents within the stream do not cause scaling, the inherent osmotic pressure does not become excessive, and proper design restrictions are met.

Table 5 - REVERSE OSMOSIS SYSTEMS WITH 97 MEMBRANE AT 90% RECOVERY

| Membrane<br>area (ft <sup>2</sup> ) | 2940      | 4200      | 5880      | 7140      | 8820      | 10,800    |
|-------------------------------------|-----------|-----------|-----------|-----------|-----------|-----------|
| Permeation rate<br>(gal/min)        | 20        | 30        | 40        | 50        | 60        | 70        |
| (gal/day) <sup>a</sup>              | 9,600     | 14,400    | 19,200    | 24,000    | 28,000    | 33,600    |
| Net Capital Cost <sup>b</sup>       | \$44,600  | \$50,800  | \$63,660  | \$69,290  | \$78,000  | \$86,500  |
| Approximate Size<br>(ft)            | 18Lx6Hx2W | 21Lx6Hx3W | 21Lx6Hx4W | 21Lx6Hx5W | 21Lx6Hx5W | 21Lx6Hx6W |

<sup>a</sup>Gallons per day are based on an 8-hr day.

<sup>b</sup>Add 10% for waste streams with chloride concentrations above 6000 ppm,  
as 304 stainless steel must be replaced with 316; and 50% for two-stage  
system for 99% recovery.

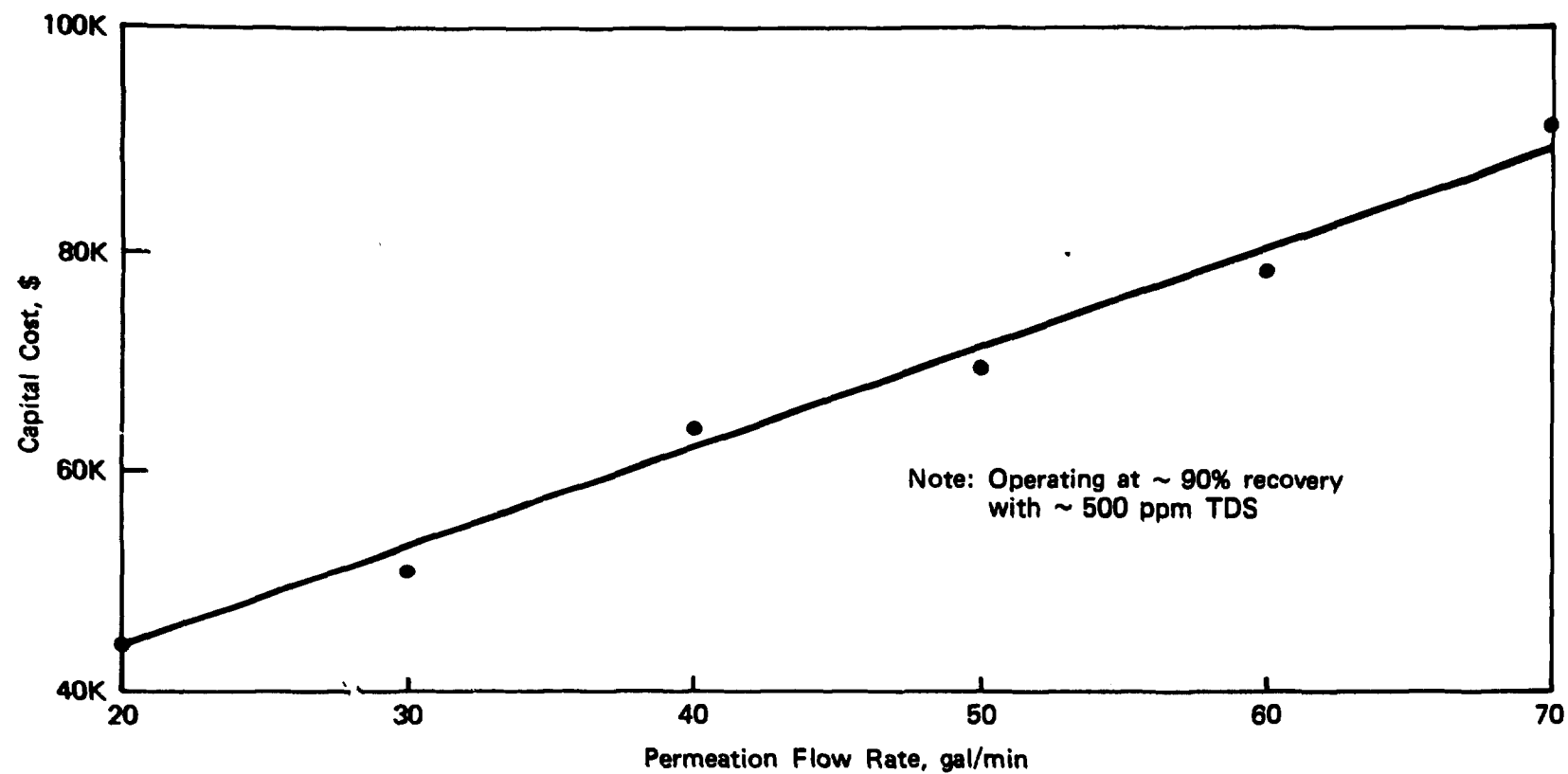


FIGURE 4 - Approximate capital cost as a function of permeation flow rate for reverse osmosis.

## REVERSE OSMOSIS PLANT DESIGN

A reverse osmosis system designed for continuous operation and producing 40 gal/min incorporates 5775 ft<sup>2</sup> of 97 cellulose acetate membrane operated at 400 psig (Figure 5). This system is rated for feed water with a constant inlet pressure between 30 to 60 psig at 77°F. Chemical feed equipment to maintain the feed solution pH at  $6.0 \pm 0.5$  is included. The purpose of the chemical feed equipment is to provide optimum life of the membrane and to reduce the possibility of precipitation of calcium and magnesium salts. Pure water recovery of the feed water can be set at 90 to 95%. Feedwater must be below 85°F and a maximum continuous water temperature below 80°F is suggested. If higher temperatures are used, it should be realized that the permeation rate will increase, but the economic life of the membrane modules will decrease due to membrane compaction. Controls are supplied to keep the high pressure pump from running dry, to protect against low flow in the system, and to protect against high temperature in the system.

## ADSORBENT EVALUATIONS

During Mound's adsorbent work the adsorption of a number of actinides and fission products has been studied. Specifically, Mound has endeavored to determine which adsorbents exhibit the highest decontamination factors for each radioactive element investigated. Those adsorbents lacking sufficient capacity to be economically viable were excluded from study. The results of this work are succinctly summarized in Table 6. Of these radioisotopes, cobalt was investigated this year.

The investigation of cobalt involved a literature search for the most promising adsorbents and two sets of experiments. The first set, which was called the main effects design experiment, was designed to determine which solution characteristics, or combinations of characteristics, most affected cobalt adsorption into the previously selected adsorbents. Eight solution variables were chosen as being the most likely to affect adsorption. These were: solution acidity, and concentrations of nickel, calcium, magnesium, ammonium, carbonate, phosphate, and sulfite ions. Four ion exchange resins were used as adsorbents: IRC-718, MSC-1, HCR-2W-H, and AG50WX8. The results of this experiment showed that hydroxide (solution pH), sulfite, and ammonium ion concentrations have an effect on cobalt adsorption.

The second set of experiments, the interaction design experiment, was performed to determine the magnitudes of the effects of the three variables and the magnitudes of the interactions of the three variables with respect to the removal of cobalt-60 from aqueous solutions by the selected cationic ion exchange resins (MSC-1, HCR-2W-H, and AG50WX8). The interaction design consisted of nine solution compositions which represented the corners and the centerpoint of a cube formed by the

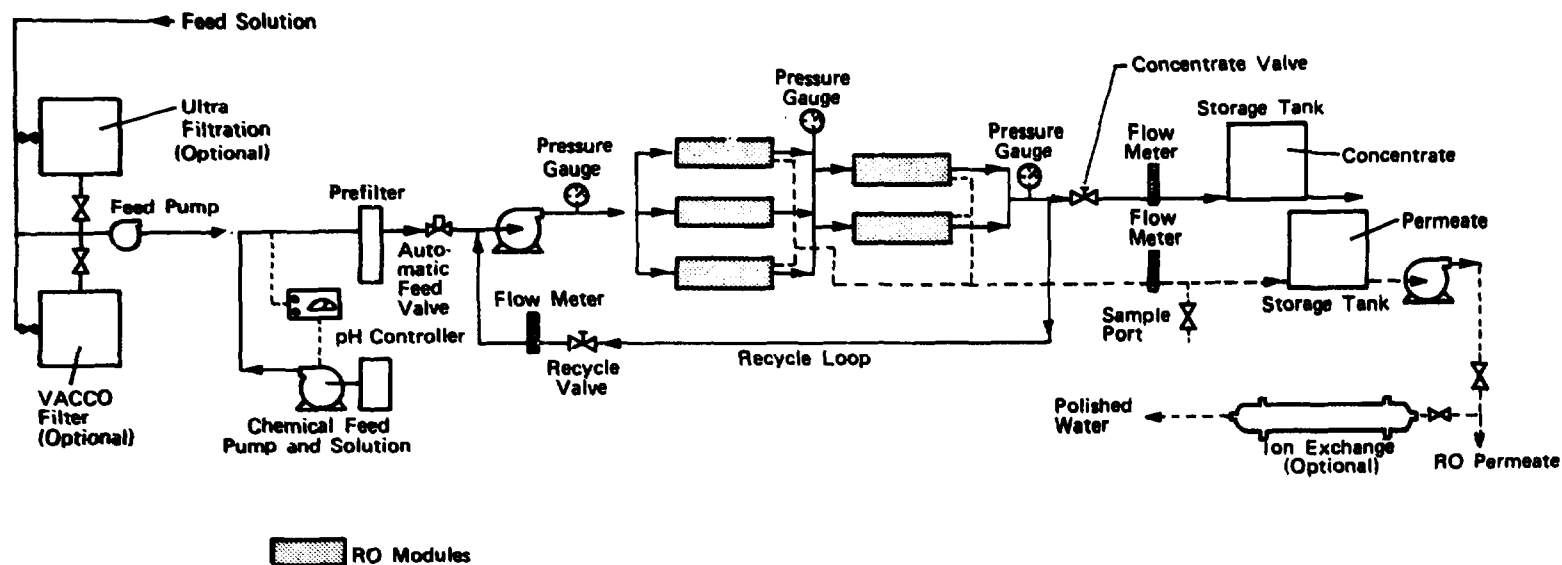


FIGURE 5 - Reverse osmosis system.

Table 6 - ION EXCHANGE/ADSORBENT TREATMENT SPECIFICATION

| Radioactive<br>Specie | Recommended<br>Cation Resin | Recommended<br>Anion Resin | Other Resins            |                      | Other<br>Adsorbents    | Recommended<br>pH Range |
|-----------------------|-----------------------------|----------------------------|-------------------------|----------------------|------------------------|-------------------------|
|                       |                             |                            | Cation                  | Anion                |                        |                         |
| $^{238}\text{Pu}$     | MSC-1                       | IRA938                     | 200,AG50WX8             | MSA-1                | Bonechar               | 5 to 8                  |
| $^{233}\text{U}$      | MSC-1                       | IRA938                     | 200,AG50WX8             | MSA-1                | Bonechar               | 5 to 8                  |
| $^{237}\text{Np}$     | MSC-1                       | IRA938                     | 200,AG50WX8             | MSA-1                | Bonechar               | 5 to 8                  |
| $^{241}\text{Am}$     | MSC-1                       | IRA938                     | 200,AG50WX8             | MSA-1                | Bonechar               | 5 to 8                  |
| $^{137}\text{Cs}$     | MSC-1                       | None                       | 200,AG50WX8<br>HCR2W-H  | None                 | Duracil,<br>White Sand | 5 to 8                  |
| $^{60}\text{Co}$      | MSC-1                       | IRA938                     | HCR2W-H, 200<br>AG50WX8 | MSA-1, SAR<br>IRA430 |                        | 3 to 6                  |
| $^{131}\text{I}$      | None                        | IRA938                     | None                    | SAR, MSA-1<br>IRA430 |                        | 6 to 8                  |

variables on three-dimensional cartesian axes (See Figure 6). The determined hydroxide ion concentration limits were: low limit, pH 3; midpoint, pH 6.5; and high limit, pH 10. The limits for sulfite and ammonium ions were: low limit, 1 ppm; midpoint, 50 ppm; and high limit, 100 ppm.

The experimental design (Figure 7) determined the solution composition for each run according to the convention of plus being the high variable limit, minus being the lower limit, and zero being the midpoint. For statistical determination of the variance, the centerpoint was replicated three times with two replicates on each of the eight corners.

Once analyzed and formed into a prediction equation, the data should give a good representation of how those three variables interact to produce an effect at any position inside or on the surface of the cube. Areas outside the cube cannot be predicted. Quite often the equation produced by this simple design will accurately describe the effects caused by the variables. Sometimes, however, a more complex model is needed.

## RESULTS

In general, all three cation exchange resins tested showed >99% removal of cobalt on the pH 3 runs and the centerpoint where the pH was 6.5. The pH 10 runs, however, varied from 92% cobalt removed to 9% cobalt removed. These results were expected. From Figure 8 it can be seen that in the pH range of 3 to 7 almost none of the ammonium ion present has been reacted to ammonia. And because of the large differences in the stability of cobalt (II) complexes and cobalt (III) complexes, and the inverse difference in the stability of cobalt (II) simple salts and cobalt (III) simple salts, at pH 3 and 6.5 all or >99% of the cobalt present will be in the form of cobalt (II) positive ions. These are easily removed by ion exchange. At pH 10, however, of 100 ppm  $\text{NH}_4^+$  put into the solution, only about 15% will be in that form, and 85% will be in the form of  $\text{NH}_3$ . With  $\text{NH}_3$  present, the formation of cobalt (III) complexes is favored over cobalt (II) simple salts. The large difference between the amounts of cobalt-60 removed by the various pH 10 runs stems from the amounts of  $\text{NH}_4^+$  and  $\text{SO}_3^{2-}$  placed originally in those runs. Four combinations of ammonium and sulfite were run at pH 10 on each of the resins. Each combination was run twice.

The pH 10 runs, their levels of ammonium and sulfite, and the results ( $\bar{y}$ ) are listed in Table 7. As can be seen from Table 7, when both  $[\text{NH}_4^+]$  and  $[\text{SO}_3^{2-}]$  were at the 1 ppm level, about 90% of the cobalt-60 was removed. This means that about 10% of the cobalt-60 present was in the form of cobalt (III) complexes which were either negative or neutral. When 1 ppm  $\text{NH}_4^+$  and 100 ppm  $\text{SO}_3^{2-}$  were present, about 63% of the cobalt-60 was removed. This is roughly 30% less cobalt-60 removed than when the  $\text{SO}_3^{2-}$  concentration was only 1 ppm. This result probably means that when the concentrations of  $\text{NH}_4^+$  and  $\text{SO}_3^{2-}$  were both 1 ppm, about 40% of the cobalt was in the form of cobalt (III) complexes, but only about 10% were

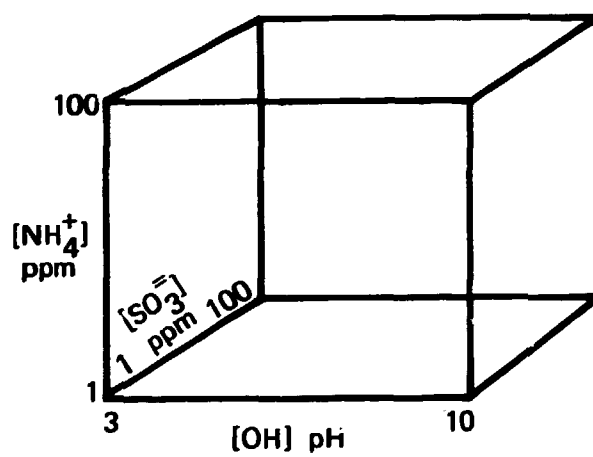


FIGURE 6 - Cubic space formed by the three variables of the cobalt-60 interaction design.



| Trial | Run #   | Y | X <sub>1</sub><br>[OH <sup>-</sup> ] | X <sub>2</sub><br>[NH <sub>4</sub> <sup>+</sup> ] | X <sub>3</sub><br>[SO <sub>3</sub> <sup>-</sup> ] | X <sub>1</sub> X <sub>2</sub> | X <sub>1</sub> X <sub>3</sub> | X <sub>2</sub> X <sub>3</sub> | Y's | $\bar{Y}$ | (Y- $\bar{Y}$ ) | (Y- $\bar{Y}$ ) <sup>2</sup> | S <sup>2</sup> | df<br>r-1 | Y <sub>PRED</sub> | $\tau = \frac{Y_{PRED} - Y_{Ob}}{S}$ |
|-------|---------|---|--------------------------------------|---------------------------------------------------|---------------------------------------------------|-------------------------------|-------------------------------|-------------------------------|-----|-----------|-----------------|------------------------------|----------------|-----------|-------------------|--------------------------------------|
| CP    | 1       |   | o                                    | o                                                 | o                                                 | o                             | o                             | o                             |     |           |                 |                              |                |           |                   |                                      |
| CP    | 10      |   | o                                    | o                                                 | o                                                 | o                             | o                             | o                             |     |           |                 |                              |                |           |                   |                                      |
| CP    | 19      |   | o                                    | o                                                 | o                                                 | o                             | o                             | o                             |     |           |                 |                              |                |           |                   |                                      |
| 1     | 5<br>16 |   | -                                    | -                                                 | -                                                 | +                             | +                             | +                             |     |           |                 |                              |                |           |                   |                                      |
| 2     | 6<br>14 |   | +                                    | -                                                 | -                                                 | -                             | -                             | +                             |     |           |                 |                              |                |           |                   |                                      |
| 3     | 7<br>11 |   | -                                    | +                                                 | -                                                 | -                             | +                             | -                             |     |           |                 |                              |                |           |                   |                                      |
| 4     | 4<br>17 |   | +                                    | +                                                 | -                                                 | +                             | -                             | -                             |     |           |                 |                              |                |           |                   |                                      |
| 5     | 8<br>13 |   | -                                    | -                                                 | +                                                 | +                             | -                             | -                             |     |           |                 |                              |                |           |                   |                                      |
| 6     | 3<br>15 |   | +                                    | -                                                 | +                                                 | -                             | +                             | -                             |     |           |                 |                              |                |           |                   |                                      |
| 7     | 2<br>18 |   | -                                    | +                                                 | +                                                 | -                             | -                             | +                             |     |           |                 |                              |                |           |                   |                                      |
| 8     | 9<br>12 |   | +                                    | +                                                 | +                                                 | +                             | +                             | +                             |     |           |                 |                              |                |           |                   |                                      |

$$\Sigma X\bar{Y} =$$

$$\Sigma X\bar{Y}/(T/2) =$$

$$\frac{2\Sigma X\bar{Y}/T}{2} = b =$$

FIGURE 7 - Cobalt-60 interaction design.

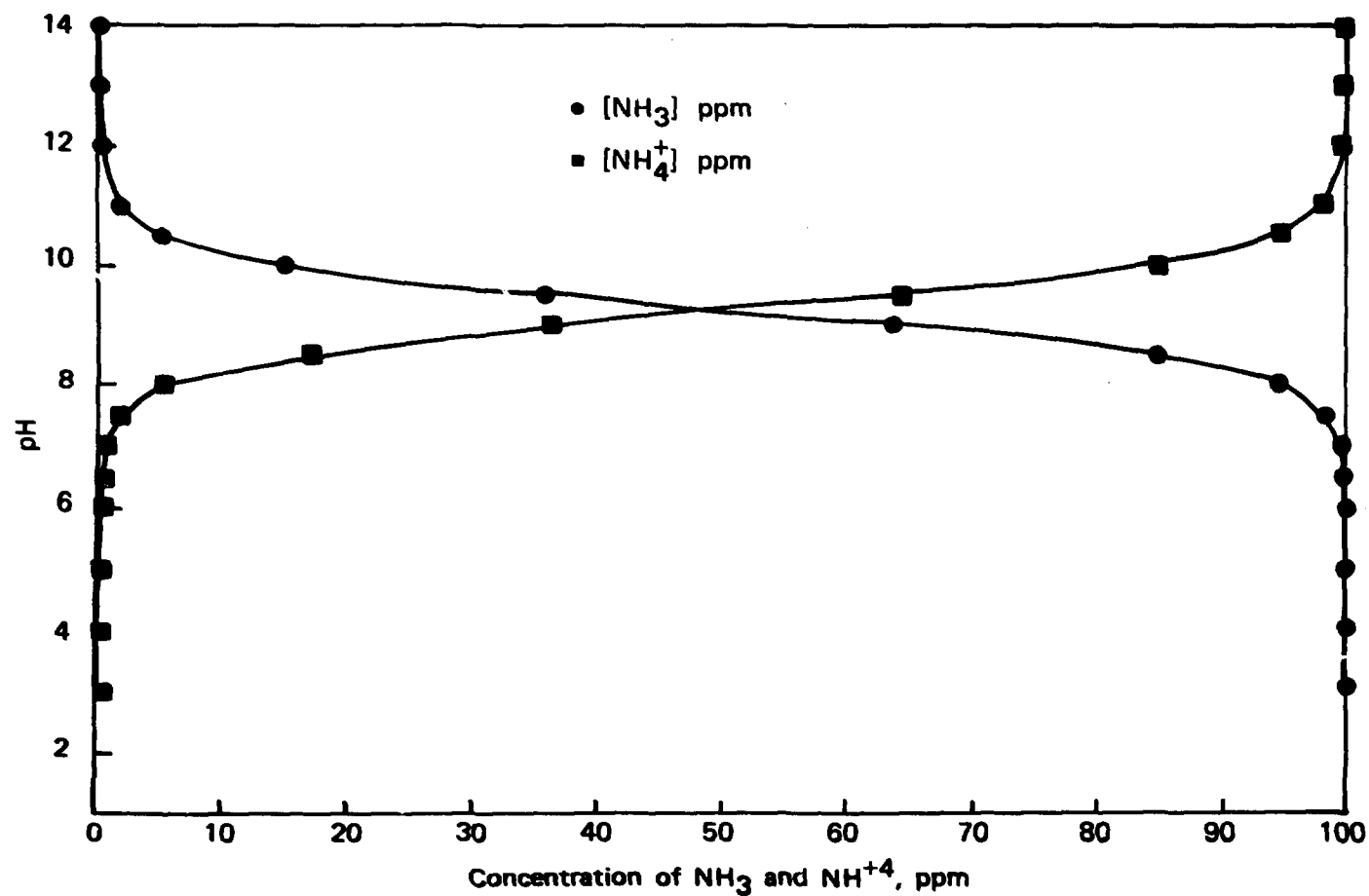


FIGURE 8 - Concentration of  $\text{NH}_4^+$  as a function of pH in the reaction  $\text{NH}_4^+ + \text{OH}^- \rightleftharpoons \text{NH}_3 + \text{H}_2\text{O}$ , calculated using  $K = \frac{[\text{NH}_3][\text{H}^+]}{[\text{NH}_4^+]} = 5.6885 \times 10^{-10}$  starting with 100 ppm  $\text{NH}_4^+$  and 0 ppm  $\text{NH}_3$ .

Table 7 - pH 10 RUNS

| Resin    | Run No. | $\text{NH}_4^+$<br>(ppm) | $\text{SO}_3^-$<br>(ppm) | $\bar{Y}$<br>% $\text{Co}_{60}$<br>Removed |
|----------|---------|--------------------------|--------------------------|--------------------------------------------|
| AG50WX8  | 6/14    | 1                        | 1                        | 89.6                                       |
|          | 3/15    | 1                        | 100                      | 63.9                                       |
|          | 4/17    | 100                      | 1                        | 47.6                                       |
|          | 9/12    | 100                      | 100                      | 20.7                                       |
| MSC-1    | 6/14    | 1                        | 1                        | 90.0                                       |
|          | 3/15    | 1                        | 100                      | 63.1                                       |
|          | 4/17    | 100                      | 1                        | 44.9                                       |
|          | 9/12    | 100                      | 100                      | 20.5                                       |
| HCR-2W-H | 6/14    | 1                        | 1                        | 91.5                                       |
|          | 3/15    | 1                        | 100                      | 61.9                                       |
|          | 4/17    | 100                      | 1                        | 41.5                                       |
|          | 9/12    | 100                      | 100                      | 13.1                                       |

negative or neutral. The others were complexes such as  $[\text{Co}(\text{NH}_3)_6]^{+3}$ ,  $[\text{Co}(\text{NH}_3)_5\text{H}_2\text{O}]^{+3}$ ,  $[\text{Co}(\text{NH}_3)_4(\text{H}_2\text{O})_2]^{+3}$ ,  $[\text{Co}(\text{NH}_3)_5\text{SO}_3]^{+1}$ ,  $[\text{Co}(\text{NH}_3)_5\text{OH}]^{+2}$ ,  $[\text{Co}(\text{NH}_3)_4(\text{H}_2\text{O})(\text{OH})]^{+2}$ , and  $[\text{Co}(\text{NH}_3)_4(\text{H}_2\text{O})\text{SO}_3]^{+1}$ . With the additional  $\text{SO}_3^-$  present in runs 3 and 15, most of the cobalt (III) present was probably in neutral or negative form, such as  $[\text{Co}(\text{NH}_3)_4(\text{SO}_3)_2]^-$  and  $[\text{Co}(\text{NH}_3)_4(\text{SO}_3)(\text{OH})]^0$ . One might think that 1 ppm  $\text{NH}_4^+$  is a very small amount, when up to six  $\text{NH}_3$  molecules can complex with one cobalt atom. However, at 10,000 counts/min/m<sup>2</sup>, the concentration of cobalt-60 is approximately 0.000004 ppm. Therefore, chemically, 1 ppm  $\text{NH}_4^+$  is a large excess. Again in Table 7, runs 4/17, the concentration of  $\text{NH}_4^+$  is 100 ppm, and the concentration of  $\text{SO}_3^-$  is 1 ppm. This combination averaged roughly 45% cobalt-60 removed. This is a further decrease from runs 3/15. In this case, about 55% of the cobalt was in the form of neutral or negative cobalt (III) complexes. From the much lower results on runs 9/12, it is believed that roughly 80% of the cobalt in runs 4/17 was in the form of cobalt (III) complexes; however, about 25% were positively charged. In runs 9/12, roughly 20% of the cobalt-60 was removed, so roughly 80% of the cobalt was in the form of cobalt (III) neutral or negative complexes. It is believed that most of the cobalt would eventually go to cobalt (III) in these solutions with  $\text{NH}_3$  present, but either more time or more molecular oxygen would be needed for the oxidation to occur.

The experimental design, results, and analyses for the resins AG50WX8, MSC-1, and HCR2W-H are shown in Figures 9, 10, and 11, respectively. All three resins behaved basically the same with only slight differences in

| Trial | Run #   | Y | X <sub>1</sub><br>[OH <sup>-</sup> ] | X <sub>2</sub><br>[NH <sub>4</sub> <sup>+</sup> ] | X <sub>3</sub><br>[SO <sub>3</sub> <sup>=</sup> ] | X <sub>1</sub> X <sub>2</sub> | X <sub>1</sub> X <sub>3</sub> | X <sub>2</sub> X <sub>3</sub> | Y's          | $\bar{Y}$ | (Y- $\bar{Y}$ ) | (Y- $\bar{Y}$ ) <sup>2</sup> | S <sup>2</sup><br>SS<br>df | df<br>r-1 | Y <sub>PRED</sub> | T=<br>Y <sub>PRED</sub><br>-Y <sub>Ob</sub> | X <sub>1</sub> X <sub>2</sub> X <sub>3</sub> |
|-------|---------|---|--------------------------------------|---------------------------------------------------|---------------------------------------------------|-------------------------------|-------------------------------|-------------------------------|--------------|-----------|-----------------|------------------------------|----------------------------|-----------|-------------------|---------------------------------------------|----------------------------------------------|
| CP    | 1       |   | o                                    | o                                                 | o                                                 | o                             | o                             | o                             | 99.7         |           | 0.2             | 0.04                         |                            |           |                   |                                             |                                              |
| CP    | 10      |   | o                                    | o                                                 | o                                                 | o                             | o                             | o                             | 99.6         | 99.5      | 0.1             | 0.01                         | 0.07                       | 2         |                   |                                             |                                              |
| CP    | 19      |   | o                                    | o                                                 | o                                                 | o                             | o                             | o                             | 99.2         |           | -0.3            | 0.09                         |                            |           |                   |                                             |                                              |
| 1     | 5<br>16 |   | -                                    | -                                                 | -                                                 | +                             | +                             | +                             | 99.8<br>99.5 | 99.7      | 0.10<br>-0.20   | 0.01<br>0.04                 | 0.05                       | 1         |                   |                                             | -                                            |
| 2     | 6<br>14 |   | +                                    | -                                                 | -                                                 | -                             | -                             | +                             | 88.1<br>91.0 | 89.6      | -1.50<br>1.40   | 2.25<br>1.96                 | 4.21                       | 1         |                   |                                             | +                                            |
| 3     | 7<br>11 |   | -                                    | +                                                 | -                                                 | -                             | +                             | -                             | 99.8<br>99.9 | 99.9      | -0.1<br>0.0     | 0.01<br>0.0                  | 0.01                       | 1         |                   |                                             | +                                            |
| 4     | 4<br>17 |   | +                                    | +                                                 | -                                                 | +                             | -                             | -                             | 40.9<br>54.2 | 47.6      | -6.70<br>6.60   | 44.89<br>43.56               | 88.45                      | 1         |                   |                                             | -                                            |
| 5     | 8<br>13 |   | -                                    | -                                                 | +                                                 | +                             | -                             | -                             | 90.8<br>99.7 | 99.8      | 0.0<br>-0.1     | 0.0<br>0.01                  | 0.01                       | 1         |                   |                                             | +                                            |
| 6     | 3<br>15 |   | +                                    | -                                                 | +                                                 | -                             | +                             | -                             | 59.5<br>68.2 | 63.9      | -4.40<br>4.30   | 19.36<br>18.49               | 37.85                      | 1         |                   |                                             | -                                            |
| 7     | 2<br>18 |   | -                                    | +                                                 | +                                                 | -                             | -                             | +                             | 99.7<br>99.7 | 99.7      | 0.0<br>0.0      | 0.0<br>0.0                   | 0.0                        | 1         |                   |                                             | -                                            |
| 8     | 9<br>12 |   | +                                    | +                                                 | +                                                 | +                             | +                             | +                             | 25.7<br>15.6 | 20.7      | 5.0<br>-5.1     | 25.0<br>26.01                | 51.0                       | 1         |                   |                                             | +                                            |

|                                    |        |       |       |       |       |       |                                             |                                       |       |
|------------------------------------|--------|-------|-------|-------|-------|-------|---------------------------------------------|---------------------------------------|-------|
| $\Sigma X\bar{Y} =$                | -177.3 | -85.1 | -52.7 | -85.3 | -52.5 | -1.5  | $b_0 = \text{Avg } \bar{Y} \text{ corners}$ | df=10                                 | -0.90 |
| $\Sigma X\bar{Y}/(T/2) =$          | -44.3  | -21.3 | -13.2 | -21.3 | -13.1 | -0.38 | $b_0 = 77.61$                               | $S^2_{\text{MDF}}=181.7$              | -0.23 |
| $\frac{2\Sigma X\bar{Y}}{T} = b =$ | -22.2  | -10.6 | -6.6  | -10.7 | -6.6  | -0.19 |                                             | $S_p^2 = \frac{181.7}{10} = 18.2$     | -0.11 |
|                                    |        |       |       |       |       |       |                                             | $S_p = 4.26, 10 \text{ df } t = 2.23$ |       |

$$Y = b_0 + b_1x_1 + b_2x_2 + b_3x_3 + b_{1,2}x_1x_2 + b_{1,3}x_1x_3 + b_{2,3}x_2x_3 + b_{1,2,3}x_1x_2x_3$$

$$Y = 77.6 - 22.2x_1 - 10.6x_2 - 6.6x_3 - 10.7x_1x_2 - 6.6x_1x_3$$

$$\text{Physical Units } x_1 = \frac{x_1 - 6.5}{3.5}; x_2 = \frac{x_2 - 50.5}{49.5}; x_3 = \frac{x_3 - 50.5}{49.5}$$

$$b \text{ conf lim} = \pm t S_p \sqrt{\frac{1}{T}}$$

$$b_{CL} = \pm 2.37 \quad \text{Model Fit} = 21.9 \pm 6.43$$

$$\bar{Y}_{CL} = \pm 6.72$$

FIGURE 9 - Experimental design, results, and analysis for AG50WX8.

| Trial # | Run # | $X_1$ | $X_2$ | $X_3$ | $X_1X_2$ | $X_1X_3$ | $X_2X_3$ | $Y$  | $Y's$ | $\bar{Y}$ | $(Y-\bar{Y})$ | $(Y-\bar{Y})^2$ | $S^2$  | df | $Y_{PRED}$ | $r = \frac{Y_{PRED} - Y_{Obs}}{X_1X_2X_3}$ |
|---------|-------|-------|-------|-------|----------|----------|----------|------|-------|-----------|---------------|-----------------|--------|----|------------|--------------------------------------------|
| CP      | 1     | 0     | 0     | 0     | 0        | 0        | 0        | 95.7 | 95.7  | 98.2      | -2.5          | 6.25            |        |    |            |                                            |
| CP      | 10    | 0     | 0     | 0     | 0        | 0        | 0        | 99.6 | 99.6  | 98.2      | 1.4           | 1.96            | 4.61   | 2  |            |                                            |
| CP      | 19    | 0     | 0     | 0     | 0        | 0        | 0        | 99.2 | 99.2  | 98.2      | 1.0           | 1.0             |        |    |            |                                            |
| 1       | 5     | -     | -     | -     | +        | +        | +        | 99.5 | 99.5  | 99.3      | 0.2           | 0.04            | 0.08   | 1  |            | -                                          |
| 16      | 16    | -     | -     | -     | +        | +        | +        | 99.1 | 99.1  | 99.3      | -0.2          | 0.04            |        |    |            |                                            |
| 2       | 6     | +     | -     | -     | -        | -        | +        | 90.0 | 90.0  | 90.0      | 0             | 0               | 0.00   | 1  |            | +                                          |
| 14      | 14    | +     | -     | -     | -        | -        | +        | 90.0 | 90.0  | 90.0      | 0             | 0               |        |    |            |                                            |
| 3       | 7     | -     | +     | -     | -        | +        | -        | 99.8 | 99.8  | 99.9      | -0.05         | 0.003           | 0.006  | 1  |            | +                                          |
| 11      | 11    | -     | +     | -     | -        | +        | -        | 99.9 | 99.9  | 99.9      | 0.05          | 0.003           |        |    |            |                                            |
| 4       | 4     | +     | +     | -     | +        | -        | -        | 35.6 | 35.6  | 44.9      | -9.3          | 86.5            | 172.98 | 1  |            | -                                          |
| 17      | 17    | +     | +     | -     | +        | -        | -        | 54.2 | 54.2  | 44.9      | 9.3           | 86.5            |        |    |            |                                            |
| 5       | 8     | -     | -     | +     | +        | -        | -        | 99.8 | 99.8  | 99.7      | 0.10          | 0.01            | 0.02   | 1  |            | +                                          |
| 13      | 13    | -     | -     | +     | +        | -        | -        | 99.6 | 99.6  | 99.7      | -0.10         | 0.01            |        |    |            |                                            |
| 6       | 3     | +     | -     | +     | -        | +        | -        | 68.9 | 68.9  | 63.1      | 5.8           | 33.64           | 68.45  | 1  |            | -                                          |
| 15      | 15    | +     | -     | +     | -        | +        | -        | 57.2 | 57.2  | 63.1      | -5.9          | 34.81           |        |    |            |                                            |
| 7       | 2     | -     | +     | +     | -        | -        | +        | 98.9 | 98.9  | 99.3      | -0.40         | 0.16            | 0.25   | 1  |            | -                                          |
| 18      | 18    | -     | +     | +     | -        | -        | +        | 99.6 | 99.6  | 99.3      | 0.30          | 0.09            |        |    |            |                                            |
| 8       | 9     | +     | +     | +     | +        | +        | +        | 28.7 | 28.7  | 20.5      | 8.2           | 67.24           | 136.13 | 1  |            |                                            |
| 12      | 12    | +     | +     | +     | +        | +        | +        | 12.2 | 12.2  | 20.5      | -8.3          | 68.89           |        |    |            |                                            |

|                                                                   |        |        |        |        |        |      |               |  |  |  |  |  |  |  |  |      |
|-------------------------------------------------------------------|--------|--------|--------|--------|--------|------|---------------|--|--|--|--|--|--|--|--|------|
| $\sum XY =$                                                       | -179.7 | -87.5  | -51.5  | -87.90 | -51.10 | 1.5  | $b_0 = 77.09$ |  |  |  |  |  |  |  |  | 3.50 |
| $\sum X^2/(n/2) =$                                                | -44.93 | -21.88 | -12.88 | -21.98 | -12.78 | 0.38 |               |  |  |  |  |  |  |  |  | 0.88 |
| $\frac{2\sum XY}{n} = b =$                                        | -22.5  | -10.9  | -6.4   | -11.0  | -6.4   | 0.19 |               |  |  |  |  |  |  |  |  | 0.44 |
| $Y = 77.09 - 22.5x_1 - 10.9x_2 - 6.4x_3 - 11.0x_1x_2 - 6.4x_1x_3$ |        |        |        |        |        |      |               |  |  |  |  |  |  |  |  |      |

|                              |  |  |  |  |  |  |  |  |  |  |  |  |  |  |  |  |
|------------------------------|--|--|--|--|--|--|--|--|--|--|--|--|--|--|--|--|
| $\sum df = 10$               |  |  |  |  |  |  |  |  |  |  |  |  |  |  |  |  |
| $\sum S^2 xdf = 387.1$       |  |  |  |  |  |  |  |  |  |  |  |  |  |  |  |  |
| $S^2 = 38.71$                |  |  |  |  |  |  |  |  |  |  |  |  |  |  |  |  |
| $S_p = 6.22$                 |  |  |  |  |  |  |  |  |  |  |  |  |  |  |  |  |
| $t = 2.23$                   |  |  |  |  |  |  |  |  |  |  |  |  |  |  |  |  |
| $bcL = \pm 3.47$             |  |  |  |  |  |  |  |  |  |  |  |  |  |  |  |  |
| $YcL = \pm 9.81$             |  |  |  |  |  |  |  |  |  |  |  |  |  |  |  |  |
| Model fit = -21.1 $\pm$ 9.39 |  |  |  |  |  |  |  |  |  |  |  |  |  |  |  |  |

FIGURE 10 - Experimental design, results, and analysis for MSC-1.

| Trial | Run # | $X_1$ | $X_2$ | $X_3$ | $X_1X_2$ | $X_1X_3$ | $X_2X_3$ | $Y$  | $Y's$ | $\bar{Y}$ | $(Y-\bar{Y})$ | $(Y-\bar{Y})^2$ | $S^2$ | $df$ | $Y_{PRED}$ | $t = \frac{Y_{PRED} - Y_{Ob}}{S_{Y_{PRED}}}$ |
|-------|-------|-------|-------|-------|----------|----------|----------|------|-------|-----------|---------------|-----------------|-------|------|------------|----------------------------------------------|
| CP    | 1     | 0     | 0     | 0     | 0        | 0        | 0        | 99.8 | 99.8  | 0.30      | 0.09          | 0.09            | 0.09  | 2    |            |                                              |
| CP    | 10    | 0     | 0     | 0     | 0        | 0        | 0        | 99.3 | 99.5  | -0.20     | 0.04          | 0.04            | 0.09  | 2    |            |                                              |
| CP    | 19    | 0     | 0     | 0     | 0        | 0        | 0        | 99.3 | 99.3  | -0.20     | 0.04          | 0.04            |       |      |            |                                              |
| 1     | 5     | -     | -     | -     | +        | +        | +        | 99.6 | 99.7  | -0.10     | 0.01          | 0.01            | 0.01  | 1    |            | -                                            |
| 2     | 6     | +     | -     | -     | -        | -        | +        | 90.9 | 91.5  | -0.6      | 0.36          | 0.36            | 0.61  | 1    |            | +                                            |
| 3     | 7     | -     | +     | -     | -        | +        | -        | 99.8 | 99.9  | 0.1       | 0.01          | 0.01            | 0.01  | 1    |            | +                                            |
| 4     | 4     | +     | +     | -     | +        | -        | -        | 38.8 | 41.5  | -2.7      | 7.29          | 7.29            | 14.05 | 1    |            | -                                            |
| 5     | 8     | -     | -     | +     | +        | -        | -        | 99.8 | 99.7  | 0.1       | 0.01          | 0.01            | 0.02  | 1    |            | +                                            |
| 6     | 3     | +     | -     | +     | -        | +        | -        | 68.6 | 61.9  | 6.7       | 44.89         | 44.89           | 91.13 | 1    |            | -                                            |
| 7     | 2     | -     | +     | +     | -        | -        | +        | 99.6 | 99.6  | 0.0       | 0.0           | 0.0             | 0.01  | 1    |            | -                                            |
| 8     | 12    | +     | +     | +     | +        | +        | +        | 17.1 | 13.1  | 4.0       | 16.00         | 16.00           | 32.81 | 1    |            | +                                            |

$$EX\bar{Y} = -190.9 \quad -98.7 \quad -58.3 \quad -98.9 \quad -57.7 \quad 0.9 \quad b_0 = 75.9$$

$$EX\bar{Y}/(T/2) = -47.7 \quad -24.7 \quad -14.5 \quad -24.7 \quad -14.4 \quad 0.23$$

$$\frac{2EX\bar{Y}/T}{2} = b = -23.9 \quad -12.4 \quad -7.3 \quad -12.4 \quad -7.2 \quad 0.11$$

$$Y = 75.9 - 23.9X_1 - 12.4X_2 - 7.3X_3 - 12.4X_1X_2 - 7.2X_1X_3$$

$$df=10$$

$$S^2_{x\bar{d}f} = 138.83$$

$$S^2_b = 13.88$$

$$S_p = 3.73 \quad t = 2.23$$

$$bCL = \pm 2.08$$

$$\bar{Y}CL = \pm 5.88$$

FIGURE 11 - Experimental design, results, and analysis for HCR2WH.

the actual values of the variable coefficients. The strongest single factor was the effect of hydroxide ion concentration, ammonium concentration was the next strongest, then sulfite concentration. Two interactions were found. Of the two interactions, the hydroxide ion concentration-ammonium ion concentration interaction was the stronger, and the hydroxide ion concentration-sulfite ion concentration interaction was the weaker. The ammonium ion concentration-sulfite ion concentration interaction was less than the response variation. The three-factor interaction,  $x_1x_2x_3$ , was also less than the response variation. In the order of decreasing effect, the variables can be listed as follows:  $[\text{OH}^-] > [\text{NH}_4^+] = [\text{OH}^-][\text{NH}_4^+] > [\text{SO}_3^{2-}] = [\text{OH}^-][\text{SO}_3^{2-}]$ .

The first generation prediction equations are listed in Table 8. As the table indicates, these equations do not predict well at the centerpoint. At pH 6.5, the centerpoint (see Figure 3), only a fraction of a percent (0.18%) of the  $\text{NH}_4^+$  present is in the form of  $\text{NH}_3$ . Since so little  $\text{NH}_3$  is present, almost all the cobalt will be removed as cobalt (II) ions. This nonlinearity of response eliminates a linear equation as a good model. Since the first model was not adequate to describe the response over the entire cubic region described by the variables, more data were necessary. Three more points were selected, to be done with one replicate each, necessitating performing six more runs. The points chosen were from the centerpoint to the center of the ammonium ion concentration-sulfite ion concentration plane. These points held the ammonium ion and sulfite ion concentrations constant and increased the hydroxide ion concentration from pH 6.5 at the centerpoint to pH 10 on the face of the  $[\text{NH}_4^+][\text{SO}_3^{2-}]$  plane (See Figure 12).

These points, written as three dimensional cartesian coordinates of the form  $[\text{OH}^-]$ ,  $[\text{NH}_4^+]$ ,  $[\text{SO}_3^{2-}]$ , are:

Point 1 (7.67, 50, 50)  
 Point 2 (8.83, 50, 50)  
 Point 3 (10.00, 50, 50)

The results of these experiments are given in Table 9.

All points of the interaction design and the three extra points were entered in a computer program to obtain prediction equations of the form  $Y = b + C_1X_1 + C_2X_2 + C_3X_3 + C_4X_1^2 + C_5X_1X_2 + C_6X_2^2 + C_7X_1X_3 + C_8X_2X_3 + C_9X_3^2$ . These equations presented in Table 10 are fairly good predictors of the percentage of cobalt removed at the center point and the corners of the cubic space defined by the variable limits. However, they are not an accurate model of the actual chemical process going on in the solutions. Contour plots from the prediction equation for MSC-1 (Figures 13 and 14) for the  $[\text{OH}^-]$ - $[\text{NH}_4^+]$  plane at the low and high levels of  $\text{SO}_3^{2-}$  show that the prediction equation gives a saddlepoint contour. It can be easily seen that this model, although a good predictor for the center point and corners, is not a good model elsewhere. For instance, at pH 3 and 50 ppm  $\text{NH}_4^+$ , 40 to 57% cobalt removed is predicted. Because of the lack of  $\text{NH}_3$  for complexing the cobalt (II) to cobalt (III) and because of the lack of  $\text{SO}_3^{2-}$  at pH 3 (See Figure 15), the cobalt present in solution would almost

Table 8 - FIRST GENERATION PREDICTION EQUATIONS

| Resin    | Equation                                                             | Center point |        |
|----------|----------------------------------------------------------------------|--------------|--------|
|          |                                                                      | Predicted    | Actual |
| AG50WX8  | $y = 103.16 - 1.29X_1 + 0.18X_2 + 0.13X_3 - 0.06X_1X_2 - 0.04X_1X_3$ | 77.78%       | 99.5%  |
| MSC-1    | $y = 103.70 - 1.38X_1 + 0.17X_2 + 0.13X_3 - 0.06X_1X_2 - 0.04X_1X_3$ | 77.23%       | 98.2%  |
| HCR-2W-H | $y = 106.39 - 1.27X_1 + 0.21X_2 + 0.11X_3 - 0.07X_1X_2 - 0.94X_1X_3$ | 78.39%       | 99.5%  |

Table 9 - RESULTS OF EXTRA EXPERIMENTS

| Resin    | Point 1                         |                                 | Point 2                         |                                 | Point 3                         |                                 |
|----------|---------------------------------|---------------------------------|---------------------------------|---------------------------------|---------------------------------|---------------------------------|
|          | Y's                             | $\bar{Y}$                       | Y's                             | $\bar{Y}$                       | Y's                             | $\bar{Y}$                       |
|          | (% $^{60}\text{Co}$<br>Removed) | (% $^{60}\text{Co}$<br>Removed) | (% $^{60}\text{Co}$<br>Removed) | (% $^{60}\text{Co}$<br>Removed) | (% $^{60}\text{Co}$<br>Removed) | (% $^{60}\text{Co}$<br>Removed) |
|          |                                 |                                 |                                 |                                 |                                 |                                 |
| AG50WX8  | 94.3                            | 92.7                            | 33.1                            | 35.0                            | 8.9                             | 16.8                            |
|          | 91.1                            |                                 | 36.8                            |                                 | 24.7                            |                                 |
| MSC-1    | 93.4                            | 91.8                            | 31.0                            | 32.0                            | 6.6                             | 14.4                            |
|          | 90.2                            |                                 | 33.0                            |                                 | 22.1                            |                                 |
| HCR-2W-H | 94.1                            | 92.6                            | 32.5                            | 33.9                            | 8.8                             | 16.4                            |
|          | 91.0                            |                                 | 35.3                            |                                 | 23.9                            |                                 |



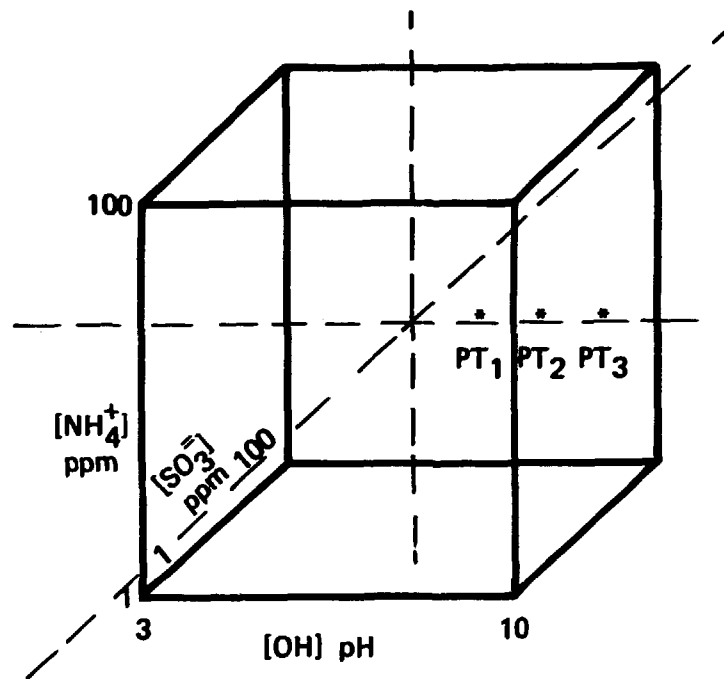


FIGURE 12 - Extra points chosen for cobalt-60 interaction design.

Table 10 - SECOND GENERATION PREDICTION EQUATIONS

| Resin    | Equation                                                                            | Center Point |        |
|----------|-------------------------------------------------------------------------------------|--------------|--------|
|          |                                                                                     | Predicted    | Actual |
| AG50WX8  | $y = -56.67 + 69.03X_1 - 1.71X_2 - 5.42X_1^2 - 0.06X_1X_2 + 0.02X_2^2 - 0.04X_1X_3$ | 95.0%        | 99.5%  |
| MSC-1    | $y = -59.13 + 70.24X_1 - 1.79X_2 - 5.51X_1^2 - 0.06X_1X_2 + 0.02X_2^2 - 0.04X_1X_3$ | 92.6%        | 98.2%  |
| HCR-2W-H | $y = -54.37 + 68.07X_1 - 1.57X_2 - 5.33X_1^2 - 0.07X_1X_2 + 0.02X_2^2 - 0.04X_1X_3$ | 98.6%        | 99.5%  |

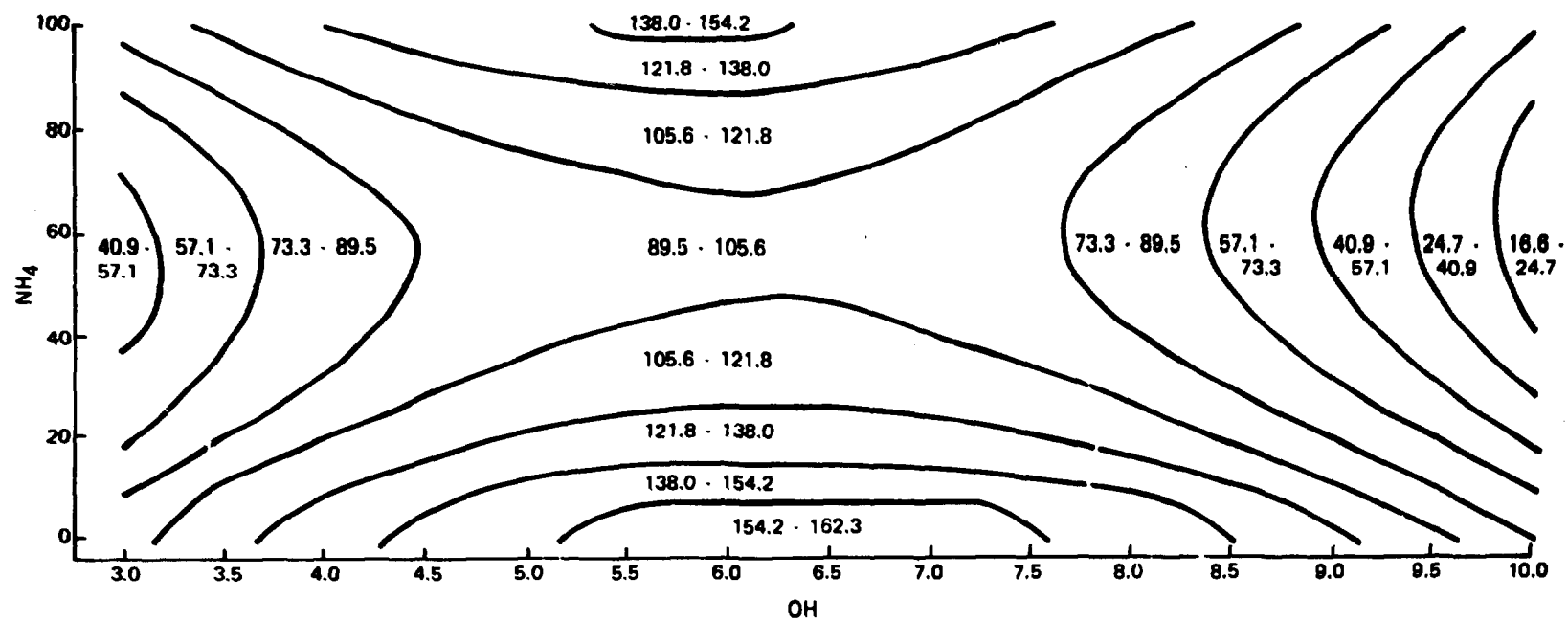


FIGURE 13 - Percent cobalt-60 removal contour plot from the prediction equation for MSC-1 for the  $[OH^-] - [NH_4^+]$  plane at the low level of  $SO_3^{2-}$ .

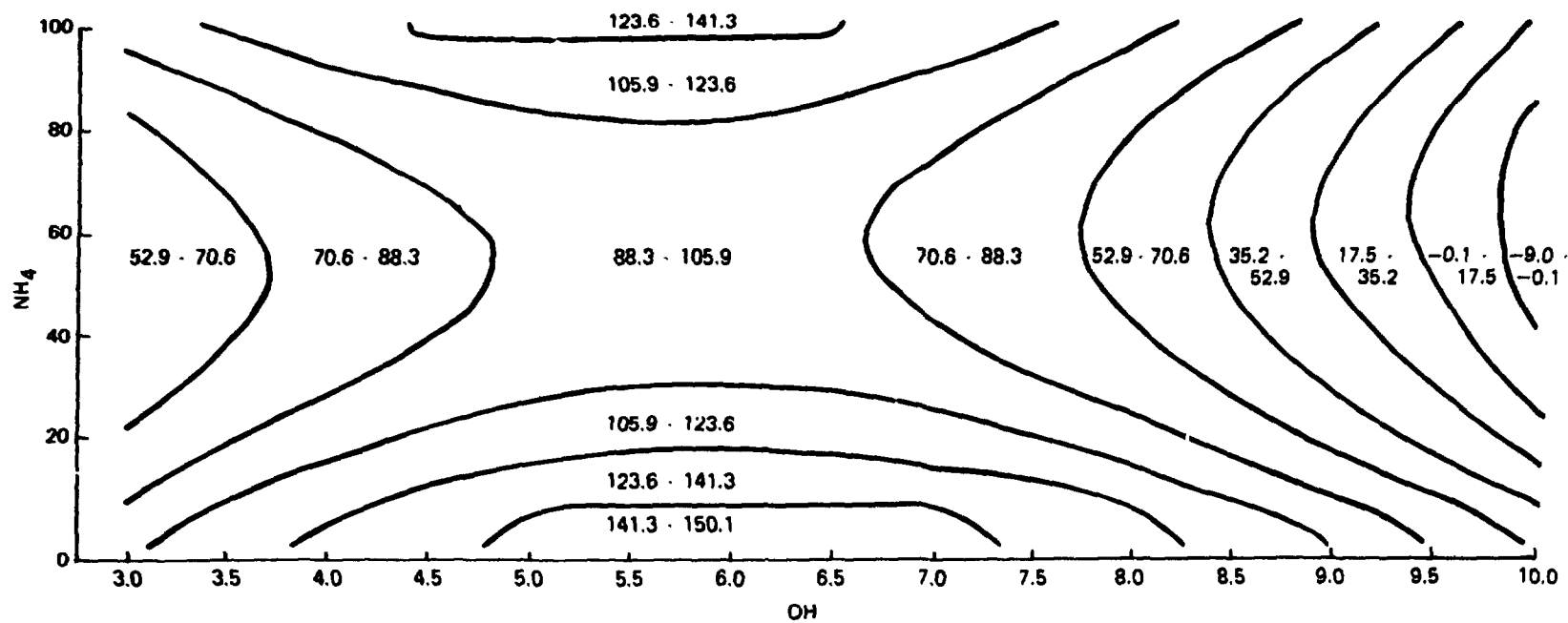


FIGURE 14 - Percent cobalt-60 removal contour plot from the prediction equation for MSC-1 for the  $[OH^-]$ - $[NH_4^+]$  plane at the high level of  $SO_3^{2-}$ .

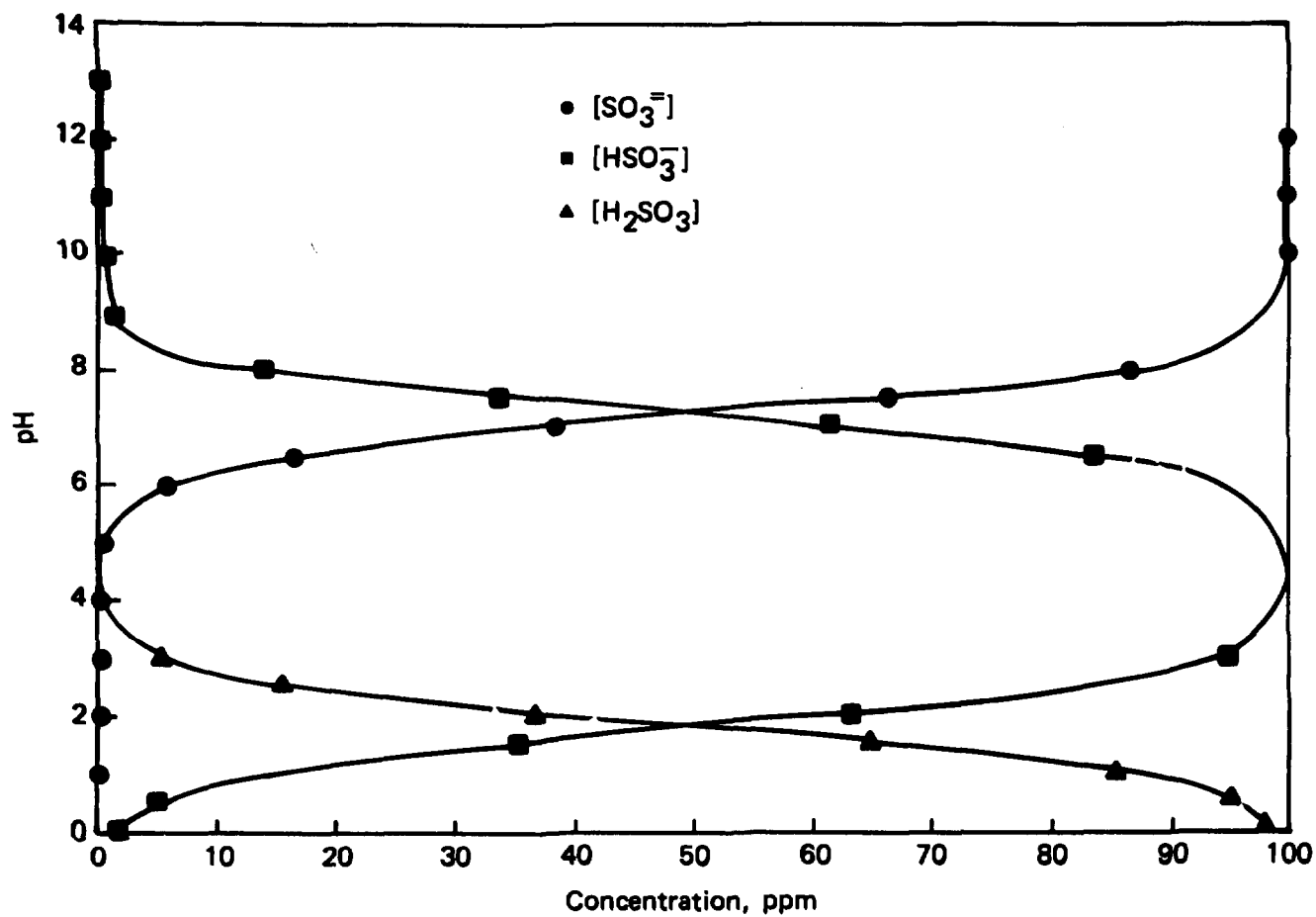


FIGURE 15 - Concentrations of  $\text{SO}_3^{2-}$ ,  $\text{HSO}_3^-$  and  $\text{H}_2\text{SO}_3$  in aqueous solution as a function of pH (calculated).

certainly be in the form of cobalt (II) ions and would be at least 99% removed from the solution by MSC-1. Since this is clearly a poor model, an attempt was made to fit the data to a quadratic with a natural logarithmic term. Each of several modifications of this mathematical form resulted in similar saddlepoint models which were also poor models. In order to produce a good model, more data points would be necessary. Because the areas where cobalt-60 may be effectively removed from solution and the areas where cation exchange does not work well have already been determined, further refinement was considered to be unnecessary.

In general if the pH of the solution is kept acidic, 99% of the cobalt can be expected to be removed from the solution by cation exchange resins. If the solution is basic and cannot be made acidic and free ammonia is present, cation and anion exchange resins must be used and will not remove the neutral complexes that are present. A larger portion of the complexes can be made negative by adding negative ligands such as  $\text{SO}_3^-$ . Other negative ions are known to form negative complexes with cobalt (III). Some of the more common ones that would be of importance in aqueous solution would be  $\text{CrO}_4^{=}$ ,  $\text{SO}_4^{=}$ ,  $\text{NO}_2^-$ ,  $\text{NO}_3^-$ ,  $\text{CN}^-$ , and  $\text{OH}^-$ .

## DISCUSSION

From the results of the cation exchange resin interaction design with cobalt-60, it can be seen that at pH 10 in the presence of free ammonia ( $\text{NH}_3$ ), the cobalt (II) ion is oxidized to cobalt (III). The simple cobalt (III) ion is unstable in water and would be reduced to cobalt (II) except for the ammonia present. The ammonia coordinates with the cobalt (III) and forms complexes that are stable in water. These complexes can be either positive, negative, or neutral. The positive complexes will be removed by a cation exchange resin just as any other positive ion would be. The neutral and negative complexes, however, would remain in the solution and pass through the resin. To further elucidate the complexation process, a complete cobalt interaction design experiment, with exactly the same solution parameters as used in the cation exchange resin experiment, was run on an anion exchange resin. Because the anion exchange resin would remove the negative complexes and let the positive complexes pass through, the anion exchange results should be an inverse of the cation results, except where neutral complexes are formed. In this manner, an approximation can be made of the amount of positive, negative, and neutral complexes formed in each solution.

The anion exchange interaction design and parameters are the same as for the cation exchange experiment. The anion resin used was MSA-1, a strong base anion exchanger in chloride form. The anion exchange interaction design was run several weeks after the cation experiment using solutions made up at the time of the cation experiment, so any complexes that would form would have had ample time for formation. As in the cation experiment at pH 3 to 7, nearly all the cobalt was in the form of cobalt (II) ions and where most were absorbed in the cation experiment,

in this case most passed through the anion resin. Therefore, once again the runs of interest in terms of cobalt (III) complex formation are the pH 10 runs (Table 11).

The complete anion exchange design and results are presented in Figure 16. The results of the pH 10 runs in Table 11 are consistent with cobalt (III) complex chemistry.

In runs 6 and 14, the solution composition is 1 ppm  $\text{NH}_4^+$  and 1 ppm,  $\text{SO}_3^-$  at pH 10 with  $4.0 \times 10^{-6}$  ppm cobalt-60. Since the pH is 10, from Figure 8 it can be discerned that about 0.85 ppm is  $\text{NH}_3$  and 0.15 ppm is  $\text{NH}_4^+$ . Cobalt (III) complexes prefer  $\text{NH}_3$  to  $\text{SO}_3^-$  for complex formation, but will form complexes with both. In this case, probably very few neutral complexes are formed, but likely some are formed. About 10 to 36% of the complexes formed are negative. Since more of the cobalt was in the form of negative complexes in the anion exchange experiment than in the cation experiment, the additional time the solution was in existence did cause the formation of more cobalt (III) complexes. In runs 3 and 15, the composition is 1 ppm  $\text{NH}_4^+$ , 100 ppm  $\text{SO}_3^-$  at pH 10, and  $4 \times 10^{-6}$  ppm cobalt-60. Here, once again, the low concentration of  $\text{NH}_3$  as compared to the  $\text{SO}_3^-$  concentration eliminates most of the neutral cobalt complexes. Between 37 and 50% of the complexes from these solutions are negative, depending on how much time they are given for formation. In Runs 4, 17, 9, and 12, where the  $\text{NH}_3$  concentration is 85 ppm, a large portion of neutral complexes is formed (See Table 11).

Table 11 - EXPERIMENTS IN WHICH pH=10

| Run No. | $[\text{NH}_4^+]$<br>(ppm) | $[\text{SO}_3^-]$<br>(ppm) | Cation<br>(Avg. % R) | Anion<br>(Avg. % R) | Neutral<br>(Prob. %) |
|---------|----------------------------|----------------------------|----------------------|---------------------|----------------------|
| 6/14    | 1                          | 1                          | 90.4                 | 35.9                | <1                   |
| 3/15    | 1                          | 100                        | 63.0                 | 50.3                | <1                   |
| 4/17    | 100                        | 1                          | 44.7                 | 15.6                | 39.7                 |
| 9/12    | 100                        | 100                        | 18.1                 | 26.2                | 55.7                 |

| Trial                           | Run #   | y | $y_1$<br>[OH <sup>-</sup> ] | $x_2$<br>[NH <sub>4</sub> <sup>+</sup> ] | $x_3$<br>[SO <sub>3</sub> <sup>=</sup> ] | $x_1 x_2$ | $x_1 x_3$ | $x_2 x_3$ | y's            | $\bar{y}$ | (y-y) | (y-y) <sup>2</sup> | S <sup>2</sup> | df | r-1 | y <sub>PRED</sub> | $y_{PRED} - y_{Ob}$ | $x_1 x_2 x_3$ |
|---------------------------------|---------|---|-----------------------------|------------------------------------------|------------------------------------------|-----------|-----------|-----------|----------------|-----------|-------|--------------------|----------------|----|-----|-------------------|---------------------|---------------|
| CP                              | 1       |   | o                           | o                                        | o                                        | o         | o         | o         | 0.00           |           |       |                    |                |    |     |                   |                     |               |
| CP                              | 10      |   | o                           | o                                        | o                                        | o         | o         | o         | 2.30           | 0.77      |       |                    |                |    | 2   |                   |                     |               |
| CP                              | 19      |   | o                           | o                                        | o                                        | o         | o         | o         | 0.00           |           |       |                    |                |    |     |                   |                     |               |
| 1                               | 5<br>16 |   | -                           | -                                        | -                                        | +         | +         | +         | 0.00<br>1.54   | 0.77      |       |                    |                |    | 1   |                   |                     | -             |
| 2                               | 6<br>14 |   | +                           | -                                        | -                                        | -         | -         | +         | 26.51<br>45.21 | 35.9      |       |                    |                |    | 1   |                   |                     | +             |
| 3                               | 7<br>11 |   | -                           | +                                        | -                                        | -         | +         | -         | 1.31<br>1.95   | 1.63      |       |                    |                |    | 1   |                   |                     | +             |
| 4                               | 4<br>17 |   | +                           | +                                        | -                                        | +         | -         | -         | 14.01<br>17.21 | 15.61     |       |                    |                |    | 1   |                   |                     | -             |
| 5                               | 8<br>13 |   | -                           | -                                        | +                                        | +         | -         | -         | 0.57<br>0.00   | 0.03      |       |                    |                |    | 1   |                   |                     | +             |
| 6                               | 3<br>15 |   | +                           | -                                        | +                                        | -         | +         | -         | 42.10<br>58.56 | 50.30     |       |                    |                |    | 1   |                   |                     | -             |
| 7                               | 2<br>18 |   | -                           | +                                        | +                                        | -         | -         | +         | 0.00<br>0.02   | 0.01      |       |                    |                |    | 1   |                   |                     | -             |
| 8                               | 9<br>12 |   | +                           | +                                        | +                                        | +         | +         | +         | 22.31<br>30.16 | 26.20     |       |                    |                |    | 1   |                   |                     | +             |
| <hr/>                           |         |   |                             |                                          |                                          |           |           |           |                |           |       |                    |                |    |     |                   |                     |               |
| $\Sigma x\bar{y} =$             |         |   | 125.57                      | -43.55                                   | 22.63                                    | -45.23    | 27.38     | -4.69     | AVG Com        |           |       |                    |                |    |     |                   |                     | -2.93         |
| $\Sigma xy/(T/2) =$             |         |   | 31.39                       | -10.89                                   | 5.66                                     | -11.31    | 6.85      | -1.17     | $b_o = 16.31$  |           |       |                    |                |    |     |                   |                     | -0.73         |
| $\frac{2\Sigma x y/T}{2} = b =$ |         |   | 15.70                       | -5.44                                    | 2.83                                     | -5.65     | 3.42      | -0.59     |                |           |       |                    |                |    |     |                   |                     | -0.37         |

FIGURE 16 - MSA-1 anion experimental design results.



## ION EXCHANGE/ADSORBENT PILOT PLANT

### INTRODUCTION

A decontamination of greater than 99% of the actinides and fission products contained in radioactive waste water can be obtained using ion exchange resins. A system for achieving this result is described in this paper. This ion exchange pilot-plant design is the culmination of five years of study of the decontamination of radioactive waste streams by ion exchange resins and other adsorbents at Mound.

In order to maintain maximum flexibility of treatments, this pilot-plant design is a conceptual design with specific flows, resins, and column specifications, but with many optional features and no rigid equipment specifications. This flexibility allows the system to be amenable to almost any radioactive waste stream. Very specific designs can be constructed from this conceptual design for the treatment of any specific waste stream.

### PROCESS DESCRIPTION

This system is designed to process waste water on a batch basis at a flow rate of 40 gal/min (24 hr/day). This is approximately 20,000 gal/8 hr. This process description is in reference to the pilot plant flow chart, Figure 17. Incoming wastewater is pumped into any of four 20,000-gal influent tanks. When one tank is nearly full and ready for processing, a sample is removed and analyzed for radioactivity, total dissolved solids, suspended solids, total organic carbon, iron content, and pH. If the waste solution is low enough in radioactivity that treatment is unnecessary, it is pumped to one of the four 20,000-gal effluent holding tanks where it is analyzed for radioactivity again and released in the effluent, or recycled, if for some reason it is found to be radioactive. If the influent tank sample is high enough in radioactivity that treatment is necessary, then the rest of the analyses must meet the limits for chemical and impurity limits (from Table 12) before the waste solution can be processed by ion exchange. These limits can be met by some combination of the pretreatment systems shown in Figure 17. These pretreatment systems, Vacco filter system, ultrafiltration, and reverse osmosis, are also effective systems for the removal of radioactivity from wastewater. If pretreatment with all three systems is necessary, it is possible that no further treatment would be required.

In such a case, the pretreated wastewater would be pumped to an effluent tank, analyzed, and released. Tanks and pumps associated with the pretreatment systems are not included in the system flow chart. If after pretreatment, or if pretreatment is not necessary, the radioactivity of the wastewater is still too high to be released, the wastewater will then

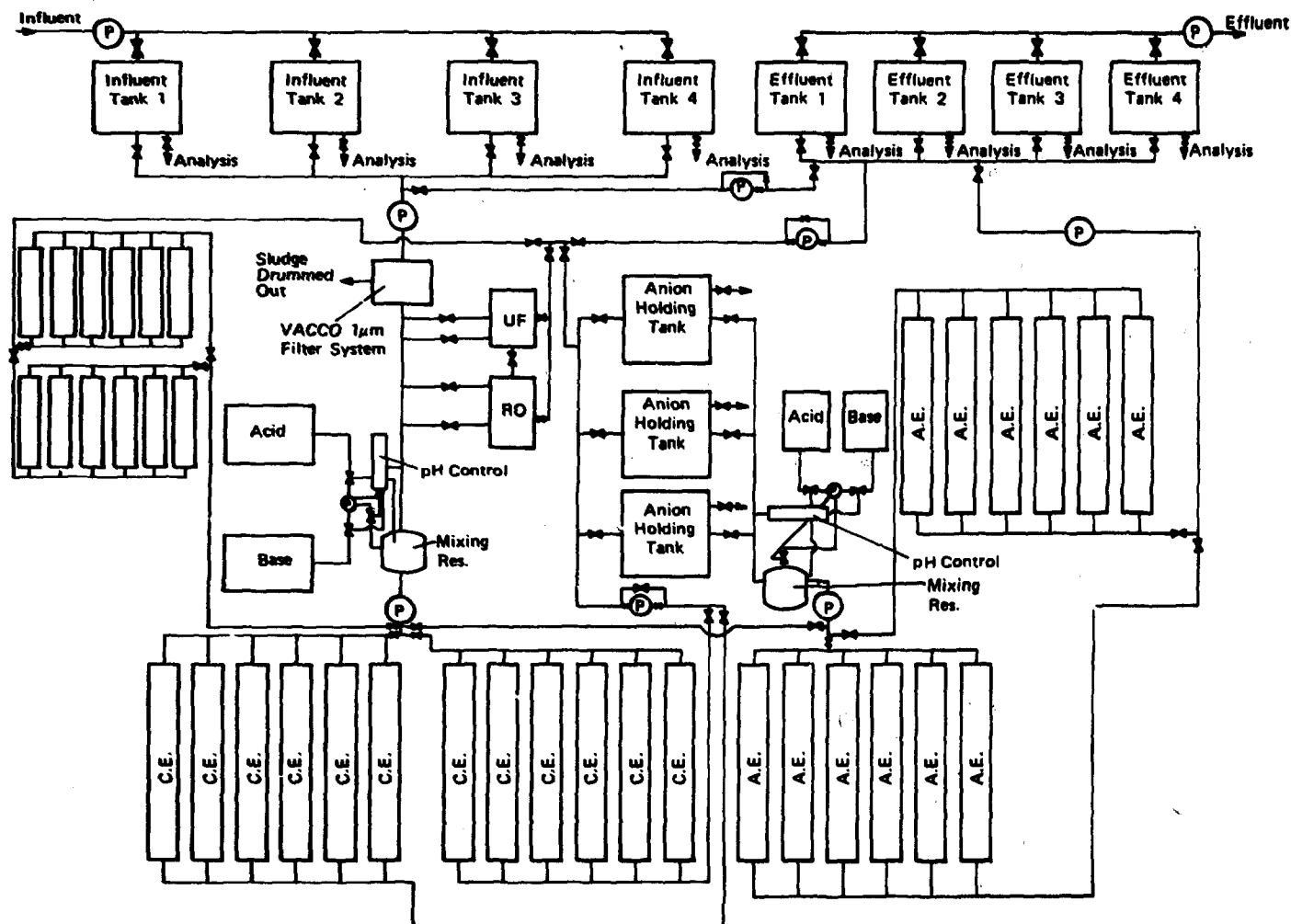


FIGURE 17 - Adsorbent pilot plant schematic (diagram).

Table 12 - WASTE STREAM CHEMICAL AND IMPURITY LIMITS

| <u>Analysis</u>                | <u>Lower Limit</u> | <u>Upper Limit</u>   |
|--------------------------------|--------------------|----------------------|
| Total Dissolved Solids         | None               | 1000 ppm             |
| Suspended Solids Particle Size | None               | 0.1 $\mu$ m Diameter |
| Total Organic Carbon           | None               | 0.01 mg/L            |
| Iron                           | None               | 1 ppm                |
| pH                             | 2                  | 11                   |

be processed by ion exchange. The first step in this process is to adjust the pH to the proper range according to the recommended pH given in Table 6. If only cesium is present, then the wastewater is pumped through the cation exchange columns to an effluent tank where it is analyzed and either released or recycled. If only iodine is present, the wastewater is pumped through the anion exchange columns to an effluent tank. If only cobalt is present, it will be pumped through the cation exchange columns to the anion holding tanks and sampled. At pH 3 to 6, no further processing should be necessary, but if it is necessary, the solution will then be pumped through the anion exchange columns to an effluent tank. If any of the actinides are present, processing through both cation and anion exchangers is necessary. Auxiliary columns are also present and could be used with bonechar for actinides, since bonechar has proven to be particularly efficient at removing most actinides. This system is designed for maximum flexibility in treating a wide variety of radioactive waste streams. If the waste stream is not greatly variable, the system need not be as flexible and could be tailored to suit only that stream. For instance, if the waste stream were free of particulate material, low in dissolved solids, and entirely contaminated with cesium-137, no pretreatment would be necessary and no anion exchangers would be necessary.

There are several basic rules for selectivity in ion exchangers. In general, the exchanger prefers the larger, more highly charged ions over the smaller, lower charged ions. For instance  $Al^{+3}$  would be preferred over  $Ca^{+2}$  or  $Na^{+}$ .

Because of the importance of charge in ion exchange, it is extremely important to know the charge of the ion one wishes to remove from solution. Many elements exhibit multiple valence states, so each element of interest will now be briefly examined to determine what charges may have to be considered in an ion exchange pilot plant of this type. The elements of interest are: the actinides; plutonium (Pu), neptunium (Np), americium (Am), and uranium (U); and the fission products; cesium (Cs), cobalt (Co), and iodine (I). The actinides may be discussed as a group since their chemistry, although somewhat different, is similar in aqueous solution. For uranium, neptunium, plutonium, and americium, the common valence states listed are +3, +4, +5, and +6 although the most stable oxidation state in solution is different for each one. They are +6 for uranium, +5 for neptunium, +4 for plutonium, and +3 for americium. Other valence states have been reported but they are unimportant in aqueous solution. The most common species in aqueous solution for the higher oxidation states of uranium, neptunium, plutonium, and americium are the oxygenated cations  $\text{MO}_2^+$  and  $\text{MO}_2^{+2}$  and the oxoanions  $(\text{MO}_6)^{-4}$ ,  $(\text{MO}_6)^{-5}$ , and  $(\text{MO}_6)^{-6}$ . Because of this wide variety of oxidation states for these actinides and the common formation of negative oxoanions, both anion and cation exchangers would be needed unless an exact characterization of the solution were performed. For instance, if the solution were analyzed and found to contain plutonium and uranium in the forms of  $(\text{PuO}_6)^{-4}$  and  $(\text{UO}_6)^{-4}$  only, then no cation resin would be necessary. Anion resins alone would remove those negative ions easily.

The fission products, iodine and cesium, will normally exist as  $\text{I}^-$  and  $\text{Cs}^+$  in aqueous solutions. Although iodine does exhibit other oxidation states, some of which are positive, these compounds are not commonly found in low level radioactive wastewater. The treatment for cesium is simply a cation exchanger and the treatment for iodine is simply an anion exchanger.

Cobalt is a different problem. Cobalt has only two major oxidation states in aqueous solution, +2 and +3. Cobalt (II) simple compounds can normally be found in aqueous solutions, whereas cobalt (III) simple compounds are unstable in aqueous solution. However, cobalt (III) forms many complexes that are stable in aqueous solution. The cobalt (II) simple compounds are no problem for ion exchange because the cobalt is always  $\text{Co}^{+2}$  and can be easily removed by cation exchangers. The cobalt presents quite diverse problems because cobalt (III) complexes with many other compounds and ions. The problem for ion exchange is that many of these complexes are negative or neutral. The positive and negative species can be removed by passing the solution through a cation exchanger and then an anion exchanger, but the neutral species cannot be removed by ion exchange.

In reference to the waste stream chemical and impurity limits (Table 12), the upper limit given on these is an extreme case where ion exchange will still work but will probably not be economical. The 1000-ppm limit for total dissolved solids is really high for ion exchange. At the flow rates given, if the waste stream contained 1000 ppm Total Dissolved Solids (TDS), as many as 15 resin changes per day would be necessary. This would

become expensive very quickly since the resins are not being regenerated but disposed of in some other way. We feel the best way to dispose of the resins would be by incineration because this would greatly reduce the volume that would have to be buried. The suspended solids should be as close to zero as possible because they could clog the ion exchange columns. The total organic carbon should be as close to zero as possible also, because organics could complex with almost any of the radioactive species and possibly neutralize their charge. Of course, if the species have no charge, ion exchange will not work. Iron concentration also should be as low as possible because iron oxidizes easily inside the resin particles to Fe(III) which will render the resin nearly useless by reducing its surface area significantly. The pH is the least restrictive of these limitations and will not really affect the resin adversely between pH 2 and 11. The pH used should be the range recommended in Table 6.

#### PLANT SPECIFICATIONS

To obtain the flow rate of 40 gal/min, a bed volume of 15 gal of 16 to 40 mesh resin would be used in six columns. This is a flow rate of 0.44 bed vol/min or approximately 20,000 gal/8 hr. The pressure produced in the column at this flow should be very low. A high pressure would indicate some blockage of the flow through the column. The column should have a length-to-diameter ratio of at least 5.0. Several possible column dimensions are given in Table 13. Rather than fabricate such a column, a commercially available column which satisfies the 15-gal bed volume (30-gal total volume) and the L/D ratio of at least 5.0 would be used. The material should be stainless steel either lined or unlined. The resin specifications are given in Table 14. Tanks are considered to be part of the building in which the system exists. All lines and pumps would be specified according to the flows, pressures, and uses.

Table 13 - COLUMN DIMENSIONS

| <u>Length</u><br><u>(ft)</u> | <u>Diameter</u><br><u>(in.)</u> | <u>L/D</u><br><u>(in./in.)</u> | <u>Volume</u><br><u>(gal)</u> |
|------------------------------|---------------------------------|--------------------------------|-------------------------------|
| 5.5                          | 12                              | 5.5                            | 32.31                         |
| 6.0                          | 11                              | 6.55                           | 29.62                         |
| 6.0                          | 11.5                            | 6.26                           | 32.37                         |
| 7.0                          | 10.5                            | 8.00                           | 31.49                         |
| 7.5                          | 10.0                            | 9.00                           | 30.60                         |
| 8.5                          | 9.5                             | 10.74                          | 31.30                         |

Table 14 - ION EXCHANGE RESIN SPECIFICATION

| Resin                | Type        | Form | Capacity         |                  | Manufacturer             | Size Mesh | Cost (\$)                      |
|----------------------|-------------|------|------------------|------------------|--------------------------|-----------|--------------------------------|
|                      |             |      | Wet Vol.<br>Min. | Wet Vol.<br>Min. |                          |           |                                |
| MSC-1                | Strong Acid | Na   | 1.7 meq/ml       | 4.5 meq/g        | Dow Chemical<br>Dowex    | 16-40     | 100/ft <sup>3</sup>            |
| IRA938               | Strong Base | Cl   | 0.5 meq/ml       | -                | Rohm & Haas<br>Amberlite | 16-40     | 336.25/ft <sup>3</sup>         |
| 200                  | Strong Acid | Na   | 1.7 meq/ml       | -                | Rohm & Haas<br>Amberlite | 16-40     | 78.40/ft <sup>3</sup>          |
| AG50WX8<br>DOWX50WX8 | Strong Acid | Na   | 1.7 meq/ml       | -                | Bio-Rad<br>Dow Chemical  | 16-40     | ~44.00/L<br>75/ft <sup>3</sup> |
| HCR2S-H              | Strong Acid | Na   | 2.0 meq/ml       | 4.4 meq/ml       | Dow Chemical<br>Dowex    | 16-40     | 85/ft <sup>3</sup>             |
| MSA-1                | Strong Base | Cl   | 1.0 meq/ml       | 4.0 meq/g        | Dow Chemical<br>Dowex    | 16-40     | 204/ft <sup>3</sup>            |
| SAR                  | Strong Base | Cl   | 1.4 meq/ml       | 3.5 meq/g        | Dow Chemical<br>Dowex    | 16-40     | 193/ft <sup>3</sup>            |
| IRA430               | Strong Base | Cl   | 1.1 meq/ml       | -                | Rohm & Haas<br>Amberlite | 16-40     |                                |

As stated earlier, this system is designed for maximum flexibility and therefore, all possible fabrication combinations could not be presented.

## OPERATING COSTS

Operating costs are based on the worst possible combination of conditions to obtain the highest estimate. The actual cost, if such a system were in operation, should be considerably less. The resin costs are based on treating a 1000-ppm total dissolved solids waste stream with both anion and cation resins. The resins used for the cost estimate are MSA-1 at \$204/ft<sup>3</sup> and MSC-1 at \$100/ft<sup>3</sup>. At 40 gal/min, if all of the capacity of the resin is taken up by ions which are part of the 1000-ppm total dissolved solids, the resins would have to be changed 10 times each 8 hr. Since more than half of the total dissolved solids in almost any waste solution would be sodium, a more reasonable upper limit would be five resin changes every 8 hr. Therefore, the range possible for resin changes is estimated at between 0.33 and 5 resin changes per 8 hr day. It is quite possible that fewer than 0.33 resin changes per day would be needed on some waste streams, but highly unlikely that more than five resin changes per day would ever be necessary. This represents a range of from \$2424/day to \$36,480/day needed for resins. The \$2424/day figure is estimated on a 20-ppm TDS stream.

The cost of chemical and radiological analyses is estimated at \$500/wk if a full-time chemist is available to perform the analyses. The manpower necessary is one chemist and two operators for an 8 hr/day, 4 day/wk operation. The chemist's salary is estimated at from \$24,000 to \$30,000/yr, and each operator is estimated at from \$17,000 to \$25,000/yr. Of course, salaries vary by geographic area, so actual salaries could be more or less than these ranges.

The cost of chemicals is expected to be no more than \$2000/mo. Electrical costs are estimated at \$1200 to \$1500/mo. Drums for drumming the waste resins and transportation and burial for each drum is estimated at \$100/drum. The number of drums needed is based on burial of the resins instead of incineration of the resin and burial of the ash. The amount of resin that can be put in a drum for burial depends on the amount of external radiation generated, but a reasonable range would be from 15 to 45 gal of resin/drum. The total cost of operation is compiled in Table 15.

These estimates also include the total salaries of the chemist and operators, whereas some cost estimates include only the actual time they work on the system. Also, cheaper rates could be obtained for the resins by buying in quantity.

The lower level for drums is based on 45 gal of resin/drum and 20-ppm TDS stream all year. The upper level is based on 15 gal of resin/drum and 1000-ppm TDS stream all year. These costs are based on using resins once and disposing of them. If the resins are eluted and regenerated, the

chemical cost/yr would be approximately doubled, but the resin costs and drum costs could be reduced by a factor of 100. The new costs would be:

|       | <u>Upper</u> | <u>Lower</u> |
|-------|--------------|--------------|
| Total | \$260,358.00 | \$151,716.00 |
|       | \$0.0652/gal | \$0.0380/gal |

#### CAPITAL COSTS

Since this is a conceptual design for an ion exchange pilot plant and the design is intended to be flexible, the capital costs could vary greatly depending on the equipment to be included. The equipment chosen would, of course, depend on the waste stream being treated. From 12 to 36 ion exchange columns could be included; from 4 to 11 pumps of varying sizes could be included; and from 0 to 3 pretreatment systems could be included. All of these varying configurations would have different numbers of valves and different amounts of piping. Three capital cost analyses by I. R. Higgins of an ion exchange plant with 12 times the flow of this pilot plant give capital costs of 5.5¢/1000 gal to 6.76¢/1000 gal. These costs estimates are old (1963) and would be expected to be much higher at 1982 prices. If fabricated, columns for this pilot plant would be between \$1000 to \$3000 each, and pumps will vary, according to size, from \$4000 to \$300. Commercial ion exchange units currently available at 40 gal/min range from about \$7000 to \$25000 depending on what regulations and codes must be met. If a specific waste stream were described, a specific pilot plant and a specific cost could be given.

Further details on these subjects can be found in: M. K. Williams and C. M. Colvin, "Development of Low Level Liquid Waste Treatment Systems: October 1981-March 1982," MLM-2908, May 28, 1982.



Table 15 - ANNUAL OPERATING COST BASED ON AN  
8 HR/DAY, 208 WORK DAY/YEAR OPERATION

|                            | <u>Upper</u>        | <u>Lower</u>     |
|----------------------------|---------------------|------------------|
| Resin                      | \$7,587,840.00      | \$504,192.00     |
| Analysis                   | 26,000.00           | 26,000.00        |
| 1 Chemist                  | 30,000.00           | 24,000.00        |
| 2 Operators                | 50,000.00           | 34,000.00        |
| Chemicals                  | 24,000.00           | 24,000.00        |
| Electric                   | 18,000.00           | 14,400.00        |
| Drums                      | <u>1,248,000.00</u> | <u>27,500.00</u> |
| Total                      | \$8,983,840.00      | \$654,092.00     |
| Total gal/year = 3,993,600 | \$2.25/gal          | \$0.16/gal       |

## MICROWAVE PLASMA INCINERATION

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## ABSTRACT

Microwave supported plasma incineration has been under development at Rockwell Hanford Operations for the past three years. The objective is to develop a technique capable of volume reduction and stabilization of radioactively contaminated or otherwise toxic organic materials not readily reacted with existing incinerators.

The initial bench scale model was capable of reacting 10 ml/hr. An incinerator with a throughput of 1 l/hr. was designed, built, and successfully tested. A large unit capable of reacting 20 l/hr. is proposed.

Combustion efficiencies greater than 99.9 percent are typically observed with a variety of organic liquids. The system is being studied to optimize operating parameters. Plans are being developed to incinerate alpha contaminated organics during the coming year.

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INTRODUCTION

Microwave energy can be used to heat gases sufficiently to generate a plasma. This microwave supported plasma can be utilized to incinerate organic liquids or vapors. Because the energy is supplied independently of the heat of reaction, normally nonreactive organics can be burned and combustion can be supported at reduced pressures.

Radioactively contaminated organic wastes are generated from nuclear fuel reprocessing operations, research centers, hospitals, power reactors, and military applications. These wastes include solvents, oils, sludges, tar, greases, and laboratory chemicals. Some organics, such as polychlorinated biphenyls (PCBs) are both chemically toxic and difficult to burn.

Storage or disposal of these wastes is complicated and in some cases prohibited because of the potential fire hazard, toxicity, or simply because they are liquid. Incineration would convert these materials to a stable waste form that would be suitable for existing or proposed storage disposal systems.

Significant problems have been encountered when attempting to burn some of these organics by ordinary flame combustion, including:

- a) Certain chemical bonds such as carbon-halogen bonds are too strong for efficient combustion in an ordinary flame. Examples include carbon tetrachloride, chloro-aromatic compounds, and fluoro-carbon compounds.
- b) The heat distribution in many flames varies from 500° to 2500°K depending on the fuel-to-oxygen ratio and location within the flame. Because of this spatial temperature distribution, five percent or more of the organic molecules may pass through a flame unreacted.
- c) An explosion hazard always exists for an open flame incinerator operating at atmospheric pressure.

Combustion in a microwave driven plasma promises to solve each of the above three problems. Molecules are excited and ionized by the microwave radiation, providing the activation energy for combustion. The uniformity of the plasma is controlled by the microwave radiation rather than combustion flame kinetics. The effective temperature can be significantly greater than 2000°K everywhere except for a thin boundary layer. Since the plasma is normally operated at pressures below those capable of self-supported combustion, the explosion hazard is minimized.

#### EQUIPMENT DESCRIPTION

The microwave plasma incinerator system is depicted schematically in Figure 1. The operation of the equipment is as follows:

The reaction chamber (8) is a four-inch diameter quartz tube 24 inches long. Microwave energy is applied to the chamber by a 12 kW generator operating at 2450 megahertz (MHz).

Fuel from a reservoir (3) flows through a flowmeter (4), vaporizes through a regulating valve and orifice (5), mixes with oxygen and enters the plasma reaction chamber (8) where combustion occurs. Either the inlet or exhaust pressure can be measured by a pressure gauge using an appropriate valve configuration (9). A quadrupole mass spectrometer connected to the gauge line can monitor partial pressures of either the inlet or exhaust gases. Combustion efficiencies are determined from mass spectrometer peak heights.

A 300 CFM (cubic feet per minute) vacuum pump exhausts the gas through a two inch iron pipe to the building ventilation system.

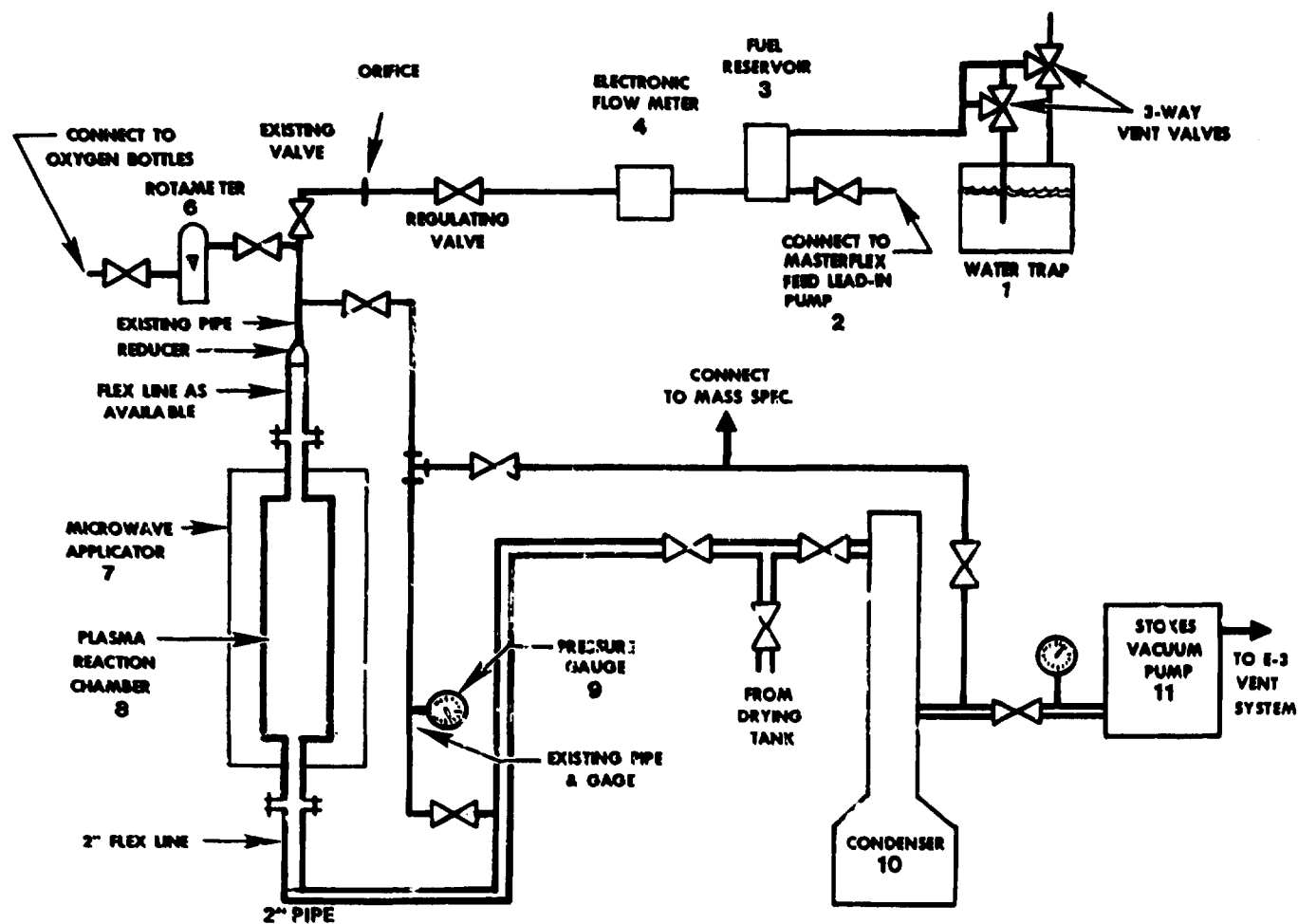


FIGURE 1

# MICROWAVE PLASMA INCINERATOR

## PLASMA PROCESS DESCRIPTION

The plasma is generated by microwave excitation of the electrons and molecules. The degree of excitation is controlled by pressure, oxygen/fuel flow rate, microwave frequency, and input power.

The electrons within the plasma are accelerated by the oscillating microwave field creating an electronic temperature in excess of 10,000°K. The molecular temperature is a function of the electron collision, and the reaction chamber wall temperatures. The collision rate is a function of the gas pressure, the electron density, the relative velocities of the colliding particles, and the collision cross section for the molecules.

In operation, the molecular temperature of the plasma is estimated to be between 700°K and 1900°K. The temperature must remain below 1900°K to avoid melting the quartz reaction chamber. A minimum temperature of 700°K must be maintained throughout the plasma volume to ensure complete combustion.

The combustion rate depends upon the molar velocity of reactants moving through the plasma flame and the combustion efficiency of the process. The molar velocity of reactants, also known as the gas throughput (Q), is defined by:

$$Q = S \cdot P, \quad (1)$$

where  $P$  = pressure,

$S$  = pumping speed.

Residence time ( $T_r$ ) is defined as the average time a reactant molecule spends within the plasma volume. This is given by the equation:

$$T_r = \frac{\text{volume}}{\text{pumping speed}}, \quad (2)$$

where volume = plasma flame volume.

The combustion efficiency ( $\eta$ ) is defined:

$$\eta = 1 - \frac{\sum P_i(\text{ex})}{P(\text{in})}, \quad (3)$$

where  $\sum P_i(\text{ex})$  is the sum of the partial pressure of all organics in the exhaust stream,  $P(\text{in})$  is the partial pressure of the subject organic in the inlet, and all measured pressures are corrected to room temperature.

Throughput and combustion efficiency are controlled by several independent process variables:

- o Plasma Pressure
- o Microwave Power
- o Pumping Speed
- o Fuel to Oxygen Ratio

Other variables, residence time and also plasma temperature are functions of the independent variables. Microwave frequency and plasma flame volume are fixed by the nature of the existing equipment.

### TESTS AND RESULTS

An experimental test plan was prepared for Fiscal Year (FY) 1982 with the following objectives:

- o Perform an  $N^4$  parametric study of combustion efficiency as a function of the four independent variables. Three values (high, medium, and low) were assigned to each independent variable so that a total of  $3^4$  or 81 independent experiments were planned. These test results will establish the acceptable independent variable ranges.
- o Determine maximum combustion efficiency.
- o Determine maximum throughput for the four inch plasma torch at a combustion efficiency greater than 99 percent.
- o Demonstrate microwave plasma incineration versatility by combusting a wide range of organic substances.

After the experiments were completed in May 1982, it was discovered that a malfunctioning leak valve in the mass spectrometer had introduced serious error into the combustion efficiency measurements. The malfunctioning leak valve was removed from the system. The experiments are currently being repeated.

The data observed to date are adequate to confirm the following:

- o Combustion efficiency - The maximum combustion efficiency as determined by the mass spectrometer is 99.9 percent or greater. The mass spectrometer measurements were confirmed by trapping combustion products on a liquid-nitrogen cooled "cold finger". A total organic carbon analysis showed less than 80 ppm organic carbon in the condensate.

- o Six types of liquid organics have been combusted:

carbon tetrachloride

methyl isobutyl ketone (hexone)

TBP-alkane mixture

machine oil

hydraulic fluid

greases dissolved in hexone

- o A combustion rate for hexone of 1  $\mu$ /hr. has been established at 99.9 percent efficiency. This was the targeted combustion rate for the incinerator.

In addition to the "cold" testing, design drawings are being prepared for installing the microwave plasma incinerator in containment prior to "hot" testing in FY 1983.

#### DESIGN PROBLEM AREAS

The following design problem areas have been identified:

- o Reactant preignition - With temperatures near 1800°K, and pressures greater than 10 Torr a flame can propagate backwards into the reactant gas mixing manifold, (which is at a higher pressure than the plasma). If the reactant volume is large, this preignition can cause a sharp pressure excursion which may break the quartz reaction chamber. This problem was solved by using a smaller mixing chamber that delivered the reactant gases directly to the plasma flame.
- o Combustion chamber packing - It was postulated that the combustion efficiency could be improved by filling the reaction chamber with quartz Raschig rings. The objective was to eliminate streamlining and increase the residence time. Instead, it was found that the rings reduced microwave penetration and quenched the plasma, thereby causing a loss of combustion efficiency. To maintain a uniform, high temperature plasma, which is the key feature of microwave plasma incineration, it is necessary to keep the reaction chamber free of any obstructions.

- o Heat exchange - The maximum microwave power utilized in this experiment is 12 kW. The power released through burning 1  $\mu$ /hr. of hexone is approximately 11 kW. The external off-gas pipe temperature does not exceed 50°C at a point 2 feet downstream from the reaction chamber exhaust. This implies that approximately 20 kW of heat is removed through the quartz chamber wall. Failure to remove this heat would cause the quartz to melt. Presently heat is removed by forced air cooling as in all previously reported models. However, this technique is a severe restriction to continued scale-up of the process and a water cooled, metallic, coaxial chamber is proposed for larger models.
- o Plasma volume - For a given pumping speed and operating pressure, a larger plasma volume would increase the molecular residence time, thereby, increasing the molecular temperature and combustion efficiency. Alternatively, a greater combustion rate at a given combustion efficiency value could be obtained by increasing both the plasma volume and the pumping speed. The chamber radius may be increased to achieve greater volume, however, the penetration of the microwave radiation through the plasma decreases exponentially with radial distance. The microwave penetration and therefore the maximum plasma diameter depends upon the microwave frequency, the available power, and the spatial focussing of that power by the applicator. The maximum diameter chamber which can be used at the 2450 MHz frequency currently available is 4 inches. Changing to 915 MHz will permit a 6 to 8 inch diameter chamber to be utilized.
- o Effluent Control - The size range of particulates, such as metallic oxides created by the plasma flame, has not yet been characterized. Such a characterization effort is in progress. The results are necessary in order to allow a filter media to be designed which will trap all radioactive particulates produced by the proposed combustion process.

#### PLANNED (FY 1983) DEVELOPMENT ACTIVITY

The incinerator will be placed in containment and the combustion of radioactively contaminated material will be demonstrated. An exhaust gas processing system will be demonstrated that will clean the off-gas to releasable levels. Solids from the exhaust will be collected by submicron filters. Condensable gases will be collected by a vacuum condenser; and the non-condensable gases will be filtered through absolute filters. If funding permits, experimentation will continue in FY 1983 toward utilizing a 915 MHz, 30 kW generator to scale the the plasma incinerator to a 20  $\mu$ /hr. combustion rate.



## DISCUSSION

The ultimate usefulness of the microwave incinerator depends upon a variety of parameters, including:

- o Throughput
- o Existence of organics not combustible by other techniques
- o Cost
- o Combustion efficiency
- o Off-gas handling requirements
- o Pumping capacity

The throughput can be increased by operating at higher pressures. Higher pressures, however, require heat removal rates that may exceed physical limits. More importantly, as pressure increases assurance against explosions is lost.

The total inventory of organic liquids at Hanford for which a benefit could be accrued by incineration does not exceed 50,000 gallons. Furthermore, the bulk of this is hexone, an easily burned organic. Any special purpose incinerator facility would cost millions of dollars which puts the incineration cost at about \$1,000 per gallon.

A good example of a recent problem for which the microwave incinerator could be gainfully applied was the need to dispose of 200 liters of PCB and alpha contaminated pump oil. A system capable of processing 1 l/hr. could have solved the problem in a week. A 1 l/hr. incinerator could be placed in a single room of an existing facility at a cost of several hundred thousand dollars, and could handle several thousand liters per year. The cost would be of the order of \$100 per liter.

The currently measured combustion efficiency of 99.9% is not necessarily acceptable for direct release of the off-gas. We are continuing to determine the ultimate combustion efficiency.

The off-gas will most certainly contain submicron particles. The extent of the difficulty these small particles may cause is still unknown. This subject should be resolved next year.

The pumping system should probably be based on a steam jet ejector. If mechanical rotary piston pumps are used for small volumes of organic liquid, and if submicron particles become trapped in the pump oil, one is simply transferring the problem, not eliminating it. With a steam jet ejector, the contaminants become incorporated into a low-level aqueous waste stream which can be mixed with the other aqueous waste.

**SUMMARY**

Substantial progress in scaling up the combustion rate of the plasma incinerator has been made. Scale-up was successfully achieved by increasing the vacuum pumping speed, by improving the microwave applicator design, and by increasing the plasma radius and, thereby, its volume for a given reactor length.

The testing and optimizing experiments performed have identified ideal operating parameters and major design difficulties.

A "hot" demonstration is planned for FY 1983 using the 4-inch diameter incinerator burning at least 1 l/hr. of liquid organic wastes. Continued scale-up of the process using the 915 MHz generator is also anticipated in FY 1983.

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VOLUME REDUCTION OF  
LOW LEVEL WASTE  
AT THE INEL  
WASTE EXPERIMENTAL REDUCTION  
FACILITY

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ABSTRACT

The Waste Experimental Reduction Facility at the Idaho National Engineering Laboratory is a full scale, low level beta/gamma waste processing facility designed to extend the useful life of the on-site waste disposal activity and to perform developmental activities in the areas of low level waste handling and processing. The volume reduction processes included are sizing of ferrous metals for charging into furnaces, melting the size reduced material in a coreless induction furnace to pour an easily packaged ingot, and incineration of combustible waste.

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INTRODUCTION

The Waste Experimental Reduction Facility (WERF) is located on the Idaho National Engineering Laboratory (INEL). The facility has been constructed for two distinct purposes. The first is to increase the useful life of the on-site radioactive disposal area, the Radioactive Waste Management Complex (RWMC). Secondly, but equally important, the full scale facility will perform development activities in the areas of low level waste handling and processing. The WERF contains three activities; the Waste Sizing Development Activity (WSDA), the Waste Melting Development Activity (WMDA), and the Waste Incineration Development Activity (WIDA). The purpose of this paper is to discuss the first two activities, but information is included on the incinerator.

BACKGROUND

There has been little incentive in the past to reduce the volume of low-level beta/gamma wastes due to the relatively easy disposal

methods for this classification of waste. However, there have been several developments which have had, or are having, a direct impact on bet./gamma waste disposal.

These events include:

1. Closure of one-half of the existing commercial disposal sites
2. Limiting volumes of wastes allowed to be received at disposal sites
3. Increased waste packaging requirements
4. Restrictions on interstate transportation of radioactive wastes
5. Increased restrictions on operation and location of new disposal sites
6. Increased packaging and transportation costs due to the above developments.

Radioactive wastes generated at the INEL (as well as some waste received from other sites) are disposed of at the RWMC. Various studies have been made to determine the remaining useful life of the RWMC, and the most probable closure date [subsurface disposal area (SDA) only] based on these studies, in 2020 (assuming no action is taken).

As the cost to handle and dispose of beta/gamma wastes increase, reducing the volume of this waste becomes economically more attractive. Economic considerations coupled with an immediate need to increase the life of the RWMC make the development of processing methods for low-level beta/gamma wastes attractive.

Strategies for extending the life of the SDA include: waste generation reduction; volume reduction by compaction at the RWMC; and volume reduction through sizing, melting, and incineration of waste material. It is estimated that mechanically reducing the size of metallic material, followed by processing in a melter, could reduce the total waste volume by about 30% and serve to extend the life of the SDA (Table 1). Incineration of combustible waste results in a volume reduction of about 40:1 and will serve to further extend the life of the SDA.

Volume reduction through sizing, melting, and incineration will be performed at the WERF. WERF, in addition to extending the life of the RWMC, will provide a full scale facility to develop techniques and methods needed for low level beta/gamma waste processing in the nuclear industry.

## Radioactive Solid Waste Forecast

| Total | 1980 | 1981 | 1982 | 1983 | 1984 | 1985 | 1986 | 1987 | 1988 | 1989 |        |
|-------|------|------|------|------|------|------|------|------|------|------|--------|
|       | 5642 | 6927 | 6750 | 6764 | 6313 | 7492 | 7092 | 7892 | 7692 | 7692 | (1979) |

## Radioactive Solid Waste Composition

| Volume<br>m <sup>3</sup> | Ferrous<br>metal | Non-ferrous<br>metal | Compactible<br>combustibles | Non-compactible<br>combustibles | Compactible<br>non-combustible | Resins | Other |
|--------------------------|------------------|----------------------|-----------------------------|---------------------------------|--------------------------------|--------|-------|
| 5700                     | 2028             | 191                  | 1366                        | 741                             | 60                             | 48     | 1269  |
| % of<br>volume           | 35.6             | 3.4                  | 24.0                        | 13.0                            | 1.1                            | 0.8    | 22.3  |

INEL-S-31 754

TABLE 1

## FACILITY DESCRIPTION

### BUILDING DESCRIPTION

The WERF was established by adapting the existing, recently decommissioned, SPERT III reactor building at the INEL.

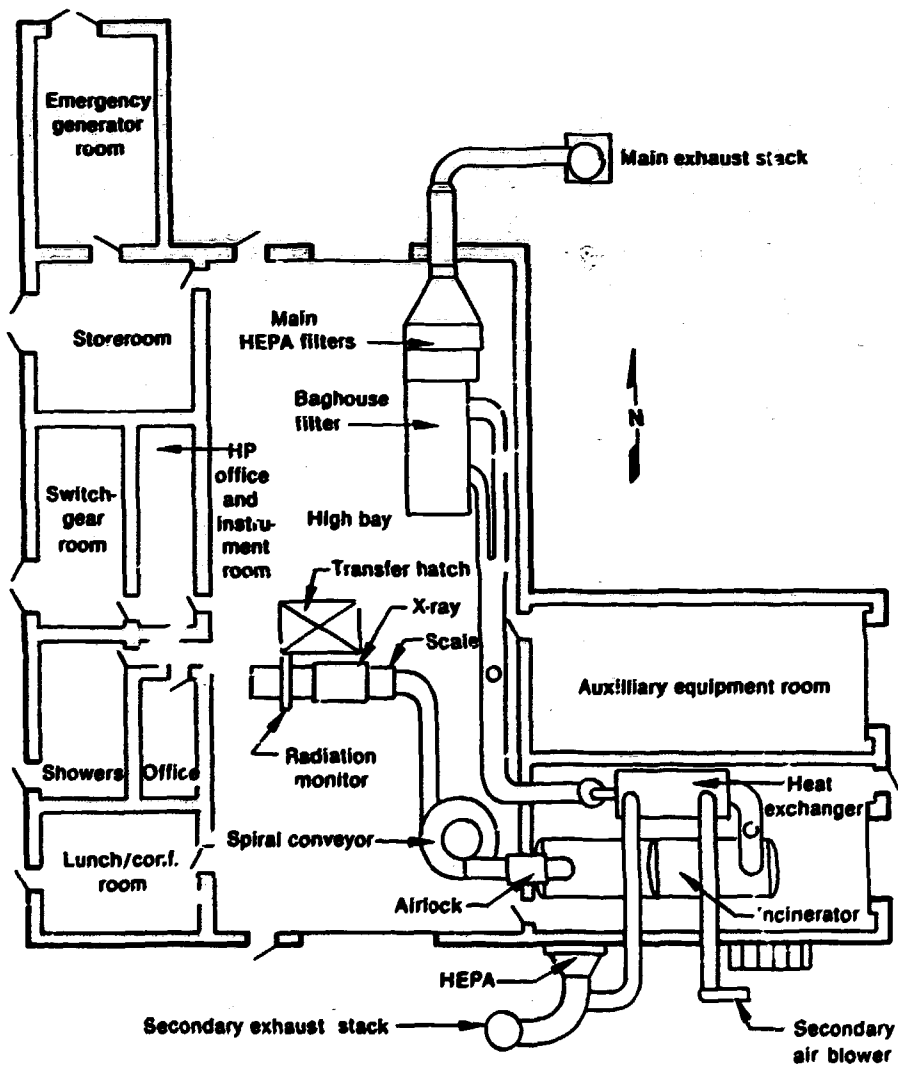
The SPERT-III (Special Power Excursion Reactor Test) reactor building (PER-609) was chosen as the site for the WERF because of the availability of the facility, physical lay-out and location, and because of the economic and environmental advantages of using an existing building as opposed to the construction of a new facility elsewhere.

SPERT-III was chronologically the second SPERT reactor in operation and was built in the late 1950's. The reactor was designed to conduct studies of high-power, high-temperature, heterogeneous, light water reactors. SPERT-III was built to determine the effects of water flow, pressure, and temperature on the transient characteristics of the reactor.

The original building has been modified to accommodate the equipment added to, and operations conducted in, WERF. A basement room extension measuring 10 ft (3.05 m) wide by 40 ft (12.20 m) long, was constructed along the north side of the existing building. This extension houses the two furnace power supply units. The second extension, 40 ft (12.2 m) wide by 42 ft (12.5 m) long, consists of both a basement and above-ground extensions and was constructed along the east side of the existing structure. This extension was built to house the incinerator, provide for auxiliary equipment, and provide separate areas in the basement for ingot cooling, ash collection, and waste processing. The third extension, 18 ft (5.5 m) wide by 26 ft (7.9 m) long, was designed as the emergency diesel generator room. This room is attached to the storeroom at the northwest corner of the original reactor building.

Total gross area of the PER-609 building with the WERF modifications is approximately 11,000 ft<sup>2</sup> (1000 m<sup>2</sup>), and the volume is approximately 207,000 ft<sup>3</sup> (5700 m<sup>3</sup>). The first floor area (Figure 1) includes: a receiving area; a process feed preparation area; high-bay process and off-gas treatment area; incinerator room and auxiliary equipment room; emergency generator room; HP office and instrument room; a switchgear room; a restroom; a lunch room; and offices.

The lower floor (Figure 2), with the exception of the control and observation room and the furnace power supply room, is intended for handling low-level beta-gamma contamination in operational amounts. Therefore it is specially equipped and ventilated to provide positive containment. All surfaces are finished to render them easily decontaminable.



INEL 23259

Figure 1 WERF Main Floor



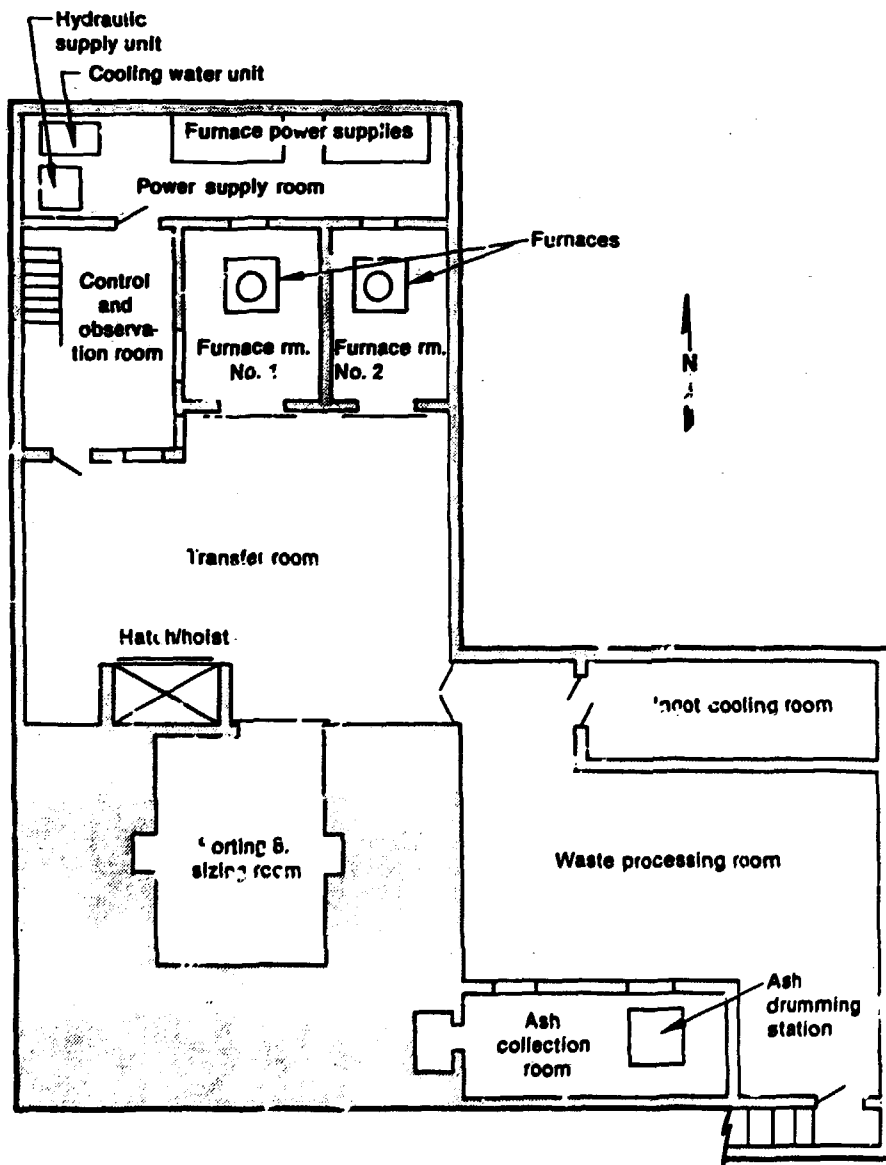


Figure 2 WERF Basement

INEL 2 3258

The reactor building itself is a two-story structure with masonry exterior walls, and a concrete and steel frame. The ceiling height is 30 ft (9.1 m) in the high-bay and 12 ft (3.7 m) in the basement. The floor is rated for heavy duty loading of 300-500 lb/ft<sup>2</sup> (1464-2441 kg/m<sup>2</sup>).

During the decommissioning a concrete pad 72 ft (22 m) by 96 ft (30 m) was built to store the contaminated piping and reactor vessel removed from the building for future processing in WERF. This pad will serve as a receiving area for waste sent to WERF by the generators. The pad is surrounded by an 8 ft (2.4 m) high chain link security fence. A 6 inch (15 cm) curb has been constructed around the pad to direct any runoff of accumulated precipitation to the northeast corner where it will be periodically collected and monitored for radioactivity.

Waste generators will be required to segregate their ferrous metallic waste and combustible waste and place it into reusable metal shipping containers which will be sent to WERF and stored on the pad. A lead storage of two to four weeks throughput of waste will be maintained on the pad. Only containers reading 50 mR/hr beta-gamma or less on contact are initially being collected for processing.

#### WASTE SIZING DEVELOPMENT ACTIVITY (WSDA) AREA DESCRIPTION

The Waste Sizing Development Activity is located in the basement of the building and consists of a sorting/sizing room and a transfer room (Figure 2).

Waste containers with metallic waste are moved from the interim storage pad via forklift to the hatch in the main floor leading to the basement. Once the hatch is opened the containers are placed on a scissors lift and lowered into the basement. The scissors lift is fully enclosed and serves as an airlock between the basement and the main floor. Lights activated by limit switches are installed and warn persons on both levels against opening both doors simultaneously.

The reactor pit has been converted into a sorting/sizing room, isolated from the remainder of the basement by equipment doors, and is maintained at a net negative pressure with respect to the main floor. In addition to sorting and sizing, small amounts of residual thermal insulation and other undesirable materials are removed from objects to be sized in this room. Due to the nature of the operations being conducted, it is required that full-time respiratory protection be worn while dust or fume producing work is in progress.

The shipping containers will be opened and the metallic objects will be sorted. Metallic wastes received must be reduced in size as much as practical for loading into the induction furnace crucible. This sizing is accomplished using several techniques.

- A. Plasma-arc torch cutting. A Thermal Dynamics PAK-44 plasma-arc torch and a Linde PCM-250 plasma cutting system are available for cutting thick gauge material. These torches are transferred-arc types using nitrogen as the plasma gas and nitrogen or argon as the shield gas.
- B. Guillotine Saw. Medium to large-diameter pipes and similarly sized structural material is cut using the guillotine saw. This reciprocating cutter clamps to the object being cut. Due to the short stroke of the blade, airborne contaminants are minimized. Since it is clamped in place, the cutting is accomplished without the operator having to be in close attendance, thereby serving to minimize personnel radiation exposure.
- C. Pipe saw. A Wacks Model E "Trav-l-Cutter" is available for use at the WERF. Formerly used in several decommissioning projects, this saw can cut large diameter pipes and tanks up to six feet in diameter. The sectioning of these objects requires approximately one minute for every inch of diameter. As with the guillotine saw, airborne contaminants are minimized and once set up the cutting is accomplished without the operator in close attendance.
- D. Plate shear. Large sheets of steel up to 3/8 inch thick are cut using this handheld, portable electric plate shear. This tool cuts rapidly (five feet per minute) and no undesirable chips are generated in the process, as with sheet metal nibblers.

Once the objects are sorted and size reduced they are placed in bins and transferred to the furnace rooms for melting. Objects too large to be size reduced in the basement, such as tanks, large pipes, etc., will be size reduced on the main floor or pad in special, temporary enclosures.

A ventilation system exhausts the sorting/sizing room, combining with the main facility exhaust and passing through the baghouse and HEPA filter bank. Point of operation exhaust is used to the maximum extent practical for all cutting operations. This serves to minimize fugitive emissions and resultant airborne radioactivity, surface contamination, and personnel exposure to toxic fumes. A precipitator is being back-fit to the Sorting and Sizing Room ventilation to reduce the rapid deposition of particulates on the baghouse bags.

#### WASTE MELTING DEVELOPMENT ACTIVITY (WMDA) AREA DESCRIPTION

The WMDA is also located in the facility basement and is comprised of an observation and control room, power supply room, two furnace rooms, and an ingot cooling room (Figure 2).

The control and observation room is used as the main entrance for personnel access to the basement, provides a place for viewing the basement activities and serves as a central point for controlling melting and incineration operations. Windows have been installed, enabling the operators and visitors to monitor melting and sizing operations. Hydraulic controls and remote furnace controls have been installed, allowing melting and pouring from this room.

The furnace equipment room was an addition to the existing facility and is located on the north end of the building adjacent to the control and observation and furnace rooms. This room serves to house the two furnace power supplies, along with the hydraulic unit and cooling system pumping station. Windows have been installed to allow the viewing of melting operations while making adjustments at the power supply.

Two furnace rooms have been constructed, each to house a coreless induction furnace and associated equipment. In furnace room #1 a 1500 lb capacity Pillar furnace has been installed (Figure 3) along with an industrial robot (Figure 4) for remote handling and maintenance. Furnace room #2 contains an Ajax 1000 lb induction furnace for future developmental work concerning metal recycling and as a backup to the main furnace.

The furnaces have been fitted with close-capture fume hoods for the removal of metal fumes and smoke. The hoods are mounted to the tops of the furnaces and connected to the main exhaust duct via telescopic ducts and swivel joints to allow for movement while pouring. The off-gas from the furnaces subsequently joins the main facility exhaust for treatment prior to release from the stack.

Once the metal has been size reduced it will be charged into the furnace either manually or remotely using the robot. The metal will then be melted, slagged, and poured into a reusable metal mold sized to hold the entire contents of the furnace (Figure 5). The mold and ingot will be removed from the furnace room to the cooling room and the slag will be compacted and/or solidified. When adequately cooled the ingot will be removed from the mold, packaged (Figure 6), and shipped to the RWMC.

#### WASTE INCINERATION DEVELOPMENT ACTIVITY (WIDA) AREA DESCRIPTION

The major portion of the WIDA is located in the new east wing of the building with some support equipment located in the high bay.

The principal equipment for this activity is a dual-chambered controlled air incinerator (Figure 7) capable of burning 400 lb per hour of 12,000 Btu/lb combustible material.

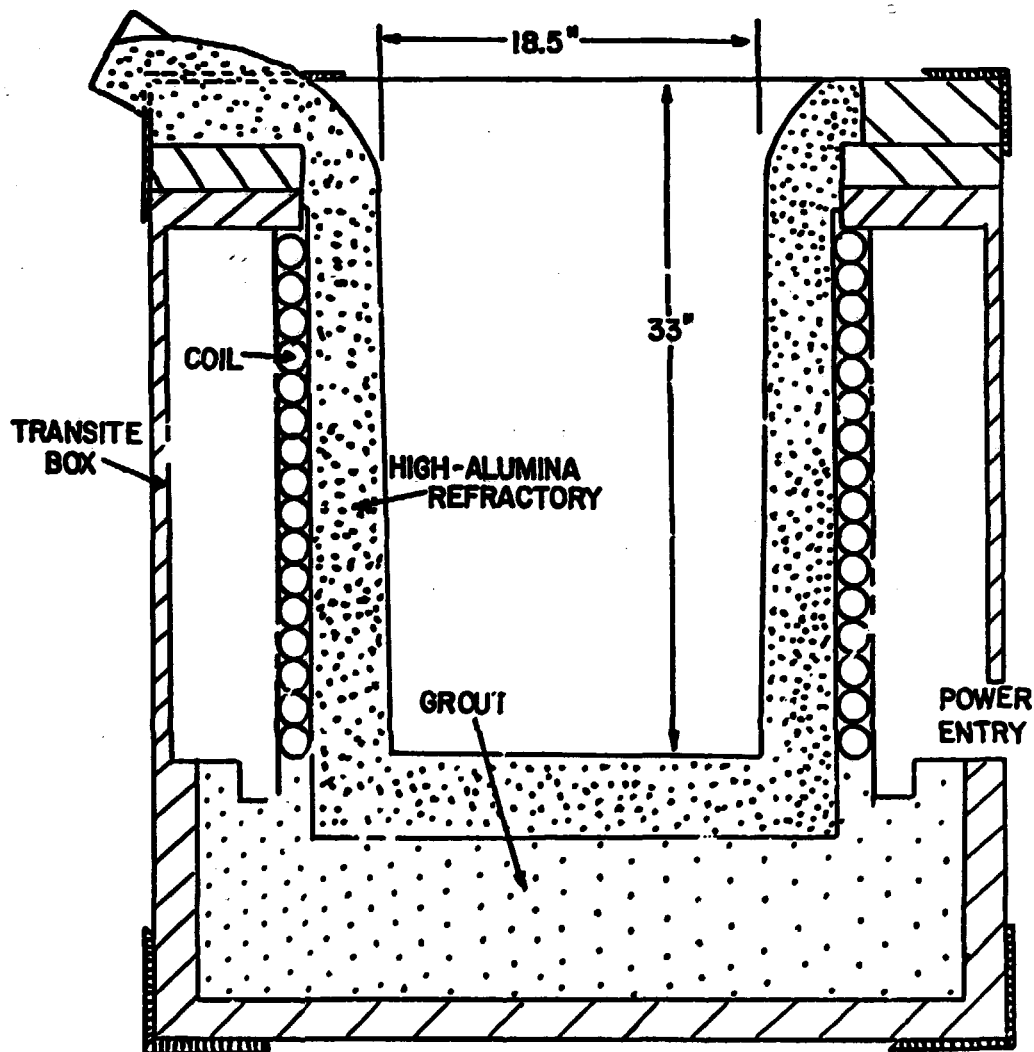


Figure 3  
CORELESS ELECTRIC INDUCTION FURNACE



Figure 4  
ROBOT

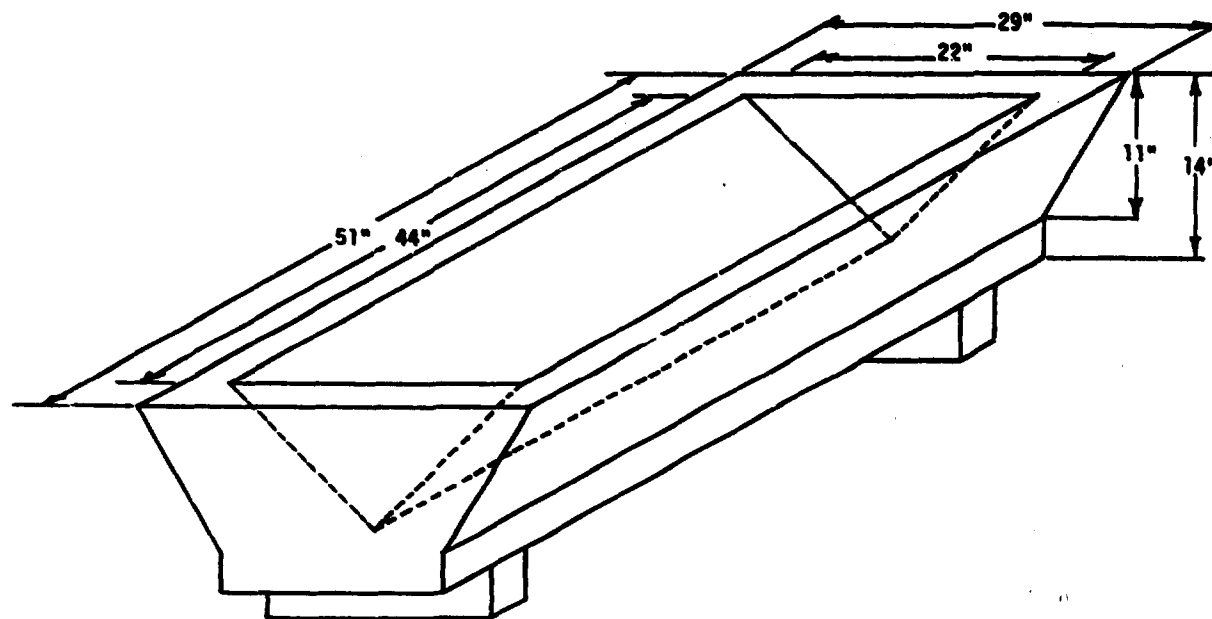


FIGURE 5  
1500 LB CAPACITY INGOT

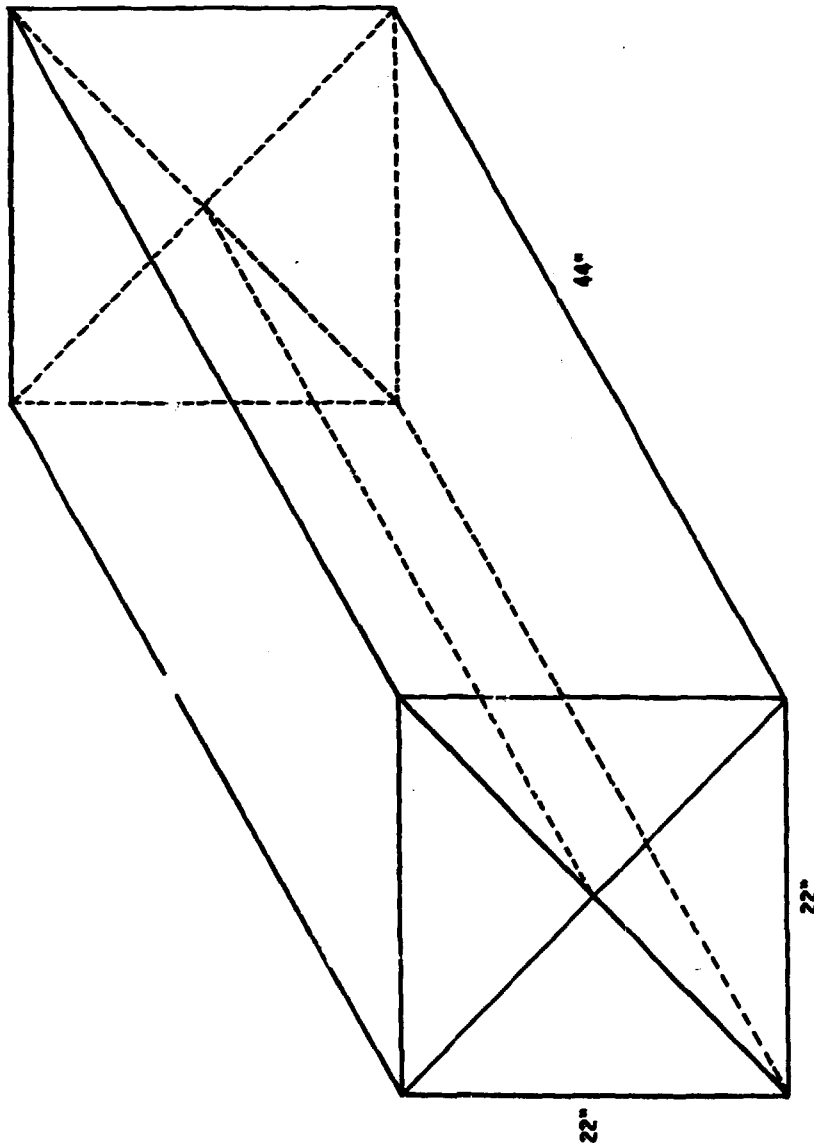


FIGURE 6  
INGOT PACKAGE (TYP)



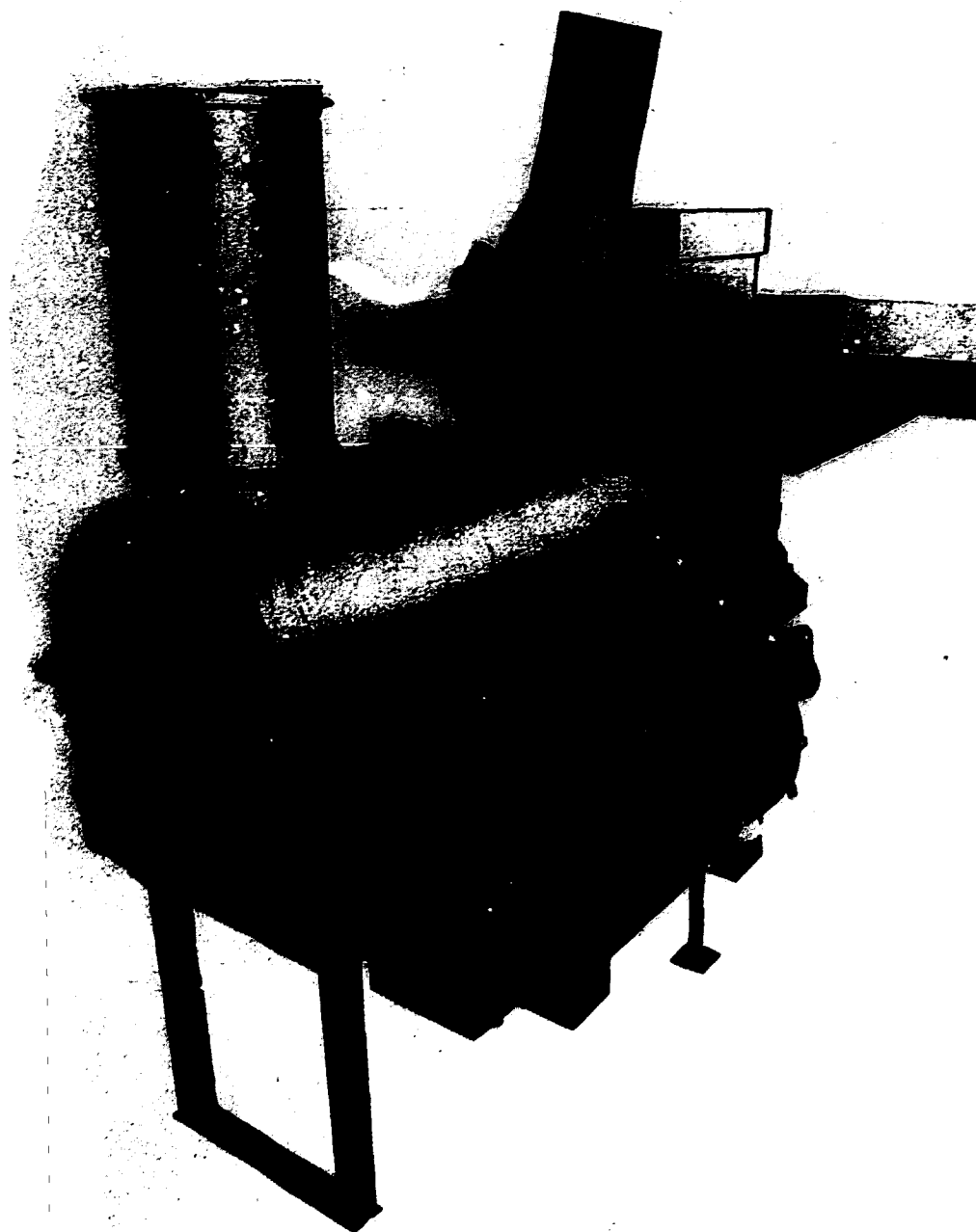


Figure 7 Controlled Air Incinerator

Combustible low-level radioactive waste as received from the generators should be well segregated as to combustible content and pre-packaged in 2 ft x 2 ft x 2 ft corrugated cardboard boxes with internal sealed polyethylene liners. These cardboard boxes containing waste are placed on a conveyor. The primary function of the feed conveying and inspection system is to transport the combustible material from the feed staging area to the incinerator feed mechanism and monitoring for specific parameters enroute.

For informational and control purposes, each box will be weighed by a conveyor mounted scale. This data provides the operator with a guide to either slow down or increase the feed rate. The incinerator is designed to handle 400 lb/hr of 12,000 Btu/lb waste. This equates to about seven to ten boxes of waste per hour, using a waste density of 5 to 7.5 lb/ft<sup>3</sup>. This data will be recorded and periodically used in evaluating and reporting the incinerator's performance.

Radiation intensity from each box will be measured and recorded. Only those boxes reading 10 mR/hr or less will be initially processed. Boxes exceeding this level will be flagged and rejected by the system for subsequent sorting or alternate disposal. In addition to providing a control for acceptance of feed material, radiation intensity also provides a rough measure of the curie content of the waste, which is another important parameter in evaluating the performance of the incinerator. This data can be compared to the curie content of the ash and refractory, giving an indication as to where the radionuclides end up and aid in determining the expected dose buildup in the vicinity of the incinerator.

Although some noncombustibles are acceptable to the incinerator, large pieces may interfere with the ash removal process and subsequent treatment methods for ash disposal (i.e., solidification). For this reason and to preclude explosive liquids, conveyed waste is also monitored for noncombustibles and liquids through X-ray inspection, using a standard airport-type X-ray unit, in-line with the conveyor. This type unit is shielded and does not significantly add to background radiation levels. A video monitor is provided in the control/observation room along with controls to allow remote operation of the X-ray unit.

Waste feed material is admitted to the bottom chamber through a gravity drop chute (Figures 8 and 9). Initial energy for combustion is provided by the lower burner using #2 fuel oil. In the lower chamber pyrolysis takes place under an oxygen deficient environment. As the gases pass into the top chamber, they are ignited by a second burner in the presence of 100% excess air. Incineration in this manner allows for complete combustion with minimal disturbance of the ash, thus minimizing the amount of particulate carried over into the exhaust system. Both chambers of the incinerator have been extended four feet in length to assure complete combustion through increased residence time (2 sec at 1100°C).

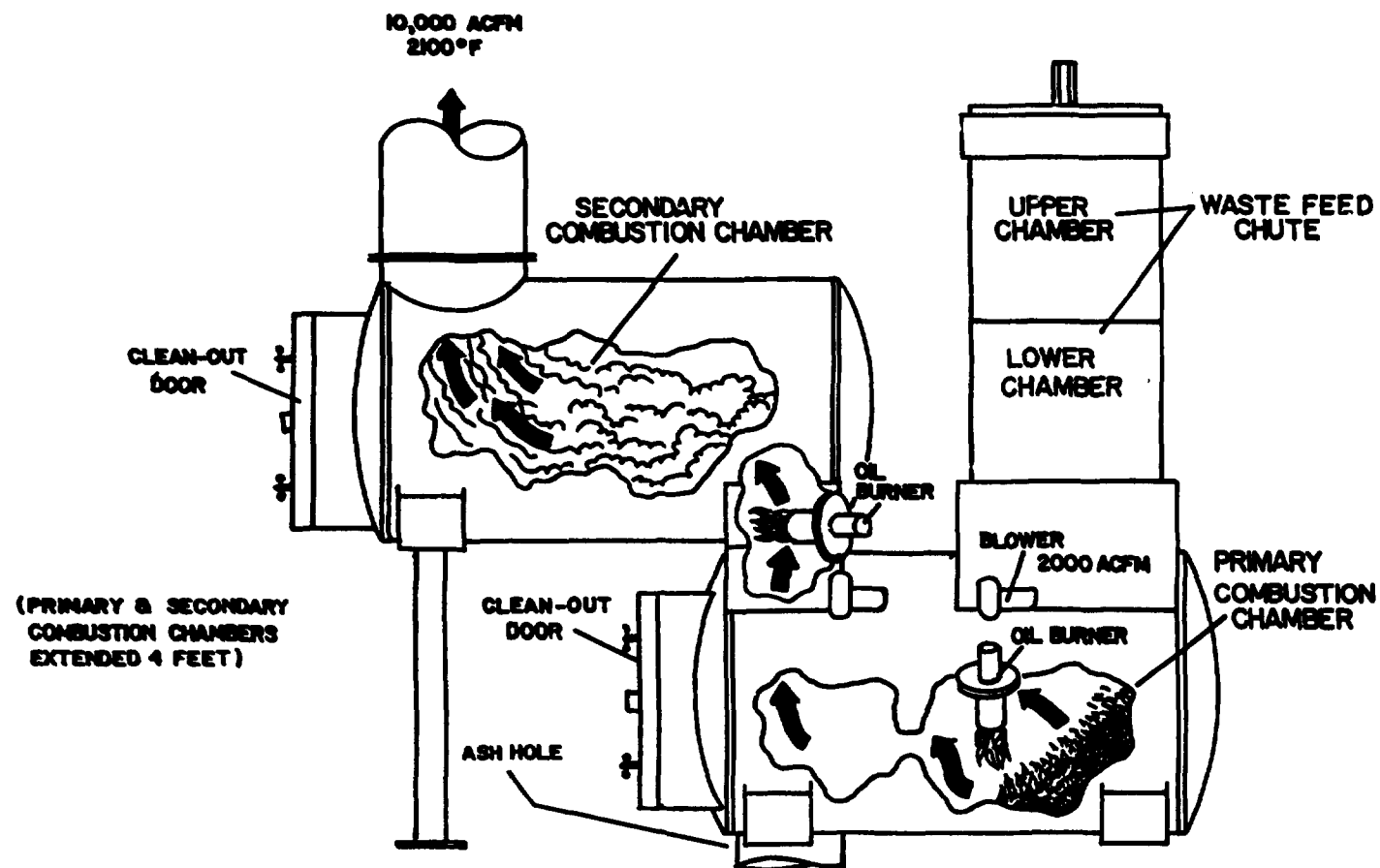


Figure 8 Controlled-Air Incinerator, cut-away view

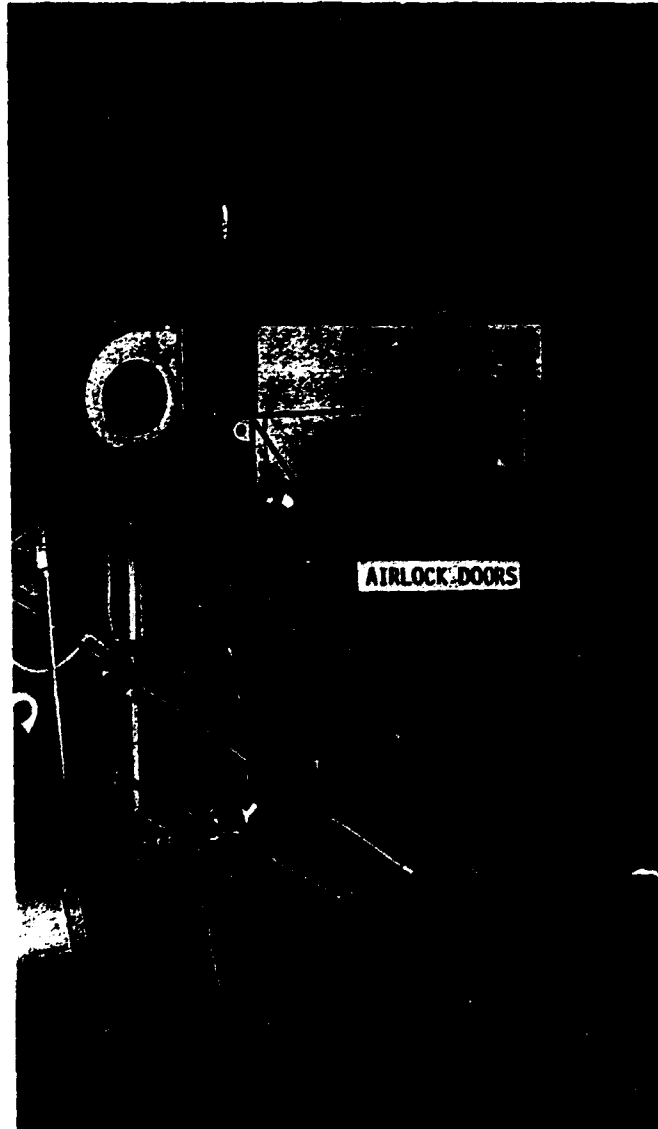


Figure 9 Controlled Air Incinerator showing Air-Lock Doors

A ram is provided in the bottom chamber that periodically pushes the ash into a hopper for storage and cooling. When cooled sufficiently, a slide gate will be opened allowing the ash to drop into drums located in the ash collection room in the east wing basement.

#### OFFGAS TREATMENT SYSTEM DESCRIPTION

Exhaust gases from the incinerator, furnaces, and exhaust from other basement activities are collected for treatment and filtration in a common system prior to discharge from the facility (see Figure 10). The integrated system, however, can be divided into several subsystems.

##### Incinerator Offgas Treatment

Combustion gases leave the incinerator at approximately 2000°F (1093°C) and are immediately cooled at 1400°F (760°C) using dilution air from outside the facility. Further cooling of the combustion gases is accomplished using a tube-and-shell air-to-gas heat exchanger, with gases leaving the heat exchanger at approximately 750°F (399°C). This two-step cooling process was selected to minimize thermal stress problems across the heat exchanger caused by the large difference in temperatures and reduce high temperature corrosion problems. The heat exchanger is able to handle the heat load imposed by burning 12,000 Btu/lb waste at the design rate of 400 lb/hr. Excess process heat is removed from the heat exchanger and incinerator room using independent exhaust systems and exhausted through a separate stack.

##### Furnace Offgas System

The induction furnaces installed in the facility have been fitted with close-capture fume hoods for the removal of smoke and fumes generated during the melting process. These are designed to operate at 2500 cfm and capture all of the smoke and fumes generated during melting. Further downstream, the ducting has been designed to provide a drop-out point for larger particulate through a reduction in velocity and cyclonic action. The air from the furnaces, along with the sorting/sizing room exhaust, joins directly with the main exhaust stream, reducing the incinerator offgas temperature to about 350°F (177°C).

##### Sorting/Sizing Room Ventilation

At present, air, smoke, and fumes are vented from the plasma-arc cutting room through a precipitator to the main exhaust stream. While inclusion of an electrostatic precipitator was not originally planned, review of the literature on plasma arc torch testing pointed out the necessity for a precipitator to preclude a rapid filter buildup of particulates.

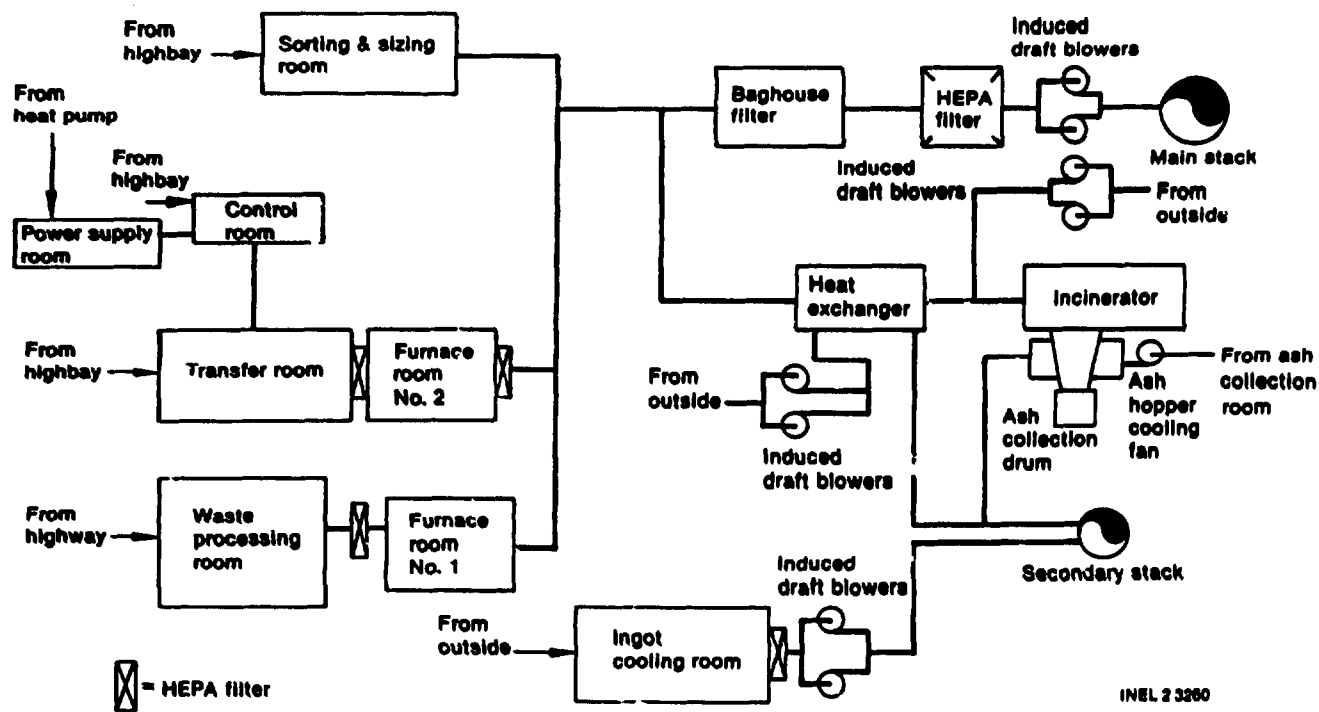


Figure 10 WERF Ventilation Schematic

### Baghouse Filter

A large capacity (16,000 cfm) baghouse filter, manufactured by Mikropul, Inc., is installed in the main exhaust stream prior to the HEPA filters. This filter eliminates much of the dust and particulate (99%) that would otherwise be deposited on the HEPA filters, resulting in longer filter life and less frequent changeout.

Air and gases introduced into the baghouse are filtered by fabric bags and the particulate matter is deposited on the outside of these bags forming a cake. Up to a point, the buildup of this cake results in an increase in filter efficiency. At a predetermined differential pressure, as sensed by a photohelic gauge, a cleaning cycle is initiated that pulses a reverse flow of compressed air through the bags, row by row, until the pressure drop is reduced to an acceptable level. In this manner, the baghouse is self-cleaning, with the resultant particulate material dropping to the hopper for subsequent removal. Since the dust loading for this facility is very low in comparison to other industries using baghouse filters, this cleaning sequence is repeated infrequently.

The bag filters are fabricated of Huyckglas, a fire resistant material which can withstand temperatures up to 500°F (260°C). In order to preclude the possibility of a fire in the baghouse, spark arrestors have been installed upstream of the filter.

### HEPA Filter Unit

A high-efficiency particulate (HEPA) absolute filter is installed downstream of the baghouse to act as a final filter for the facility exhaust. The efficiency of the HEPA filter is 99.97% for removal of particles of 0.3 micron mass median diameter (MMAD). The efficiency is higher than 99.97% for particles larger than 0.3 microns.

The HEPA filters are designed to operate at temperatures up to 500°F (260°C) and to withstand a pressure loading of 10 in w.g. across the entire face of the filter for short periods without permanent damage.

### Stack Effluent Monitoring

A continuous air monitor is installed to constantly sample the air released from the facility through the main stack. This sampling is not accomplished isokinetically since the stack release rate is reasonably constant. The rate of sample flow can be varied in the range of 10-100 liters per minute. The sample is drawn through 20 ft (6.1 m) of 3/8 in. (0.95 cm) tubing. Indication of the airborne radioactivity of the main stack effluent is displayed on the front panel of the unit in the highbay, and a remote alarm is provided in the control and observation room. In addition, the filter is routinely removed,

replaced by a new filter, and analyzed for gross radioactivity using a simultaneous alpha and beta-gamma smear counter.

#### PROJECT STATUS

Construction of the sizing and melting portion of the WERF including the baghouse, main HEPAs, both stacks, the wing to house the incinerator, and the main exhaust ducting is complete.

The Safety Assessment Document (SAD) has received EG&G's internal approval and is presently undergoing DOE review. The Operational Safety Requirements (OSRs) are completing EG&G internal review. The operating and maintenance procedures for the sizing and melting operations have been written and approved.

Initial startup and checkout operations for sizing and melting were initiated mid-August on cold materials and will probably require several months.

The incinerator Title II design has been completed and the incinerator order was placed with Environmental Control Products. Delivery was made on August 16th and 17th, 1982.

#### FUTURE DEVELOPMENT

As stated in the introduction, the WERF has two primary goals. The first is to increase the useful life of the RWMC. The second goal is to provide a full scale facility to perform development activities relating to low level waste handling and processing. Some of the potential development activities include:

1. Studies of actual partition ratios for radioactive contaminants in metallic waste.
2. Studies of temperature dependence of partition ratios.
3. Studies on refractory affinity for radionuclides.
4. Studies on slagging/fluxing techniques.
5. Development of alternate off-gas filtering/treatment systems.
6. Development of ash solidification processes.
7. Development of production scale capability to recycle metal by contamination removal during melting.
8. Development of remote systems for handling, sizing, melting and incineration of wastes and related processes.



9. Development of processes to dispose of non-nuclear hazardous wastes.

Operation of the WERF will provide an assessment of the desirability and economic feasibility of establishing similar capabilities elsewhere. Results of WERF operation and development activities will undoubtedly benefit the nuclear industry as a whole, especially in the areas of decontamination and decommission work.

*Aug*

## EVALUATION OF A JOULE-HEATED MOLTEN GLASS FURNACE SYSTEM FOR WASTE PROCESSING

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### ABSTRACT

The joule-heated glass furnace system is currently being evaluated at Mound as a means of reducing the volume of low-level radioactive waste similar to that found in light water reactor facilities. The furnace utilizes molten soda-lime silica to initiate and support combustion of the waste feed and to serve as an immobilization matrix. Dry solid waste, filter sludge, cartridge filters, and ion exchange resin are being introduced in a series of nonradioactive experiments to determine the combustion efficiency of the furnace unit. Results of experiments on the incineration of dry solid waste have shown excellent combustion capabilities. Upon completion of the nonradioactive studies, radioisotope behavior in the burn chamber will be investigated.

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### INTRODUCTION

The use of a joule-heated glass furnace for treatment of low-level Light Water Reactor (LWR) waste was conceived as a method for combining volume reduction and immobilization in one step. For the past 1-1/2 years, Mound has been preparing and evaluating a commercially available joule-heated glass furnace unit, coupled with a wet scrubbing system, in support of milestones B and G of the Defense Low Level Waste Management (LLWM) Program. The purpose of the glass furnace evaluation project is to advance and document incinerator technology for such combustibles as solids, resins, and sludges, and to develop a stable waste form for subsequent disposal.

---

\*Mound is operated by Monsanto Research Corporation for the U.S. Department of Energy under Contract No. DE-AC04-76-DP00053.

## FURNACE SYSTEM DESCRIPTION

The glass furnace unit at Mound (Figure 1) is a 8-1/2 x 5-1/2 x 3-1/2 ft burn chamber sheathed in a 1/4 in. 310 stainless steel skin lined with 9 to 12 in. of an aluminum zirconium silicate fused-cased refractory. This chamber is filled to a depth of 10 in. ( $\sim 15 \text{ ft}^3$ ) with soda-lime silica cullet. The cullet is initially heated to its melting point ( $1350^\circ\text{F}$ ) by the propane burner shown in Figure 2, after which the glass is able to conduct electricity, and the six high purity iron electrodes, located below the glass surface, assume control of the melt temperature. When the melt has reached a temperature of approximately  $2400^\circ\text{F}$ , waste is introduced into the burn chamber via the feed hopper/screw feeder, and immediately ignites due primarily to radiation from the glass. Primary combustion products are swept along the surface of the melt by combustion air streams. Ash from the combustion process falls to the glass surface where it is eventually incorporated into the melt, while the gaseous combustion products exit the furnace chamber by means of the offgas tee, and continue on to the wet scrubbing system.

The wet scrubbing system being used with the glass furnace was originally developed for use on Mound's cyclone incinerator. In that capacity, it has been providing environmentally safe offgas by effectively removing particulates, acid gases, and radioactivity, as well as reducing offgas temperatures (Klingler, 1981). Figure 3 is a schematic of the offgas scrubbing system. As offgas exits the glass furnace, it travels directly to the deluge tank where it is sprayed with caustic solution for acid, heat, and large particle removal. The gas then travels to a venture scrubber/cyclone demister for small particulate removal. Two stages of HEPA filtration provide backup protection.

## NONRADIOACTIVE EXPERIMENTATION

### WASTE CHARACTERIZATION STUDIES

Waste characterization studies were conducted to correlate wastes generated by LWR facilities with the glass furnace's potential use (Alexander, 1981). It was found that the major limitation effecting the suitability of the waste form is the fact that soda-lime silica can incorporate only minute quantities of sulfate, borate, and chloride. Should large quantities of any of these substances be added to the furnace, they would tend to pool on the top of the melt and affect the efficiency of the furnace until they eventually boil away.

As a result of these studies, four waste types were selected for investigation in the nonradioactive experiments: 1) dry solid waste composed of paper, plastics, rubber, and cloth, 2) ion exchange resin of both the anionic and cationic type, 3) filter sludge composed of diatomaceous

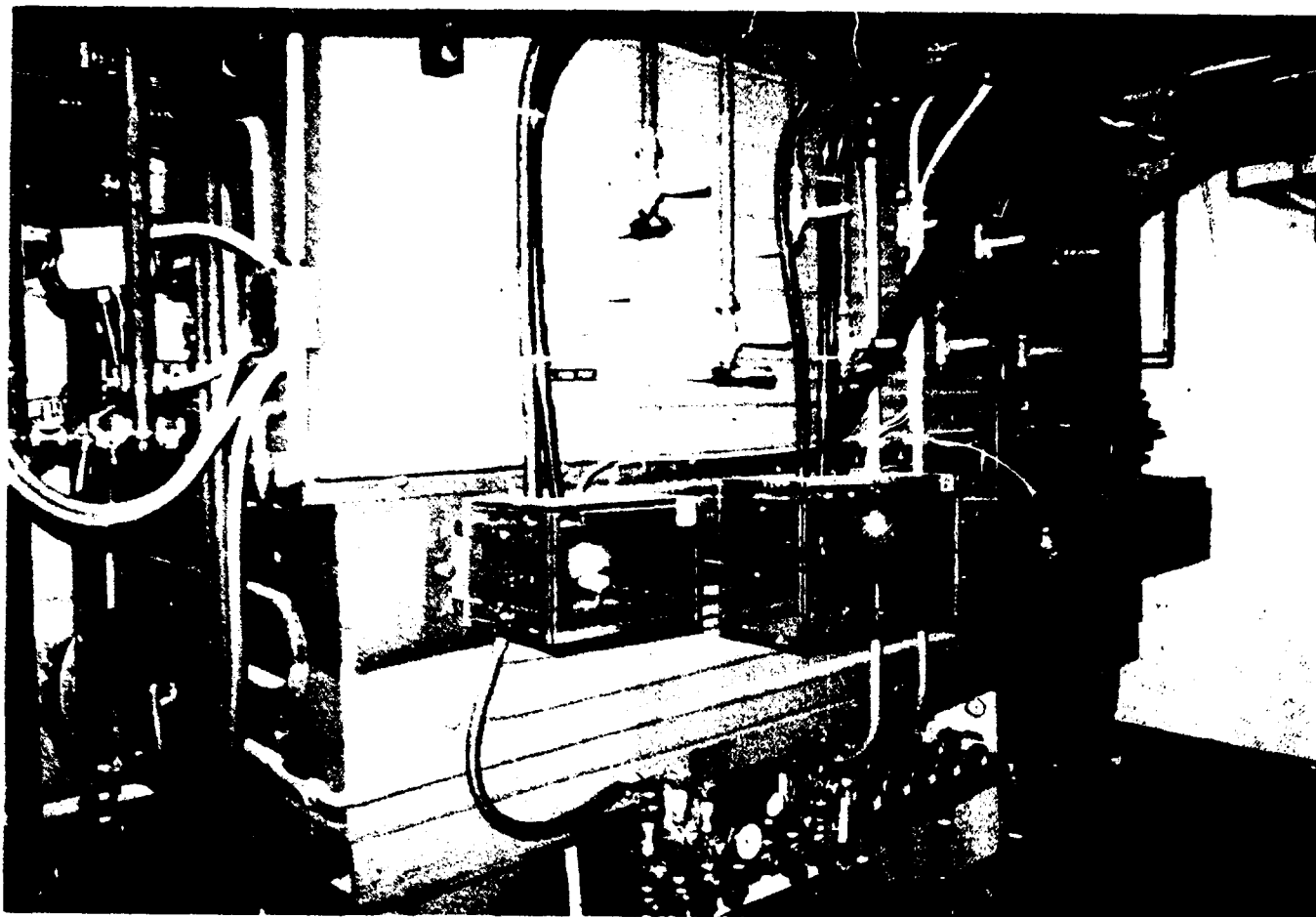


FIGURE 1 - Joule-heated glass furnace unit under evaluation at Mound.

**FIGURE 2**  
**FURNACE CHAMBER PROVIDES RESIDENCE TIME**  
**AT HIGH TEMPERATURES**

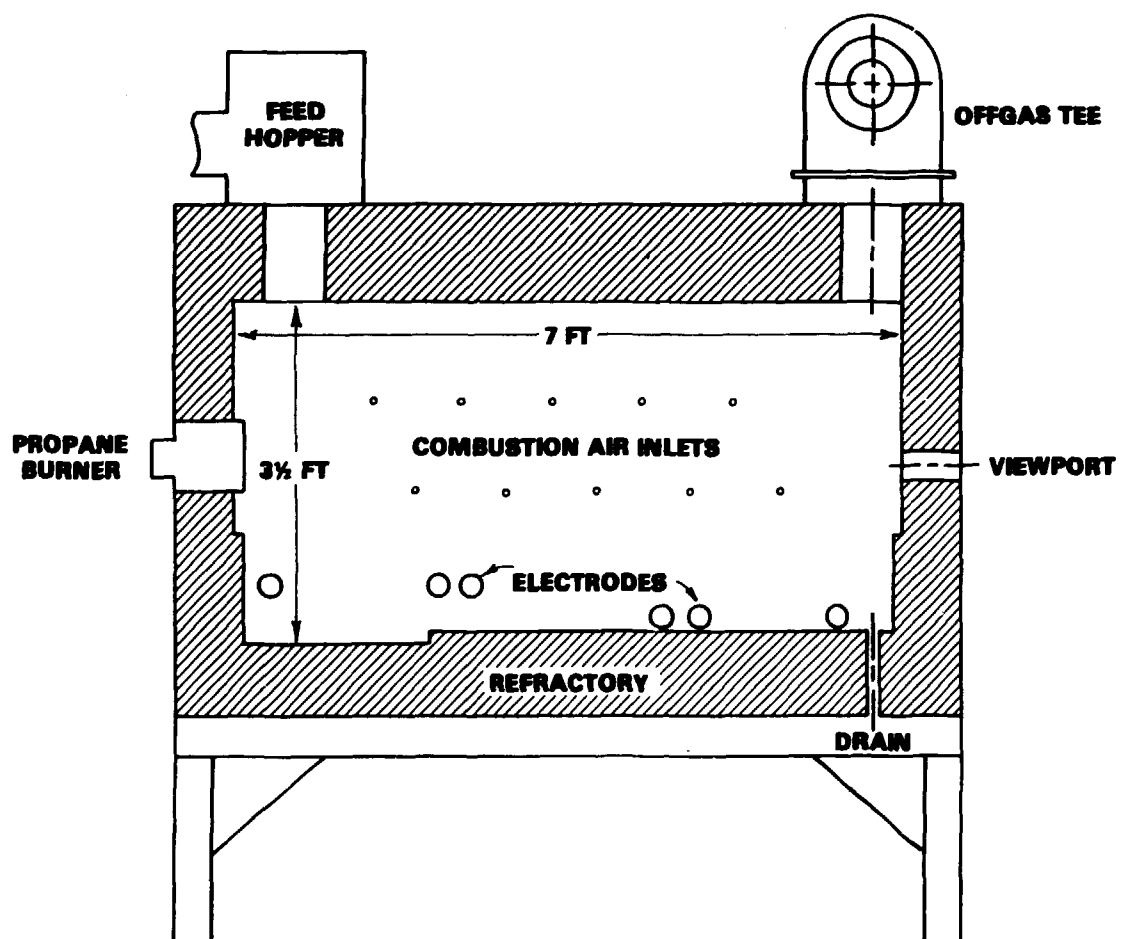
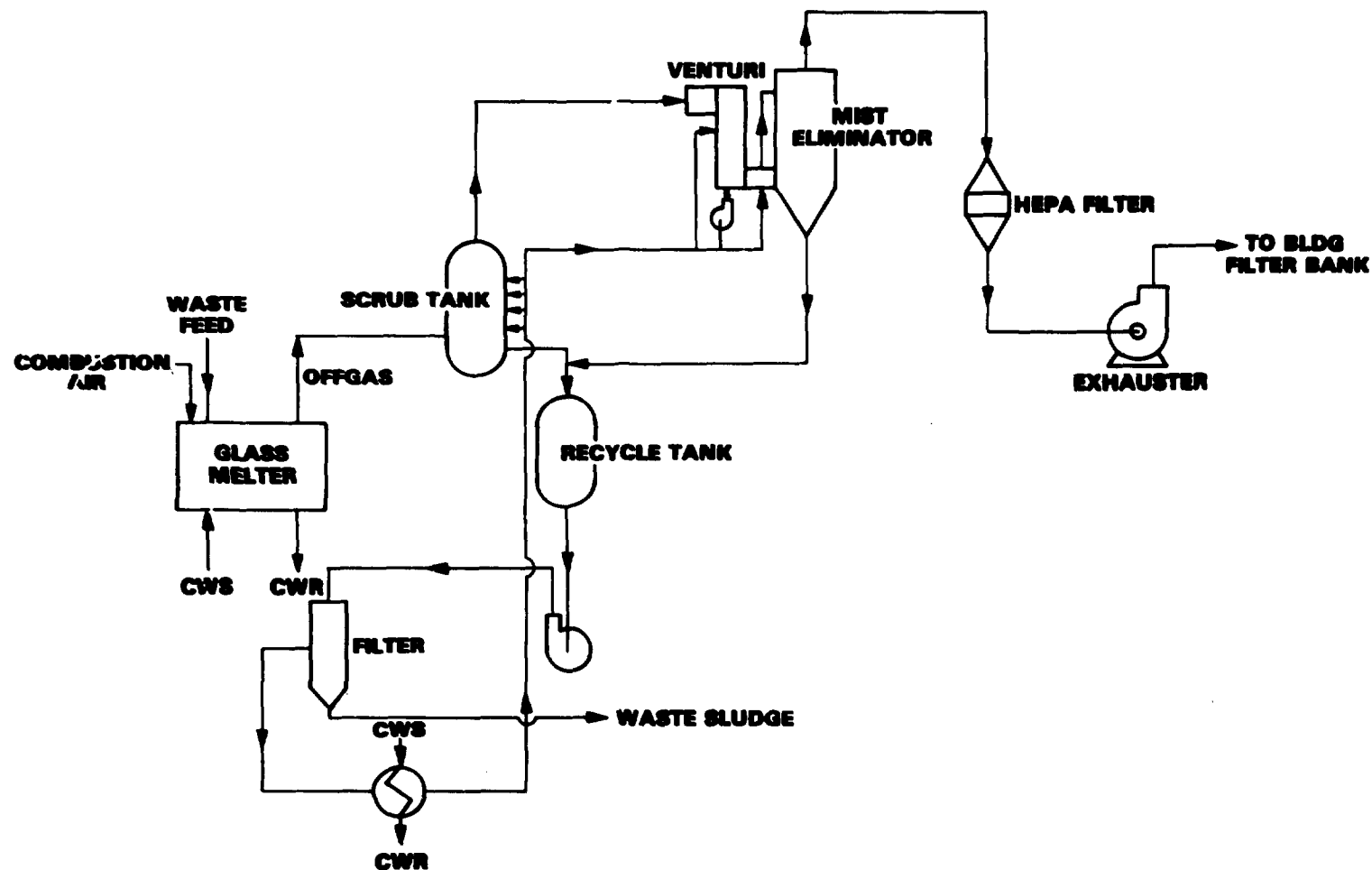


**FIGURE 3**  
**WET SCRUBBING SYSTEM PROVIDES ADEQUATE TREATMENT OF MELTER EXHAUST**



earth, organic cellulosic filter aid, and powdered ion exchange resin, and 4) cartridge filters having glass and plastic filter surfaces and non-metallic cores.

#### COMBUSTION EFFICIENCY EXPERIMENTATION

The experimental matrix developed for the nonradioactive experiments (see Table 1) focuses on the determination of the combustion efficiency of the furnace unit. This matrix is being used for each waste type with feedrate and percent excess oxygen values chosen to ascertain operating limits. Final results will provide a means of evaluating for equal variance at each set of conditions and discerning whether the combustion efficiency can be modeled as a linear function of feedrate and total oxygen.

Offgas monitoring capabilities have provided the means for achieving an extensive range of data. ORSAT gas analysis methods, EPA Method 5 sampling, gas chromatography, and mass spectrometry are used to determine CO, CO<sub>2</sub>, N<sub>2</sub>, O<sub>2</sub>, hydrocarbon, and particulate concentrations exiting the furnace during incineration. Current data, acquired upon completion of the experimental work involving dry solid waste, confirm that excellent combustion occurred at feedrates up to, and including, 50 lb/hr for as low as 30% excess oxygen (see Table 2). Higher feedrates for dry solid waste have not been investigated due to rate limits on the present feeding mechanism.

Experiments in progress involve burning filter sludge at feedrates of 50, 100, and 150 lb/hr. The elevated heat requirements on the system for the evaporation of the moisture content of sludge appears to be the limiting factor on the furnace in this case. Water streams, of up to 200 lb/hr, introduced to the furnace prior to sludge burning experiments, provided insights on furnace behavior under high moisture loading. Plenum temperatures tend to decline rapidly because the melt is unable to replenish energy absorbed by the evaporation process for water rates greater than 180 lb/hr. The actual combustion occurring during incineration of filter sludge is on a much lower scale than that encountered for dry solid waste, and is not expected to be a problem.

#### GLASS DURABILITY STUDIES

To determine the optimum waste loading for the soda-lime silica employed in the glass melter, samples of the glass were obtained with loadings of up to 100 wt % of a simulated ash/sludge mixture (for composition see Table 3). These samples were subjected to leach testing using the MCC-1 Static Leach Test Method. Three-, 7-, 14-, and 28-day tests were run using water as the leachant and maintaining a temperature of 90°C. Figures 4 and 5 show the normalized elemental mass loss for two of the major leach-rates, silica and sodium, at various waste loadings as a function of time.

**TABLE 1**  
**NON-RADIOACTIVE EXPERIMENTATION DETERMINES**  
**COMBUSTION EFFICIENCY**

**EXPERIMENTAL MATRIX**

| <b>SET</b> | <b>MORNING CONDITIONS</b> |                                   | <b>AFTERNOON CONDITIONS</b> |                                   |
|------------|---------------------------|-----------------------------------|-----------------------------|-----------------------------------|
|            | <b><u>FEEDRATE</u></b>    | <b><u>TOTAL O<sub>2</sub></u></b> | <b><u>FEEDRATE</u></b>      | <b><u>TOTAL O<sub>2</sub></u></b> |
| 1          | LOW                       | LOW                               | INTERMEDIATE                | INTERMEDIATE                      |
| 2          | HIGH                      | LOW                               | HIGH                        | HIGH                              |
| 3          | INTERMEDIATE              | INTERMEDIATE                      | LOW                         | HIGH                              |
| 4          | HIGH                      | HIGH                              | INTERMEDIATE                | INTERMEDIATE                      |
| 5          | LOW                       | HIGH                              | LOW                         | LOW                               |
| 6          | INTERMEDIATE              | INTERMEDIATE                      | HIGH                        | LOW                               |



**TABLE 2**  
**DRY SOLID WASTE INCINERATION EXHIBITED**  
**EXCELLENT COMBUSTION**

| <b>FEEDRATE #/HR</b><br><b>% EXCESS AIR</b> | <b>OFFGAS SPECIES (WT %)</b> |                       |                      |                      |                     |
|---------------------------------------------|------------------------------|-----------------------|----------------------|----------------------|---------------------|
|                                             | <b>CO</b>                    | <b>CO<sub>2</sub></b> | <b>O<sub>2</sub></b> | <b>N<sub>2</sub></b> | <b>HYDROCARBONS</b> |
| 20/30                                       | 0                            | 2                     | 16.3                 | 81.7                 | 0                   |
| 20/90                                       | 0                            | 8.34                  | 9.6                  | 80.5                 | 0                   |
| 35/60                                       | 0.2                          | 6                     | 12.6                 | 81.2                 | 0                   |
| 50/30                                       | 0                            | 6.2                   | 12                   | 81.8                 | 0                   |
| 50/90                                       | 0                            | 4.4                   | 15                   | 80.4                 | 0.2                 |

**TABLE 3**  
**ANALYZED COMPOSITION OF 0 TO 100% WASTE GLASS**

| <b>OXIDE</b>                   | <b>0%</b> | <b>10%</b> | <b>20%</b> | <b>30%</b> | <b>40%</b> | <b>100%</b> |
|--------------------------------|-----------|------------|------------|------------|------------|-------------|
| SiO <sub>2</sub>               | 74.8      | 73.4       | 70.1       | 68.8       | 58.7       | 39.0        |
| CaO                            | 6.15      | 5.9        | 5.5        | 5.4        | *          | *           |
| MgO                            | 3.80      | 3.5        | 3.4        | 3.3        | *          | *           |
| Na <sub>2</sub> O              | 14.49     | 14.5       | 14.2       | 13.8       | 13.6       | 12.5        |
| Al <sub>2</sub> O <sub>3</sub> |           | 1.0        | 3.5        | 4.6        | 12.1       | 14.6        |
| TiO <sub>2</sub>               |           | 0.04       | 0.3        | 0.5        | 1.4        | 3.5         |

\*VALUES NOT OBTAINED

**FIGURE 4**  
**NORMALIZED ELEMENTAL MASS LOSS FOR SODIUM**  
**AS A FUNCTION OF TIME**

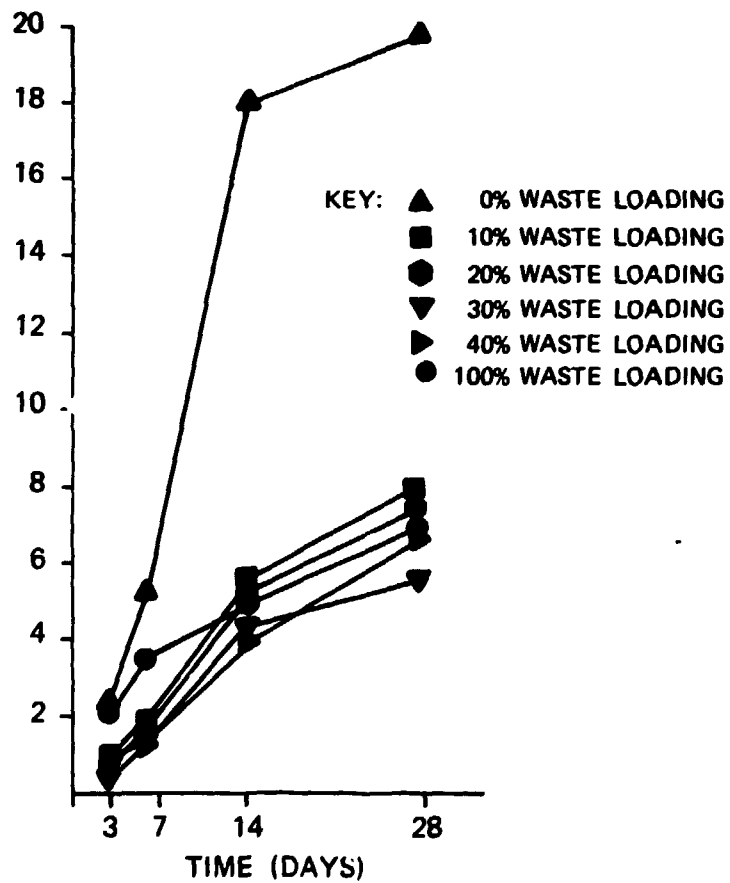
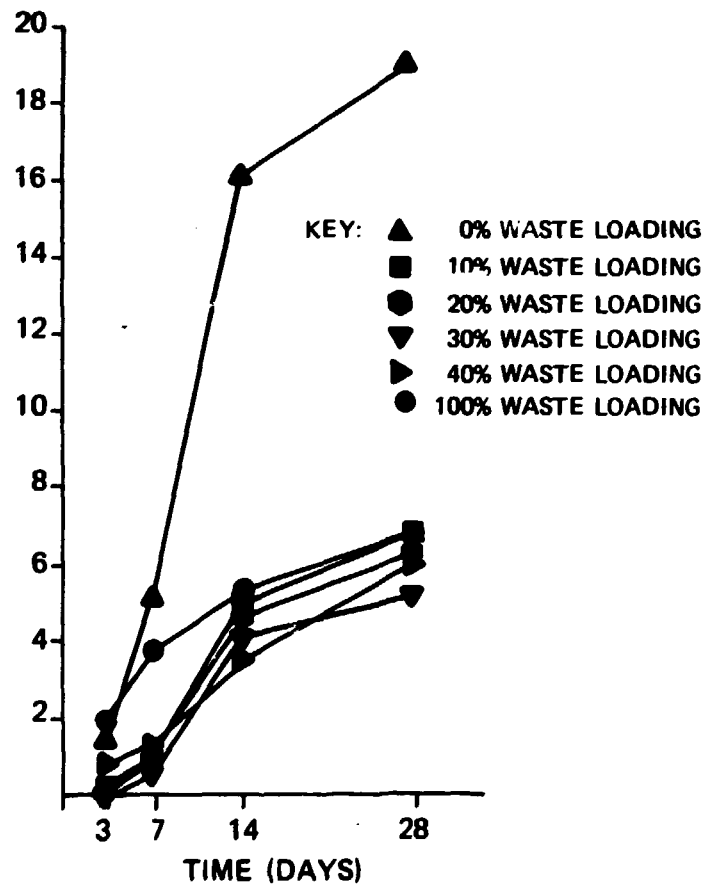


**FIGURE 5**  
**NORMALIZED ELEMENTAL MASS LOSS FOR SILICA**  
**AS A FUNCTION OF TIME**



From these figures it is seen that higher waste loadings show greatly improved durability from that of the original soda-lime silica (0%). This is believed to be caused by the high aluminum concentration in the waste, which essentially transforms the soda-lime glass into an alumina-silicate with correspondingly higher durability.

## COMPATIBILITY STUDIES

With the objective of determining corrosion caused by waste loading of the soda-lime silica, compatibility studies were conducted in which fingers of the alumina zirconium silicate fused-cast refractory, identical to those used in the furnace, were suspended in batch melts of 0, 10, 20, 30, and 40 wt % waste loading at 2400°F for one week. Figure 6 shows the corrosion, in inches, which occurred at the interface between the molten glass and air. Here again, as in the durability studies, the higher waste loading provided the best results; corrosion was negligible at 40% waste loading.

The temperature dependence of the corrosion caused by 0% waste loading of soda-lime silica on the refractory was also investigated, as shown in Figure 7. With an increase in temperature to 2650°F, corrosion more than tripled. As a result, incineration and idle temperature will be maintained at, or below, 2400°F. It was also found during the course of this work that higher waste loadings require temperatures in the 2600°F range to lower the viscosity enough to facilitate pouring. Thus, it may be necessary to raise the melt temperature for furnace draining; however, this will be accomplished in as short a time as possible to minimize the negative effects of such an action.

## PROBLEMS

Several problems have been identified during the evaluation of the furnace unit. One major concern is the rate at which ash is incorporated into the glass matrix. Several hours seem to be required before convective currents in the melt completely dissipate the ash after a waste burn. Further investigation, such as long-term continuous burning, will be required to delineate the problem. We are also considering changes in furnace design that may significantly improve the dissolution rate. Another problem, which will require redesign at a later date, is inadequate drainage. Draining the glass furnace is accomplished by melting the glass plug formed in the drain. To do this, the cooling water that ensures maintenance of the plug must be turned off, and the drain's resistance heater must be switched on. Current drain design allows drain temperature of only 1500°F to be achieved, resulting in a very slow exiting glass stream. It will be necessary to redesign the drain to allow temperatures of at least 2400°F to be achieved.

**FIGURE 6**  
**REFRACTORY/GLASS COMPATABILITY STUDIES**  
**REVEALED INCREASED REFRACTORY LIFE AT HIGHER**  
**GLASS WASTE LOADINGS AND LOWER TEMPERATURES**

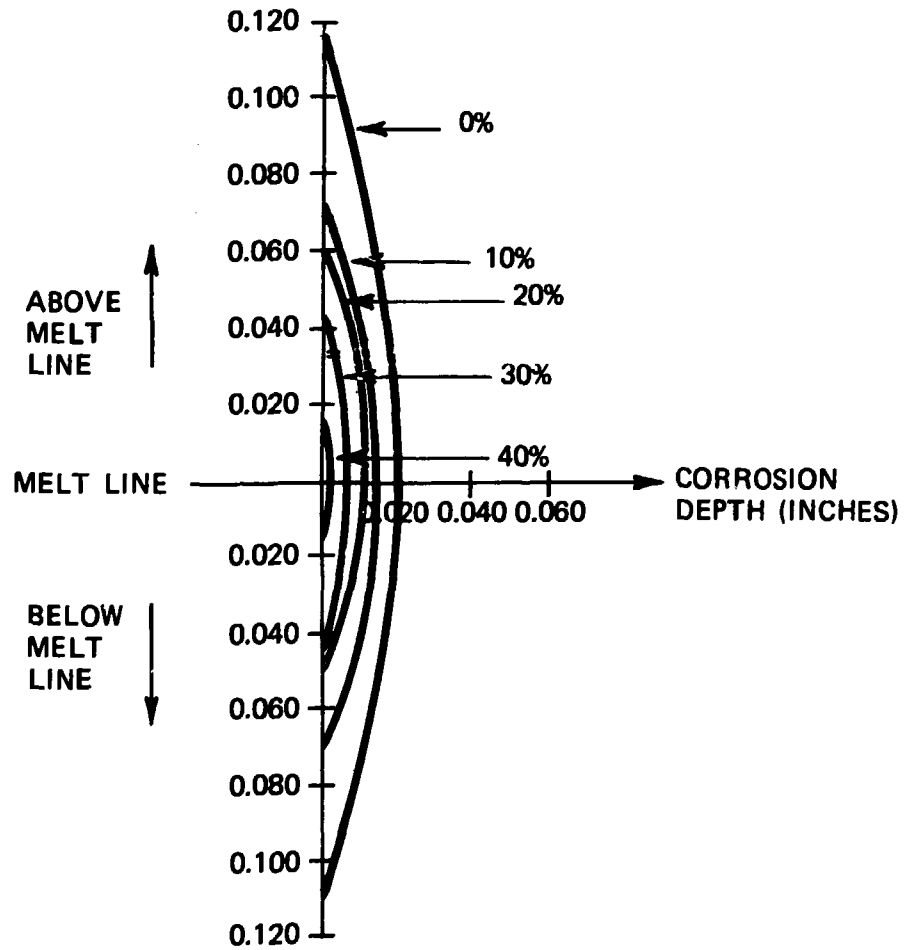
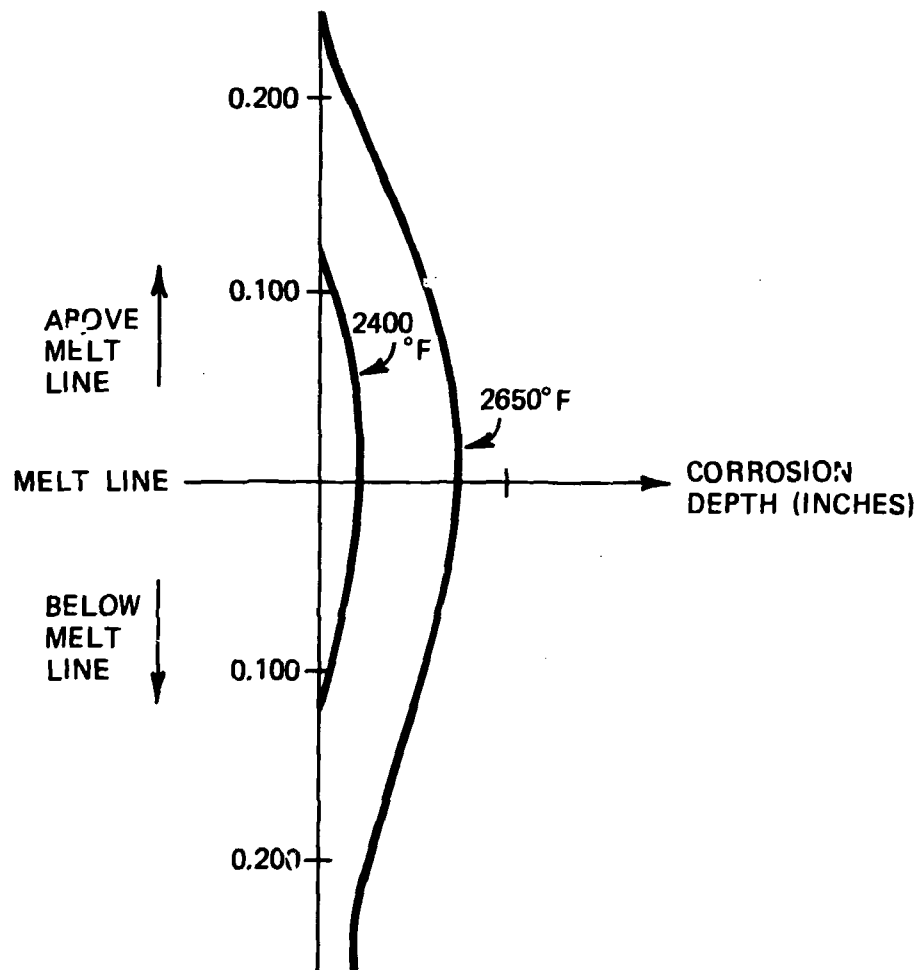


FIGURE 7  
REFRACTORY CORROSION AS A FUNCTION OF  
TEMPERATURE (0% WASTE LOADING)



## RADIOACTIVE EXPERIMENTATION

Upon completion of the nonradioactive studies, modifications will be made to the furnace system to supply the necessary shielding and enclosures for the next phase of activity, which will focus on radioisotope behavior in the burn chamber. At this time, preliminary studies will be made using waste spiked with stable isotopes of cesium and cobalt to simulate a radioactive situation. Data from these studies will be used as a basis for conducting tests using spikes of cobalt-60, cesium-137, and iodine-131, as well as possible burning of actual LWR facility waste.

## CONCLUSION

The evaluation of the joule-heated glass furnace system currently under way at Mound for the volume reduction and immobilization of low-level waste has focused on the determination of combustion efficiency for various waste forms, durability of the soda-lime silica as an immobilization matrix, and corrosion resistance of the furnace refractory. For the incineration of dry solid waste, the glass furnace exhibited excellent combustion efficiency for feedrates of up to 50 lb/hr. Further experiments on the incineration of filter sludge, cartridge filters, and ion exchange resins are being completed. No major problems are anticipated.

The chemical durability of soda-lime silica under various waste loadings using the MCC-1 static leach test was evaluated. It was found that high waste loadings did not adversely affect leach rates.

Corrosion resistance of the furnace refractory was also evaluated under various waste loadings. It was found that the higher waste loadings gave better results with greatly decreased corrosion rates.

Future work will focus on radioisotope behavior in the furnace unit.

## REFERENCES

L. M. Klingler, Mound Cyclone Incinerator, Volume 1. Description and Performance, MLM-MU-81-72-0003, December, 1981.

B. M. Alexander and L. M. Klingler, Glass Furnace Project: October 1980-March 1981, MLM-2825, May 6, 1981.

INSTITUTIONAL INCINERATOR AT  
UNIVERSITY OF MARYLAND



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Low-Level Waste Management Program  
EG&G Idaho, Inc.  
Idaho Falls, Idaho 83415

ABSTRACT

A controlled air incinerator for low-level radioactive waste has been installed and tested at the University of Maryland, Baltimore Campus. EPA method testing indicates full compliance with environmental regulations. Licensing activities are now in progress. The State of Maryland is in the public comment period for the incinerator.

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BACKGROUND

To date, three reports have been issued on the incinerator at the University of Maryland.(1,2,3) The project began in August of 1979 when the commercial burial grounds were shutting down at a furious pace and continued capability to dispose of institutionally generated waste was in doubt. This disposal resource has not been interrupted in the intervening three years, but the cost of disposal for this waste has doubled and the distance required for transport to burial has increased sharply.

Many universities and hospitals generate significant volumes of radioactive waste containing only small amounts of activity.(4) An NRC staff paper (5) recommended incinerating a large component of this waste, liquid scintillation media. No technical reason exists for not incinerating most of the balance of this material. This project demonstrates the activities supporting an incinerator at one major university.

REGULATORY ENVIRONMENT

Maryland is an NRC agreement state. Thus, the state government takes on certain of the responsibilities that would normally fall to the NRC in regulating things nuclear. Among them is the regulation of licensees such as the University of Maryland. The subject incinerator falls under the Radiation Regulatory Authority of the state. Appendix A of the reference (3) is reproduced here, detailing the various



regulatory agencies that the incinerator falls under.

### TEST DATA

EPA Method 5 testing was performed on the incinerator this spring. The results showed that all nonradioactive emissions are within acceptable levels for the waste forms to be burned. Reference (3) describes the methodology of control of radioactive emissions from this system. Basically, the radiation safety office will decay short lived nuclides and survey the waste for significant hard gamma radiation that could potentially be in the waste. These physical controls, as well as other administrative controls, will be used to ensure compliance for radioactive release.

### PRESENT ACTIVITIES

As of this writing the University's incinerator license is under consideration by the state. A 45 day public comment period began on August 6, 1982. After the comment period closes on September 20, the regulators will decide whether significant public comment has been received and whether a public hearing is required. At that time the regulators will either issue the license or schedule a hearing.

A related matter on regulation is the size of the license fee to be paid by the University of Maryland. The regulator has the right to set the licensing fee at any level considered appropriate. The licensee has the right to appeal the fee. This matter is still in negotiation between the University and the state.

### REFERENCES

1. Use Plan for Demonstration, Radioactive Waste Incinerator, EGG 2192, L. R. Cooley, M. R. McCampbell, and J. D. Thompson.
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3. Current Practice of Incineration of Low-Level Institutional Radioactive Waste, EGG-2076; L. R. Cooley, M. R. McCampbell, and J. D. Thompson.
4. Interim Report: Low-Level Waste, Institutional Waste Incinerator Program, EGG-WM-5116; J. D. Thompson.
5. Study of Alternative Methods for Management of Liquid Scintillated Counting Washes, NUREG-0656; L. Roche-Farmer.

APPENDIX A  
REGULATORY REQUIREMENTS RELATED TO THE OPERATION  
OF A RADIOACTIVE WASTE INCINERATOR

Table A-1. Radioactive Waste Incinerator  
Items for Compliance

| <u>SUBJECT TO<br/>REQUIREMENTS</u> | <u>JURISDICTION</u>            |                                |                                                                                          |
|------------------------------------|--------------------------------|--------------------------------|------------------------------------------------------------------------------------------|
|                                    | <u>FEDERAL</u>                 | <u>STATE OF MD.</u>            | <u>LOCAL (Balto. City)</u>                                                               |
| <u>Environmental</u>               |                                |                                |                                                                                          |
| Air emissions monitoring           | EPA <sup>1</sup><br>EPA<br>EPA | HMH <sup>2</sup><br>HMH<br>HMH | Balto. City Health Dept.<br>(requirements same as<br>State of Maryland)                  |
| Other hazardous material equipment | EPA<br>EPA                     | HMH<br>HMH                     |                                                                                          |
| License, Permit, or Approval       | EPA                            | HMH                            |                                                                                          |
| <u>Radioactivity</u>               |                                |                                |                                                                                          |
| MPC in Air                         | NRC <sup>3</sup>               | Rad. Control <sup>4</sup>      | Baltimore<br>City Health<br>Department<br>(same requirements<br>as State of<br>Maryland) |
| Gaseous Effluent                   | NRC                            | Rad. Control                   |                                                                                          |
| Solid By-Product (ash)             | NRC                            | Rad. Control                   |                                                                                          |
| LS Wastes                          | NRC                            | Rad. Control                   |                                                                                          |
| Neighboring Community              | NRC                            | Rad. Control                   |                                                                                          |
| License, Permit, or Approval       | NRC                            | Rad. Control                   |                                                                                          |

<sup>1</sup> EPA - Environmental Protection Agency

<sup>2</sup> HMH - Department of Health and Mental Hygiene

<sup>3</sup> NRC - Nuclear Regulatory Commission

<sup>4</sup> Radiation Control - Division of Radiation Control (of Department HMH)

Table A-1. Radioactive Waste Incinerator  
Items for Compliance (continued)

| <u>SUBJECT TO<br/>REQUIREMENTS</u>                                 | <u>JURISDICTION</u>   |                                  |                                                |
|--------------------------------------------------------------------|-----------------------|----------------------------------|------------------------------------------------|
|                                                                    | <u>FEDERAL</u>        | <u>STATE OF MD.</u>              | <u>LOCAL (Balto. City)</u>                     |
| <u>Personnel</u>                                                   |                       |                                  |                                                |
| Operating Personnel                                                | NRC                   | Rad. Control                     | Baltimore<br>City Health                       |
| Workman's Comp.                                                    | none                  | HMH                              | Department                                     |
| Occupational Health                                                | none                  | none                             | (same requirements<br>as State<br>of Maryland, |
| Neighboring Community                                              | NRC                   | Rad. Control                     |                                                |
| License, Permit, or Approval                                       | NRC                   | Rad. Control                     |                                                |
| <u>Other</u>                                                       |                       |                                  |                                                |
| Requires Approval or<br>Acknowledgement<br>of Another Jurisdiction | NRC (state,<br>local) | Rad. Control<br>(federal, local) | Dept. of Environmental<br>Health, Div. of      |
|                                                                    | EPA (state, local)    | HMH (federal)                    | Air Pollution Control<br>(federal, state)      |

**Table A-2. Regulatory Requirements to be Met  
for the Operation of a Radioactive Waste Incinerator**

| ITEMS FOR COMPLIANCE            | REGULATORY AGENCY REQUIREMENT                         |                                                                               |
|---------------------------------|-------------------------------------------------------|-------------------------------------------------------------------------------|
| Environmental                   | DHMH                                                  | EPA                                                                           |
| 1. Necessary permits, approvals | To construct incinerator                              | (Concurrent with with State) (Balto. City approvals are concurrent with State |
|                                 | To operate incinerator (1 year)                       |                                                                               |
|                                 | *Designated Hazardous Substance permit                | **Permit to operate hazardous materials waste disposal                        |
|                                 | *A New Source Impacting a Non-Attainment Area (NSINA) |                                                                               |
| 2. Air - Emissions              | Particulate matter                                    | (EPA and Balto. City approvals are with State.)                               |
|                                 | Sulfur oxides                                         |                                                                               |
|                                 | Carbon monoxide                                       |                                                                               |
|                                 | Ozone                                                 |                                                                               |
|                                 | Nitrogen dioxide                                      |                                                                               |
|                                 | Lead                                                  |                                                                               |
|                                 | Hydrocarbons                                          |                                                                               |
|                                 | Visible emissions                                     |                                                                               |
|                                 | Organic solvents                                      |                                                                               |
| 3. Air - monitoring and testing | Monitoring of emissions and record keeping            |                                                                               |
| 4. Hazardous Materials          | Toluene, Xylene                                       | Toluene, Beryllium, Xylene,                                                   |
| 5. Equipment                    | 1) Air control devices<br>2) Exhaust ducts/chimneys   |                                                                               |

\*The State will evaluate whenever it is uncertain which permits are unnecessary.

\*\*The status of this type of permit is still uncertain. See page 2 of text.

**Table A-2. Radioactive Waste Incinerator  
List of Requirements by Regulatory Agency (continued)**

| <u>ITEMS FOR COMPLIANCE</u>                        | <u>REGULATORY AGENCY REQUIREMENT</u>                                                                                                                                                                                                                                                                                                                |                                                                  |
|----------------------------------------------------|-----------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------|------------------------------------------------------------------|
| <u>Radioactivity</u>                               | <u>Division of Radiation Control</u>                                                                                                                                                                                                                                                                                                                | <u>NRC</u>                                                       |
| 1. Permits or Approval Needed                      | To incinerate radioactive waste                                                                                                                                                                                                                                                                                                                     | To incinerate radioactive waste (Concurrent with State approval) |
| 2. Maximum Permissible Concentrations (MPC) in Air | Submit list of isotopes and calculations<br><br>COMAR, Part D, Appendix A Table II, MPC measured at stack exit                                                                                                                                                                                                                                      | (Concurrent with State approval)                                 |
| 3. Gaseous Effluent                                | Limits from stack<br><br>10% MPC limits for air                                                                                                                                                                                                                                                                                                     | (Concurrent with State approval)                                 |
| 4. Required Information                            | Incinerator characteristics<br><br>Ash monitoring procedures<br><br>Handling/disposing ash<br><br>Maximum number burns per week<br>Compliance and notification of NRC and local agencies<br><br>Specify nuclides<br><br>Specify waste forms<br><br>Specify volume to be burned<br><br>Location of incinerator with respect to neighboring buildings | Compliance and notification of State and local agencies          |
| 5. Personnel                                       |                                                                                                                                                                                                                                                                                                                                                     |                                                                  |
| Staff                                              | Procedures and prevention from overexposure                                                                                                                                                                                                                                                                                                         | (Same as with State)                                             |
| Occupational Health                                | None                                                                                                                                                                                                                                                                                                                                                | None                                                             |

Table A-3. Comments on Regulatory RequirementsEnvironmentalPermits and Approvals

## EPA

- o National Emissions Standards for Hazardous Air Pollutants (NESHAPS) - The UMAB is located in a nonattainment area; i.e. the ambient air quality has not met desired criteria as specified in National Resource Conservation Act; and because UMAB plans to incinerate the designated hazardous substances toluene and xylene. At present, no specific designation has been made in reference to incinerating xylene and toluene as liquid scintillation wastes.

## DHMH

- o Permit required to construct incinerator; allow 60 days for processing; construction must begin within one year of issuance.
- o Permit required to operate incinerator.
- o Approval may be required for all new incinerating sources located within a nonattainment area (NSINA).
- o Approval may be required under new source performance standards, depending on the capacity of incinerator.
- o Permit may be required for designated hazardous substances because hospital and laboratory wastes will be incinerated.

Air - emissions requirement**DHMH****1. Particulate matter:****a) Primary ambient air quality standards:**

- 1) 75 micrograms/m<sup>3</sup> annual geometric mean
- 2) 260 micrograms/m<sup>3</sup> maximum 24-hour concentration; not to be exceeded more than once per year

**b) Secondary ambient air quality standards: 150 micrograms/m<sup>3</sup> Maximum 24-hour concentration; not to be exceeded more than once per year.****c) 0.03 gram/SCFD (68.7 mg/dscm)****2. Sulfur oxides:****a) Primary ambient air quality standards (measured as SO<sub>2</sub>):**

- 1) 80 micrograms/m<sup>3</sup> (0.03 ppm) annual arithmetic mean.
- 2) 365 microgram/m<sup>3</sup> (0.140 ppm) maximum 24-hr concentration; not to be exceeded more than once per year.

**b) Secondary ambient air quality standards: 1300 mg/m<sup>3</sup> (0.50 ppm) maximum 3-hour concentration not to be exceeded more than once per year.****c) Maximum discharge to atmosphere: gases containing no more than 500 ppm sulfur dioxide.**

3. Carbon Monoxide:

a) Primary and secondary ambient air quality standards

- 1) 10 milligrams/m<sup>3</sup> (8 ppm), maximum 8-hour concentration not to be exceeded more than once per year
- 2) 40 milligrams/m<sup>3</sup> (35 ppm) maximum 1-hour concentration not to be exceeded more than once per year.

b) (Applicable specifically to Area III - Baltimore metropolitan area)  
Maximum rate of discharge of carbon monoxide gas: 500 pounds/day (227 kilograms) and maximum concentration of 12% by volume unless CO gas is burned with excess oxygen at 1300° F (704°C) or more for 0.3 second or longer in a direct flame afterburner of DHMH approved equivalent control method which reduces the concentration of CO in the effluent to 1% by volume or less. The control method shall be equipped with an indicating pyrometer positioned in the working area readily visible to the operator.

4. Ozone:

- a) The primary and secondary ambient air quality standard is 235 micrograms/m<sup>3</sup> (0.12 ppm).
- b) The standard is attained when the expected number of days/calendar year with maximum hourly average concentrations above 0.12 ppm is equal to or less than one.

5. Nitrogen Dioxide: The primary and secondary ambient air quality standard: 100 micrograms/m<sup>3</sup> (0.05 ppm) annual arithmetic mean.

6. Lead: The primary and secondary ambient air quality standard: 1.5 micrograms/m<sup>3</sup> maximum arithmetic mean averaged over a calendar quarter.



7. Hydrocarbons: The primary and secondary ambient air quality standard guideline: 160 micrograms/m<sup>3</sup> (0.24 ppm) maximum 3-hour concentration (6 a.m. to 9 a.m.) not to be exceeded more than once per year.
8. Visible Emission: No emissions visible to the human eye may be discharged.
9. Organic Solvents:
  - a) Cannot discharge any emissions of organic material in which organic solvent or organic solvent-containing material in direct contact with flame is in excess of 15 pounds per day unless discharge is reduced by 85% or more overall.
  - b) Exceptions:
    - 1) Volatile content of material is only water and organic solvents.
    - 2) Organic solvents comprise not more than 20% by volume of volatile content.
    - 3) The volatile content is not photochemically reactive as defined (this includes toluene).
    - 4) The use of any material, in any installation or building described in subsection B (1): Does organic solvent content exceed 20% by volume of the material.

#### Air - Monitoring and Testing

##### DHMH

Monitoring - The DHMH can require UMAB to install, use, and maintain monitoring equipment or employ other methods to determine quantity or quality of emissions discharged into the atmosphere and to maintain records and make reports of these tests to the Department according to its specifications.

Hazardous Material

## EPA

Comply with emission standards for hazardous materials as directed by this agency.

Equipment

## DHMH

1. Possible testing of equipment by Department for compliance ("Test Methods for Stationary Sources") with Maryland State Bureau of Air Quality and Noise Control, March, 1976).
2. Incinerator must be provided with openings (for emission testing methods) either in exhaust gas duct work, stack, or chimney.
3. Incinerator must be either
  - a) Crematory or pathological incinerator used to burn Type 4 waste (as defined by Incinerator Institute of America (IIA)) or medical wastes
  - OR
  - b. Pathological incinerator used to burn medical waste, which has been approved by the Department (this standard applies to the system at UMAB).

Radioactivity

## MPC in Air

## NRC

1. Averaged Over 24 hour Period - Gaseous effluent from incinerator stack will not exceed limits for air in Appendix B, Table II, 10 CFR Part 20.

2. Averaged Over One-year Period - Gaseous effluent from incinerator stack will not exceed 10% of limits for air in Appendix B, Table II, 10 CFR, Part 20\*.

### Licenses and Approvals

#### Division of Radiation Control

1. An amendment to an existing license or issuance of a separate license.  
Information Required:

- a. List of all radionuclides to be burned and calculations demonstrating air concentrations of effluents at the point of release (i.e. stack) are in accordance with COMAR, Part D, Appendix A, Table II.
- b. Incinerator characteristics (e.g. - stack height, air flow, proximity to neighboring buildings, rated capacity, etc.).
- c. Describe method of measurement or estimation of concentration of radioactive material appearing in ash residue.
- d. Describe procedures to be followed to prevent overexposure of personnel during all phases of the operation, including instruction to personnel handling the combustibles and the ash.
- e. State maximum number of burns to be performed in any one week and the maximum number of burns in any one year.
- f. Describe location of incinerator with respect to occupied buildings in vicinity. Include a sketch with distances, prevailing wind directions, and designation of "North".

\*Note: or Code of Maryland Regulations (COMAR) Part D, Appendix A, Table I.

- g. Provide evidence of compliance with or acknowledgement by NRC and local agencies concerning the incineration of radioactive material.**
- 2. NRC - Submit evidence that all state and local regulations concerning incineration of radioactive waste have been met by this institution.**

TABLE A-4.  
 INFORMATION REQUIRED FOR COMMISSION APPROVAL OF  
 TREATMENT OR DISPOSAL BY INCINERATION

Revised January 8, 1981

1. State specifically the isotopes you wish to incinerate. For each isotope listed, you should submit calculations demonstrating that air concentrations of the effluents at the stack are in accordance with the requirements of Section 20.106 of 10 CFR Part 20.
2. Submit the characteristics of the incinerator such as height of the stack, height of and distance to buildings in the surrounding areas, rated airflow of the incinerator in cubic feet per hour or similar units and its proximity to any air intake ducts.
3. The gaseous effluent from the incinerator stack should not exceed the limits specified for air in Appendix B, Table II, 10 CFR Part 20, when averaged over a twenty-four (24) hour period.
4. In order to be in compliance with the ALARA philosophy stated in Section 20.1(c) of 10 CFR Part 20, the gaseous effluent from the incinerator stack should be a fraction (approximately 10%) of the limits specified for air in Appendix B, Table II, 10 CFR Part 20, when averaged over a one year period.
5. Describe the method of measurement or estimation of the concentration of radioactive material appearing in ash residue.
6. Describe the procedures for handling and disposing of ash from the incinerator.
7. Describe procedures to be followed to prevent overexposure of personnel during all phases of the operation, including instruction given to personnel handling the combustibles and the ash.
- 8a. Submit evidence that all State and local regulations concerning incineration of radioactive material have been met by your institution.
- 8b. Submit evidence that all State and local jurisdictions have been notified of your plans to incinerate radioactive waste.
9. State the maximum number of burns to be performed in any one week and the maximum number of burns per year.

## COMMERCIAL INCINERATION DEMONSTRATION

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## ABSTRACT

Low-level radioactive wastes (LLW) generated by nuclear utilities presently are shipped to commercial burial grounds for disposal. Increasing transportation and disposal costs have caused industry to consider incineration as a cost-effective means of volume reduction of combustible LLW. Repeated inquiries from the nuclear industry regarding the applicability of the Los Alamos controlled air incineration (CAI) design led the DOE to initiate a commercial demonstration program in FY-1980. Development studies and results in support of this program involving ion exchange resin incineration and fission/activation product distributions within the Los Alamos CAI are described.

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INTRODUCTION

Since 1975, low-level waste transportation costs have increased by 50 percent and disposal fees have risen five-fold.(1) This adverse expenditure trend has caused many within the commercial nuclear industry to view advanced volume reduction techniques with renewed interest. Among the alternative technologies being considered, incineration of combustibles is recognized as one of the most effective methods for reducing both the mass and volume of waste shipments.

At present, several universities and hospitals are operating, installing, or have committed to install combustion processes to treat institutional wastes.(2) Nuclear utility commitment in this area, however, has been substantially less. Factors in this hiatus include cost, technical, and licensing uncertainties associated with reactor waste combustion systems. To fill this technical need and in response to continued, substantial utility interest in the Los Alamos controlled air incineration (CAI) system (developed for transuranic (TRU) waste treatment), DOE initiated this program in FY-1980 to support demonstration at a nuclear utility site. An architect-engineering firm has been selected by the DOE to prepare a licensing submittal for a LLW incinerator based on the Los Alamos CAI design at an operating commercial power reactor. The license application is scheduled to be submitted

for Nuclear Regulatory Commission review by the end of the second quarter of CY-1983.

An important objective of the program is to determine the key design and operating conditions necessary to convert the CAI from a TRU waste incineration system into one which can accept nuclear utility low-level wastes.

An important difference between the two waste types is the radio-nuclide makeup of each. Transuranic wastes generated by the defense program are contaminated with non-volatile heavy metal radioisotopes, primarily Pu-239. Power reactor wastes, on the other hand, contain a wide variety of fission and activation products of varying volatility. In addition, power reactors generate significant quantities of spent ion exchange resins. These resins have been identified as a challenging combustion problem in earlier tests in other systems. Data and results obtained from the incineration tests described in this paper will provide a basis for design and licensing of the CAI for incineration of LLW at the selected nuclear utility site.

#### COMMERCIAL INCINERATION DEMONSTRATION PROGRAM

The goal of this program is demonstration of CAI/LLW technology at a nuclear utility site. Program objectives contributing to this goal include resolution of technical uncertainties through development studies at Los Alamos, preparation of a design specifically adapted to nuclear utility LLW handling and treatment requirements, NRC licensing of the process, and installation/operation at a utility site. The selected project approach is a DOE-sponsored joint effort involving participation of Los Alamos, the nuclear utility which will serve as demonstration site, and a subcontractor to serve as interface in design, licensing, fabrication, and installation activities.

#### PROCESS DESCRIPTION

The CAI/LLW demonstration program has expanded the capabilities the transuranic waste incineration process shown in Figure 1. The CAI/TRU system which has been described in earlier reports (3,4) is based primarily on commercially available equipment components.

Much of the equipment is enclosed to provide containment of plutonium to meet health and safety standards for transuranic materials. As demonstrated with Pu-239 contaminated solid wastes, the CAI/TRU process consisted of four major subsystems: solid feed preparation, incineration, offgas cleanup, and scrub solution recycle. A liquid/slurry feed system and high intensity liquid burner have since been incorporated into the process to support LLW and hazardous waste incineration studies. The heart of the system is a dual-chamber controlled air

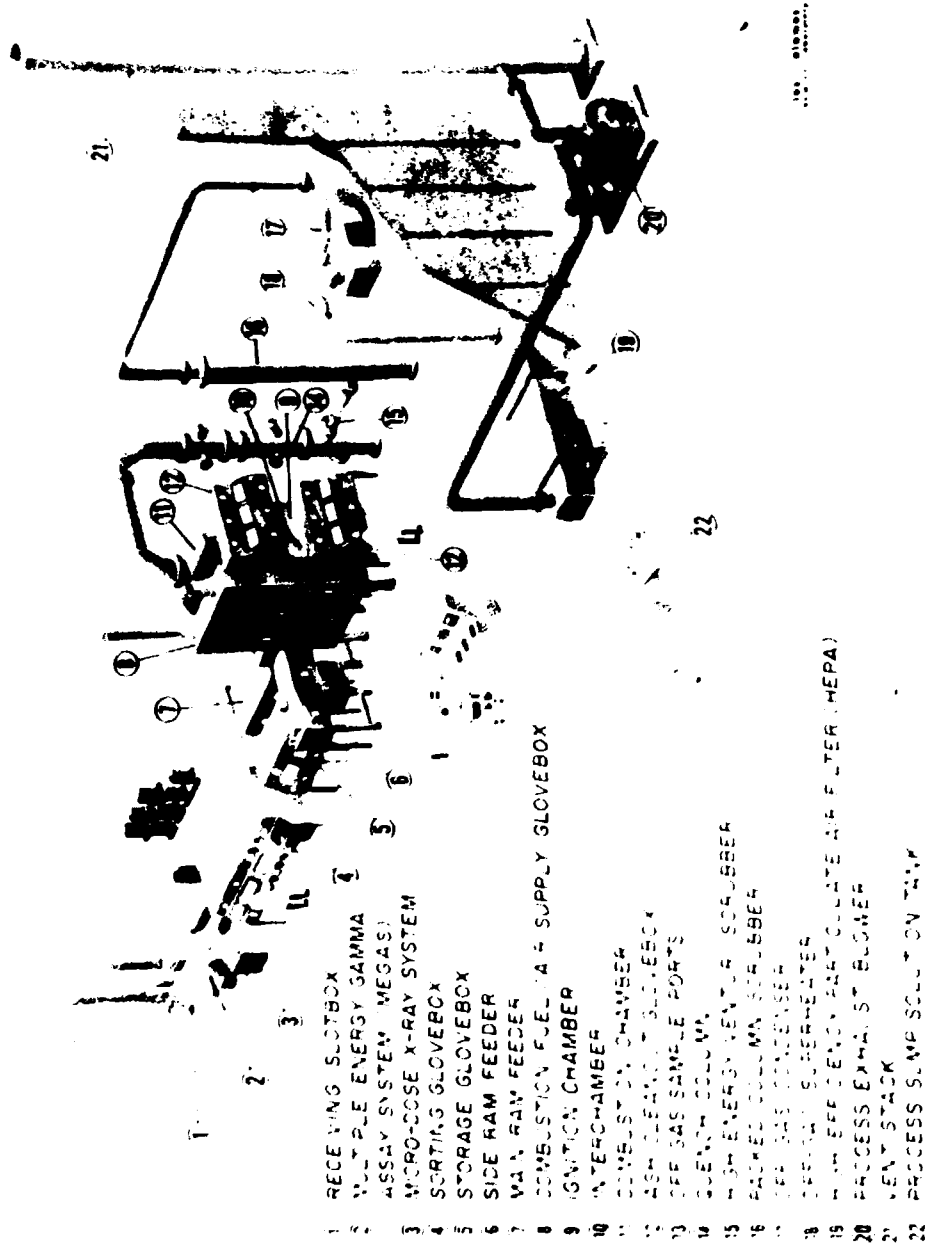


Fig. 1. CONTROLLED-AIR TRANSURANIC WASTE INCINERATION PROCESS



incinerator which exhausts to a high-energy aqueous offgas cleanup train. Significant features of the design include secondary containment throughout the process to prevent alpha contamination release, an instrumentation control system substantially upgraded from conventional nonradioactive systems, and a high level of redundancy among critical process operations.

### LOS ALAMOS PROGRAM

The major objectives of the Los Alamos program in support of the CAI/LLW demonstration are to: 1) develop and transfer design and operating data based on experimental tests to the liason subcontractor, and 2) provide technical assistance to both the subcontractor and utility through licensing and completion of the demonstration. Los Alamos, therefore, assumes the lead technical role in establishing design and operating parameters for the CAI/LLW demonstration process.

### DEVELOPMENT STUDIES

Experimental development studies in support of the CAI/LLW demonstration program have focused on two areas: 1) evaluation of fission/activation product distributions within the process, and 2) feasibility of incineration of ion exchange resins.

### Fission and Activation Product Studies

Two tests were conducted to determine the behavior of fission and activation products within the CAI system. Each test used as feed material, a solid combustible composition (Table 1) typical of nuclear power plant dry active waste. The simulated waste was packed in .057 m<sup>3</sup> cardboard boxes at a package density of 79.7 kg/m<sup>3</sup> (4.54 kg per box). Fission and activation products used in both tests were selected as being representative of the physical and chemical properties of a broader selection of isotopes which would be expected to occur in actual reactor LLW. The radionuclides used in these studies included I-131, Cs-137, Ru-106, Fe-59, and Co-60. Feed conditions for both tests are given in Table 2. The order in which the isotopes appear in Table 2 for each test is the order in which they were fed to the incinerator. In Run FAP-1, isotope feed periods were sequential, whereas Run FAP-2 provided a four hour period between each isotope feed period in which non-spiked boxes were fed at the same rate as spiked boxes. Run FAP-1, conducted in July 1981, involved the use of hybrid germanium detectors installed at various locations in the offgas cleanup system (Fig. 2). The purpose of the detectors was to measure the gamma activity and thereby quantity of each isotope in the gas phase at each location, indicated by the numbers in Fig. 2. Some difficulties were

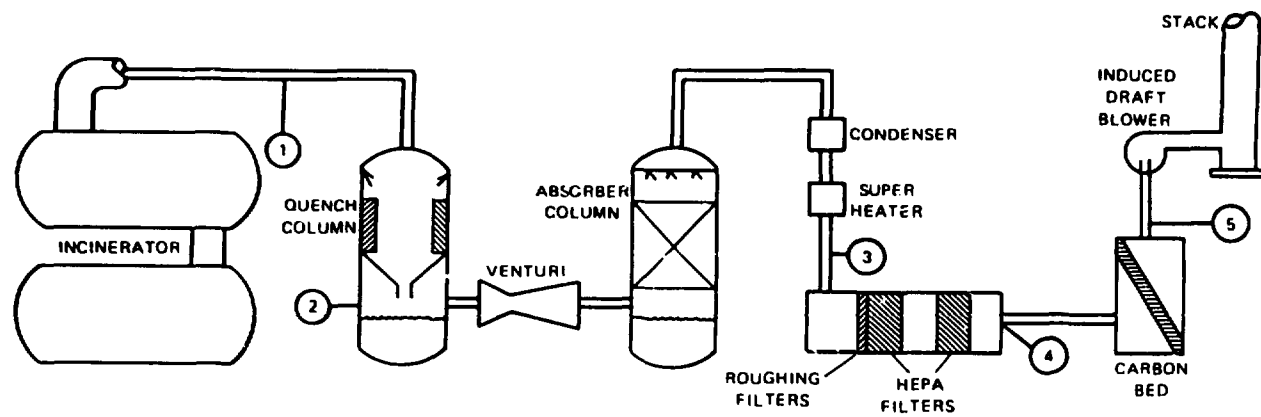


Fig. 2. CAI offgas cleanup system showing locations of hybrid germanium detectors.

encountered in obtaining representative gamma detection at locations 1 and 2. At location 1, a gas side stream was withdrawn from the hot duct and passed through the detector with some uncertainties arising in quantifying the fraction withdrawn. Due to the configuration at location 2, the base of the quench column, the detector counted not only activity in the gas phase but also in the liquid draining down through the column. Despite these problems, some interesting results were obtained from the test burn. For example, count rates for all isotopes at each detector location reached a maximum approximately six hours after completion of feeding each isotope. Data from run FAP-1 is being used to supplement data generated by run FAP-2.

**TABLE 1**  
**COMBUSTIBLE COMPOSITION FOR FISSION/ACTIVATION**  
**PRODUCT INCINERATION TESTS**

| <u>Component</u> | <u>Percent by Weight</u> |
|------------------|--------------------------|
| Cellulosics      | 35                       |
| PVC              | 12                       |
| Polyethylene     | 23                       |
| Rubber           | 30                       |

Run FAP-2, conducted in August 1982, involved the use of modified EPA Method 5 sample trains as a means of sampling the gas phase for radionuclides. Gas sampling was carried out at locations 1,4, and 5 of Figure 2, and in addition, just downstream of the absorber column. Sampling was not done at the base of the quench column. The technique involved the steady withdrawal of a measured volume of gas at each location over the feed duration of each isotope. The sample trains consisted of two liquid-filled impingers, an activated carbon impinger, and a silica gel impinger in series to condense and adsorb all volatile isotopes in the gas sample. Preliminary results of sample counting indicate that for I-131 and Cs-137, the two most volatile isotopes used in the test, absolutely none of either isotope was detected in the carbon bed outlet (location 5, Fig. 2) Trace quantities of I-131 and no Cs-137 were detected in the gas phase downstream of the HEPA filters. Volume and mass reduction factors based on feed and ash were approximately 400:1 and 40:1, respectively for Run FAP-2. A comprehensive report will be issued summarizing the results of both tests with emphasis on radionuclide decontamination factors and material balances throughout the system.

TABLE 2  
RADIONUCLIDE FEED CONDITIONS FOR FISSION/ACTIVATION  
PRODUCT INCINERATION TESTS

| TEST<br>NUMBER     | ISOTOPE | ISOTOPE<br>CONCENTRATION |          | FEED RATES      |                    | FEED<br>DURATION<br>(h) |
|--------------------|---------|--------------------------|----------|-----------------|--------------------|-------------------------|
|                    |         | (mCi/box)                | (mCi/kg) | WASTE<br>(kg/h) | ISOTOPE<br>(mCi/h) |                         |
| FAP-1 <sup>1</sup> | I-131   | .125                     | .028     | 36.4            | 1.0                | 8                       |
| FAP-1 <sup>1</sup> | Cs-137  | .125                     | .028     | 36.4            | 1.0                | 4                       |
| FAP-1 <sup>1</sup> | Ru-106  | .125                     | .028     | 36.4            | 1.0                | 4                       |
| FAP-1 <sup>1</sup> | Fe-59   | .125                     | .028     | 36.4            | 1.0                | 4                       |
| FAP-1 <sup>1</sup> | Co-60   | .125                     | .028     | 36.4            | 1.0                | 4                       |
| FAP-2 <sup>2</sup> | I-131   | .313                     | .069     | 36.4            | 2.5                | 4                       |
| FAP-2 <sup>2</sup> | Cs-137  | .156                     | .034     | 36.4            | 1.25               | 4                       |
| FAP-2 <sup>2</sup> | Ru-106  | .156                     | .034     | 36.4            | 1.25               | 4                       |
| FAP-2 <sup>2</sup> | Fe-59   | .156                     | .034     | 36.4            | 1.25               | 4                       |
| FAP-2 <sup>2</sup> | Co-60   | .156                     | .034     | 36.4            | 1.25               | 4                       |

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(<sup>1</sup>) Continuous monitoring using hyperpure germanium gamma detectors

(<sup>2</sup>) Offgas sampling using modified EPA method 5 sample trains.

### Ion Exchange Resin Incineration Tests

Test burns were conducted to examine the feasibility of combustion of ion exchange resins in two feed modes: 1) batch solid feed, and 2) continuous slurry feed. For the batch feed mode, cation and anion powdered and beaded resins were fed separately into the incinerator in 0.057 m<sup>3</sup> cardboard boxes (9.08 kg/box). Although the resins were clean and nonradioactive, observation of burning characteristics showed the absence of problems reported with other incineration systems such as violent spalling reactions and melting.

As an alternative to batchwise feeding of resins to the incinerator, a test was conducted in April 1982, to establish the feasibility of continuous resin slurry feed. A liquid/slurry feed/burner system consisting of an agitated feed tank, positive displacement progressing cavity pump, and a commercial high intensity vortex liquid burner was installed as a new addition to the CAI process. The burner (Fig. 3) is mounted on the lower chamber of the CAI firing at a 45° angle downward into the chamber. The burner nozzle can accept high concentration slurries via external atomization with air or steam. Combustion air is supplied tangentially into the burner wind box creating high turbulence for thorough mixing of the atomized feed stream and air. Natural gas can be supplied at the burner should supplemental heat be required. A test was conducted using this arrangement to burn clean non-radioactive beaded cation exchange resins.

After a number of laboratory feed preparation studies, a slurry composition was established which would provide both suspension of the resin beads in the liquid during transport to the burner, and sufficient heating value to maintain a stable flame. The slurry was prepared by adding the wet resin to a 50/50 w/w emulsion of No. 2 fuel oil and water. The emulsion was formed by adding a small quantity of commercial surfactant. Resin concentrations tested were as high as 50 weight percent of the slurry on a wet resin basis. Optimum flame stability was achieved at slurry feed rates of 60 to 82 kg/h (15 to 20.5 kg/h dry resin). Based on the overall excellent performance of the CAI during these tests, both batch and continuous slurry feed appear to be viable methods of introducing ion exchange resins into the incinerator. A detailed report summarizing the results of the resin incineration tests will be issued in early FY-1983.

### PROGRAM PLANS FOR FY-1983.

Involvement of the utility GPU-Oyster Creek, NJ will require, as a minimum, site-specific waste and site boundary release limit information, participation in the licensing proceedings with the selected subcontractor Gilbert-Commonwealth, and CAI/LLW operation. At the outset, tasks, objectives, and responsibilities will be redefined to reflect the capabilities and circumstances of all participants.

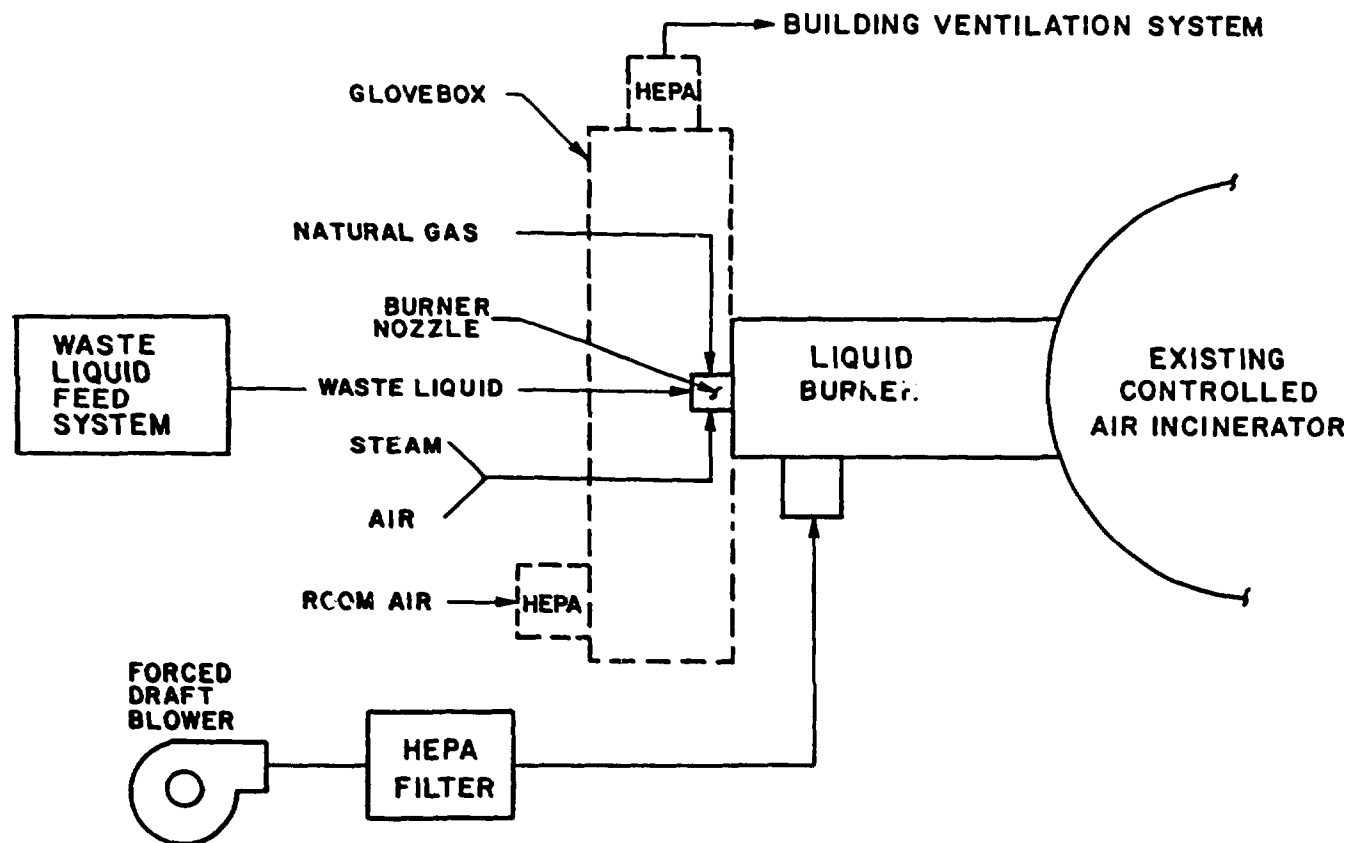


Fig. 3. Liquid/slurry burner system mounted on lower chamber of CAI.

Anticipated Los Alamos activities will include incineration tests using site-specific or non-radioactive loaded resins. These experiments will serve to verify design and operating parameters and provide offgas emissions information for the CAI demonstration unit. Further assistance will be provided to the subcontractor and host utility during start-up, non-radioactive shakedown tests, and demonstration operation of the CAI/LLW system.

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3. L.A. Stretz, L.C. Borduin, W.E. Draper, and R.A. Koenig, Radioactive Waste Incineration Studies at the Los Alamos Scientific Laboratory, Los Alamos Scientific Laboratory report LA-UR-80-1408 (1980).
4. J.S. Vavruska, L.A. Stretz, and L.C. Borduin, Hazardous and Radioactive Waste Incineration Studies at Los Alamos, Los Alamos National Laboratory report LA-UR-81-1170 (1981).

PROPOSED WASTE ACCEPTANCE CRITERIA  
FOR DEPARTMENT OF ENERGY  
SHALLOW LAND BURIAL SITES



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ABSTRACT

Waste acceptance criteria were developed as part of an overall effort to establish general criteria for the shallow land burial of Department of Energy low-level radioactive waste. Other areas of criteria development include site selection, design, operations, closure/post-closure, corrective measures, and quality assurance. All the criteria are based on a concept of low-level waste management which recognizes the use of system performance objectives.

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INTRODUCTION

Fifteen criteria relating to radionuclide content, waste form, structural stability, packaging and labeling and dual hazards are defined. Factors are identified when additional information is needed to determine compliance with a criterion. Definitions pertinent to the interpretation of the waste acceptance criteria are also included.

DEFINITIONS

**Barrier:** Any medium that retards the movement of emplaced radioactive material or reduces the probability of human access to the material. (Examples are engineered features such as the waste container, waste form, or backfill material; a natural geologic medium; or institutional site access and use restrictions.)

**Complexing Agent:** A non-metallic ion or molecule that forms coordinate covalent bonds with metal ions.

**Corrosive:** Material that exhibits the characteristic of corrosivity as defined in 40 CFR 261.22.

**Criterion:** A standard rule or test on which a judgment or decision may be based.



**Disposal:** Isolation that relies on a combination of natural (environmental) and man-made barriers, does not permit easy human access to waste after emplacement, and does not require perpetual human maintenance and monitoring.

**Free Liquid:** Liquid not chemically or physically bound to the waste material and which can be removed by gravity drainage.

**Ignitable:** Material that exhibits the characteristic of ignitability as defined in 40 CFR part 261.21.

**Low-Level Wastes:** Any radioactive waste containing less than 100 nCi/g of transuranium radionuclides and not classified as high-level waste or uranium mill tailings.

**Performance Objective:** Ground rules for the shallow land burial of low-level radioactive waste.

**Performance Period:** The period of time the disposal site must meet performance objectives.

**Reactive:** Material that exhibits the characteristic of reactivity as defined in 40 CFR 261.23.

**Shallow Land Burial:** The disposal of radioactive waste in near surface excavations where the disposed waste is covered with a soil overburden.

**Stabilized Waste:** Waste that has undergone treatment and/or packaging to improve its radionuclide retention characteristics.

**Waste Containers:** The disposal containment vessel for waste materials, including any liner or shielding materials that are intended for disposal.

**Waste Form:** The waste package less the container, if any, and consisting of the low-level waste either treated or untreated, including any inert fillers, as presented for disposal.

**Waste Package:** The assemblage of low-level waste that is disposed. This normally includes the container plus the contained material.

## PERFORMANCE OBJECTIVES

The performance objectives for the land disposal of low-level radioactive waste are based on limits of contaminant release to the accessible environment. Performance objectives are the ground rules for land disposal. Criteria are set forth to ensure the performance objectives will be met. A site specific performance assessment is necessary to document compliance.

- a.) Every shallow land disposal site shall be operated in conformance with Department of Energy occupational exposure standards.
- b.) The annual dose from radionuclides released from any shallow land burial site to persons outside the site boundary from all pathways shall not exceed the following:

- 25 MREM/YR - - Whole body
- 75 MREM/YR - - Thyroid
- 25 MREM/YR - - Any organ

- c.) Waste shall not be disposed of in a shallow land burial site that results in a waste inventory that poses a potential threat to health and safety after a performance period of 500 years has ended.

The performance objectives are the key to determining what wastes are acceptable for land disposal. The performance period in conjunction with nuclide half-life can be used to determine a concentration limit for each radionuclide accepted for burial at low-level waste disposal sites.

#### WASTE ACCEPTANCE CRITERIA

Waste acceptance criteria are standard rules by which the acceptability of a low-level waste package at a shallow land burial facility can be judged. Both site specific and generic criteria are given. Generic criteria are applicable at all DOE sites as written. The site specific criteria recognize the natural and engineered variability of Department of Energy disposal sites. They require a determination to be made, based on the site performance characteristics, of what wastes are acceptable such that the criterion is met.

1. Disposal sites shall develop site specific acceptance criteria for on-site and off-site generated waste as required for that site to meet performance objectives and general waste acceptance criteria. Special shipments not covered by criteria shall only be accepted after approval by proper administrative authority.

2. Waste acceptance criteria shall include maximum allowable concentrations for each radionuclide disposed at the site. Maximum allowable concentrations shall be based on an assessment of retention characteristics of the waste package and of the site.

3. The waste form geometry shall be such that criticality safety is maintained. Factors to be considered under this criteria include:

- a.) Potential for thermal or radiative energy output from the waste form.
- b.) Waste container properties.
- c.) Effect of thermal or radiative energy output on the performance of the disposal site.

5. Low-level waste containing pathogenic, infectious, or toxic material shall be treated and/or packaged to reduce to the maximum extent practical the potential hazard. Factors to be considered under this criteria include:

- a.) Type, concentration, and quantity of infectious pathogenic, or toxic material.

6. Low-level waste which is hazardous due to ignitable, reactive, or corrosive characteristics shall be rendered non-hazardous.

7. Low-level waste in a gaseous form shall be packaged at a pressure that will not jeopardize container integrity or performance objectives. Factors to be considered under this criteria include:

- a.) Physical properties of the container.
- b.) Physical and chemical properties of the gas.

8. Low-level waste consisting of or containing powders, ash, or similar respirable particulates shall be treated and stabilized or otherwise contained to reduce radiation or chemical exposure potential to personnel during normal staging, transport, handling, and disposal.

9. Low-level waste packages shall contain no more than 1.0 volume percent free liquid and such liquid shall be non-corrosive to the waste container.

10. Waste packages (container and contents) containing quantities of chelating or complexing agents, or organic material whose decomposition products could jeopardize site performance objectives by increasing radionuclide migration shall not be accepted.

11. Structural stability of the waste package (container and contents) shall be such that performance objectives of the site are not jeopardized and can withstand normal loads imposed in handling, shipping, and stacking of waste packages. Waste containers must be either nonflammable or self-extinguishing. Factors to be considered under this criteria include:

- a.) Compressibility of the waste package.
- b.) Shipping and handling methods to be used.
- c.) Time frame over which the waste package will maintain its physical dimensions.

12. Off-site shipments of low-level waste shall be in compliance with DOE Transportation requirements - DOE Order 5480.1.

13. Labeling and shipping paper requirements shall be established by each disposal site for shipments of on-site generated waste. The labels and shipping papers shall facilitate recordkeeping, handling, disposal, and reporting information for SWIMS. Factors to be considered under this criteria include:

- a.) Labeling that requires the following minimum information be affixed to the waste package:

- 1.) "Radioactive waste material" and radiation symbol.
  - 2.) Shippers identification number.
  - 3.) Maximum radiation level.
  - 4.) Dual hazards identified.
- b.) Shipping paper tracking or accounting system that assures prompt identification of missing or misdirected packages.
- c.) Shipping papers that contain the following minimum information:
- 1.) Date of shipment.
  - 2.) Date of receipt.
  - 3.) Gross volume and weight of the package.
  - 4.) Total curie content.
  - 5.) Principal radionuclides.
  - 6.) Any associated hazards.
  - 7.) Level of surface contamination.
  - 8.) Direct radiation level of the package.
  - 9.) Information as required for the National Report System (SWIMS).

14. Package exteriors must be reasonably free of dirt, moisture, and rust. Surface radioactivity contamination shall be less than 2200 DPM/100 CM<sup>2</sup> beta-gamma and less than 220 DPM/100 CM<sup>2</sup> alpha. Packages with surface exposure rates greater than 200 MREM/HR shall be color-coded or otherwise clearly identified to ensure appropriate handling.

15. Standardized, reproducible tests which simulate expected disposal site conditions shall be applied as needed to assess waste form and/or package performance. When extrapolated, test results shall be applied conservatively.

## CONCLUSIONS

The fifteen waste acceptance criteria presented are intended to provide the framework for determining what low-level waste packages are acceptable for shallow land burial. The criteria are based on a systems concept of waste management which recognizes the use of performance objectives. The criteria are the tools for ensuring the site will meet performance objectives for the long term confinement of low-level radioactive waste.

LOW LEVEL WASTE FORM CRITERIA\* 

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## INTRODUCTION

The purpose of the Low Level Waste Form Criteria program at Brookhaven National Laboratory is to provide technical assistance to the Low Level Waste Licensing branch (LLWLB) of the Nuclear Material Safety and Safeguards Division of the Nuclear Regulatory Commission in those areas of waste management related to the disposal of low level wastes. This technical assistance is in the form of both experimental studies and critical reviews in the areas of waste forms and containers, as well as the interaction of wastes with disposal sites. This program is complemented by another technical assistance program at BNL related to non-fuel cycle wastes as well as a third program concerned with waste management problems to related the clean up activities at Three Mile Island Unit 2.

During the past year, most of the activities in this program have been in support of the new regulation governing the disposal of low level radioactive wastes, 10 CFR 61 (proposed), as well as the LLWLB draft Technical Position on Waste Form (BTP). The activities have been reflected in tasks conducted in five areas:

1. Technical considerations for high integrity containers (HIC's) for the disposal of spent ion exchange resin waste.<sup>(1)</sup>
2. Permissible radionuclide loadings for organic ion exchange resin wastes from nuclear power plants.<sup>(2)</sup>
3. Technical factors affecting low level waste form acceptance requirements of the proposed 10 CFR 61 and draft BTP.<sup>(3)</sup>
4. Development of a computer code to calculate the dose to the public from a well or river via the modeling of groundwater transport of radionuclides in proximity to a low level waste disposal site.<sup>(4)</sup>

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\*Work performed under the auspices of the United States Nuclear Regulatory Commission.

5. Analyses of soils obtained at the low level waste disposal sites at Hanford, Washington; Barnwell, South Carolina; and Sheffield, Illinois.(5,6)

These efforts will be briefly described below.

#### TECHNICAL CONSIDERATIONS FOR A HIGH INTEGRITY CONTAINER FOR RESIN WASTES

In this project an attempt was made to characterize the drainable liquids which might be present in a high integrity container of spent ion exchange resins. A survey of nuclear power plant experience with respect to resin usage was conducted. This information supplemented the results of earlier surveys by Lin<sup>(7)</sup> and Clark<sup>(8)</sup>. As a result of these surveys, information supplied by manufacturers and suppliers of organic ion exchange resins, and a review of the pertinent literature, it was concluded that it is not possible to accurately define the characteristics of drainable liquids which may be present in a HIC. However, several observations were made.

- The pH of the liquid can vary from 2 to 11. Solutions with pH <4 are generally corrosive to carbon steel.
- Chloride ions (Cl<sup>-</sup>) may be present in the drainable liquids in a disposal liner at concentrations of hundreds of parts per billion. Chloride is corrosive to stainless steel and, therefore, the effect of Cl<sup>-</sup> on the container lifetime should be considered if stainless steel is to be used as a liner material.
- The resin waste within a particular disposal liner may come from a single demineralizer in a plant or from a spent resin holding tank where resins from several demineralizers are mixed.
- The radionuclide concentration in the free liquid remaining after dewatering will be lower than that on resins.
- The gross  $\beta, \gamma$  activity on ion exchange resins being disposed of varied from <1 to 5000  $\mu$ Ci/mL. This, coupled with the large variations in the radioisotopes present, will affect the total radiation dose delivered to the resins after burial, thus, affecting resin degradation.
- Gas buildup inside a disposal liner may result from both radiolysis and biodegradation. The radiolysis of water or the ion exchange resins will produce primarily hydrogen gas. Biodegradation of organic carbon will yield carbon dioxide. More work is needed in the area of biodegradation of this waste stream.

A second area of study in this program was an assessment of the extent of dewatering which should be required for resin wastes prior to disposal of resins in a HIC. From a consideration of industry experience and the availability of water in organic ion exchange resins, it was concluded that a limit of 0.5% free water in a HIC was reasonable and achievable. This limit would, however, require that the utilities demonstrate the accuracy of the dewatering methods used to achieve it.

#### PERMISSABLE RADIONUCLIDE LOADINGS FOR ORGANIC ION EXCHANGE RESIN WASTES

In this task, a survey of resin usage at operating nuclear power plants was conducted to determine the amount and identity of radionuclides typically found in spent ion exchange resin. Further, an attempt was made to estimate the dose received by resins during their useful lifetime. From this information as well as a review of the literature relevant to radiation damage to organic ion exchange resins, it was concluded that the maximum permissible dose to resin should be restricted to  $10^8$  rad. This recommendation is in contrast to the curie loading limit given in the BTP. At this level, a measurable amount of damage to the resin will probably occur. However, this damage would be expected to be insignificant at a power plant or disposal site.

In addition to providing the rationale for a loading limit for ion exchange wastes, a test procedure for demonstrating that a particular ion exchange medium could be safely loaded above that limit was also provided. The test is basically a short-term irradiation of the medium in which the gas generation of the irradiated material is measured. The reference material for the test irradiation is a mixed bed organic resin (1:1 cation:anion in the  $H^+/OH^-$  form). The amount of gas evolved from this mix was 0.3 mL of gas at STP/g of dried resin at a total dose of  $10^8$  rad. The gas generation from the reference material at  $10^8$  rad is the maximum amount of gas acceptable. Thus, the gas generated for a particular medium must be less than 0.3 cc of gas at STP/g of dry resin when it is irradiated to the anticipated dose.

#### TECHNICAL FACTORS AFFECTING WASTE FORM REQUIREMENTS OF THE PROPOSED 10 CFR 61 AND DRAFT BTP

This task provides support to the NRC in specific areas relating to the proposed regulation, 10 CFR 61, and the draft BTP. Six specific areas of questions have been identified by NRC. They are:

1. What is the technical basis for limiting the pressure (1 atm) and activity (100 curies) of radioactive gases? Should limit(s) different from those set out in proposed 10 CFR 61.56 be established?

2. What would be required to demonstrate that a proposed stable waste will be recognizable for 300 or 500 years?
3. What is the feasibility of achieving less than 5 percent deformation in wastes considering current methods and practices of solidification using cement, asphalt and vinyl ester-styrene systems which may not result in completely filling the container with solidified waste. Can modified (but practicable) procedures help to achieve this or should the 5% limit be changed to reflect current practice?
4. Are the testing techniques to demonstrate resistance to biodegradation specified in ASTM G21 and ASTM G22 adequate to test wastes typically solidified using cement, asphalt or vinyl ester-styrene processes? If these tests are inadequate, identify alternative testing required to demonstrate stability from biodegradation.
5. Is the testing specified in accordance with ASTM B553 (to assure that wastes can be shown to be resistant to thermal degradation) adequate for solidified wastes? If this testing is not adequate, identify what alternative testing is required to show resistance to thermal degradation.
6. What basis should be used to determine if a waste should be considered as explosive or pyrophoric?

These areas have been given consideration by the staff and a draft report on this task is currently under review by the NRC.

#### MODELING OF GROUNDWATER TRANSPORT

The original purpose of this task was to evaluate the effect of leach rate on the migration of radionuclides in the groundwater. In particular, it was intended to determine if a leach rate could be set for given radionuclides such that the environmental performance criteria (10 CFR61, Section 41) were met at three hypothetical disposal sites. Two, one-dimensional transport models were to be used for this task. The first was a model developed by Dames and Moore for the Atomic Industrial Form<sup>(9)</sup> and the second was a modification of this model made by the Technology for Energy Corporation.<sup>(10)</sup> Once these leach rates were set, current technology in waste solidification would be reviewed to determine if these leach rate requirements could be met by current practice.

In the course of this assessment, a composite code, GWMBNL1, was developed to assist in the evaluation of the sensitivity dose rate to the public to leach rate. In the application of this model to the hypothetical disposal sites, it was felt that the objectives of the task



could not be adequately met and the remaining effort in this area has been discontinued.

#### ANALYSES OF SOILS FROM LOW LEVEL WASTE DISPOSAL SITES

The properties of the soils at low level waste disposal sites are one of many factors needed to assess the long-term performance of HIC's and solidified waste after disposal. Thus, in order to provide the NRC with a data base useful in assessing such performance, the staff has conducted soil analyses at three low level burial sites located at Barnwell, South Carolina; Hanford, Washington; and Sheffield, Illinois. Measurements were performed in four areas: soil resistivity, soluble ion content, moisture content and soil acidity. The results are summarized in Table 1. It is evident from the data obtained that none of the soil at the three sites is expected to be severely corrosive towards steel. Further, none contains high levels of species such as chloride and sulfate which are known to exacerbate corrosion. Finally, only trace levels of sulfides were detected in the undisturbed soils indicating that prior to burial of the waste the soils themselves are not reducing. Of the three sites, the Sheffield site, based on soil properties, is likely to be the most corrosive site.

#### FUTURE WORK

It is anticipated that future work in this program will continue to consist of a variety of experimental and critical review tasks. Current effort is devoted to a task in which the thermal cycling test and biodegradation tests recommended in the draft BTP will be conducted on typical power plant waste streams (boric acid, and sodium sulfate evaporator concentrates, mixed bed ion exchange resins, and filter cake sludge with powdered resin) solidified using cement, bitumen and vinyl ester-styrene copolymer binders. The purpose of this task is to demonstrate that the test methods and acceptability criteria set forth in the areas of biodegradation and thermal cycling are reasonable and can be achieved.

Table 1

## Summary of Results of Soil Analyses at Barnwell, Hanford, and Sheffield

|                                               | Barnwell                                     | Hanford                                      | Sheffield                                    |
|-----------------------------------------------|----------------------------------------------|----------------------------------------------|----------------------------------------------|
| Soil Resistivity (ohm-cm)                     |                                              |                                              |                                              |
| Field                                         | 160,000 to 100,000                           | 23,000 to 570                                | 6,300 to 3,300                               |
| Laboratory                                    | 360,000 to 83,000                            | 300,000 to 16,000                            | 62,000 to 3,200                              |
| Saturated paste                               | 130,000 to 89,000                            | 15,000 to 6,000                              | 13,000 to 2,500                              |
| Soluble Ions (meq/100 g of soil) <sup>a</sup> |                                              |                                              |                                              |
| Ca <sup>+2</sup>                              | 1.0x10 <sup>-2</sup> to ND                   | 8.2x10 <sup>-2</sup> to 3.9x10 <sup>-3</sup> | 2.6x10 <sup>-1</sup> to 2.8x10 <sup>-2</sup> |
| Mg <sup>+2</sup>                              | 7.4x10 <sup>-3</sup> to ND                   | 3.3x10 <sup>-2</sup> to 4.5x10 <sup>-3</sup> | 1.4x10 <sup>-1</sup> to 2.0x10 <sup>-2</sup> |
| K <sup>+</sup>                                | 4.3x10 <sup>-3</sup> to 6.7x10 <sup>-5</sup> | 2.5x10 <sup>-2</sup> to 2.2x10 <sup>-3</sup> | 6.6x10 <sup>-3</sup> to 6.7x10 <sup>-4</sup> |
| Na <sup>+</sup>                               | 2.7x10 <sup>-2</sup> to 3.8x10 <sup>-3</sup> | 1.8x10 <sup>-1</sup> to 5.3x10 <sup>-3</sup> | 2.3x10 <sup>-2</sup> to 4.7x10 <sup>-3</sup> |
| HCO <sub>3</sub> <sup>-</sup>                 | 1.9x10 <sup>-2</sup> to ND                   | 8.6x10 <sup>-2</sup> to 1.3x10 <sup>-2</sup> | 9.0x10 <sup>-2</sup> to 4.9x10 <sup>-3</sup> |
| SO <sub>4</sub> <sup>-2</sup>                 | 1.2x10 <sup>-2</sup> to ND                   | 4.2x10 <sup>-2</sup> to 4.0x10 <sup>-3</sup> | 2.8x10 <sup>-1</sup> to 8.3x10 <sup>-3</sup> |
| S <sup>-2</sup>                               | 1.0x10 <sup>-3</sup> to ND                   | 2.5x10 <sup>-4</sup> to ND                   | 1.4x10 <sup>-3</sup> to ND                   |
| Cl <sup>-</sup>                               | 1.9x10 <sup>-2</sup> to 8.3x10 <sup>-4</sup> | 1.6x10 <sup>-1</sup> to 9.6x10 <sup>-4</sup> | 1.4x10 <sup>-2</sup> to 1.2x10 <sup>-4</sup> |
| Moisture Content (%)                          | 20.6 to 6.3                                  | 6.5 to 0.67                                  | 25.2 to 4.8                                  |
| Soil acidity                                  |                                              |                                              |                                              |
| Soil pH                                       | 5.4 to 4.8                                   | 6.5 to 4.0                                   | 7.8 to 6.2                                   |
| CaCl <sub>2</sub> extract pH <sup>b</sup>     | 4.1 to 3.7                                   | 7.7 to 7.1                                   | 7.6 to 6.9                                   |
| Saturated paste extract pH                    | 8.2 to 3.2                                   | 7.6 to 6.6                                   | 8.2 to 5.8                                   |
| Total Acidity (meg/eq/100 g dry soil)         | 8 to ND                                      | 4 to ND                                      | 7 to ND                                      |

<sup>a</sup>ND means not detectable.<sup>b</sup>1:2 mix soil: 0.01M CaCl<sub>2</sub>.

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## ACKNOWLEDGEMENTS

We would like to acknowledge the contributions of the following investigators to this program: S.F. Chan, J. Coleman, D. Dougherty, C. Hsu, M. Lin, D.R. MacKenzie, P.L. Piciulo, C. Sastre, C.E. Shea, and F. Vaslow.

## SPECIAL WASTE FORM LYSIMETERS

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## ABSTRACT

Lysimeters to measure radionuclide migration from well-characterized special waste forms obtained from power reactor sites have been installed at the Savannah River Plant low-level waste burial ground. Ten lysimeters containing five types of waste forms have been operating since March 1982. Percolate water and soil moisture samples show no evidence of radionuclide migration after six months of operation. Projected improvements in sample collection and analysis methods will give greater sensitivity for low-level radionuclide detection. The lysimeter experiment is expected to operate for ten years.

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INTRODUCTION

Lysimeter studies at the Savannah River Plant (SRP) are designed to measure radionuclide migration from well-characterized sources under field conditions. The Savannah River Laboratory (SRL) has conducted a variety of lysimeter tests since 1973, in ongoing studies with typical defense low-level waste forms. In 1981, the SRL lysimeter program was expanded to include special waste forms typical of those produced at commercial power reactors. The SRL special waste form lysimeter studies are funded by the DOE Low-Level Waste Management Program (LLWMP).

The purpose of this LLWMP task is to measure radionuclide leaching and migration from encapsulated low-level solid wastes produced at power reactors and emplaced in a humid shallow land burial site. The use of lysimeters will permit direct measurement of migration of radioactivity from these waste forms and will provide a technical basis for evaluating how the forms will behave in an actual burial trench. In addition, leaching and migration results may be compared with those from unencapsulated defense wastes, to measure the benefits of encapsulation.

## STATUS AND ACCOMPLISHMENTS

The SRL Special Waste Form Lysimeter Program was initiated in April 1981, and construction of 14 lysimeters was completed in October 1981. In late February and early March of 1982, the experiment was started when waste forms were buried in 10 of the lysimeters. One other lysimeter was filled with soil but no waste, to serve as a control.

The waste forms were acquired from power reactor sites by Brookhaven National Laboratory (BNL) under subcontract with SRL. BNL also will provide source terms and results of full-scale leach tests for each type of waste form. These characteristics will be given in a report to be issued by BNL in September 1982.

Percolate water first appeared in the lysimeter sumps in May 1982, and the lysimeters have operated properly since then. Monthly analyses of percolate water have shown no radionuclides so far, as expected. Typically, lysimeter tests are long-term experiments, and the special waste form lysimeters may be observed over a 10-year period.

## DESCRIPTION OF LYSIMETERS

Construction Features. The special waste form lysimeters are 10 feet deep and 6 feet in diameter, constructed of fiberglass, which is expected to give better service than metal. The 14 lysimeters include 12 intended for waste forms and 2 controls. Soil used is typical of the SRP low-level waste burial ground, where the lysimeters are located. Other design improvements are soil moisture samplers and soil sample ports.

Each lysimeter contains two soil moisture samplers for obtaining water samples from the unsaturated zone. The samplers are implanted in the soil two inches below the buried waste. The commercially available samplers are porous ceramic cups connected to vacuum lines that extend to the surface. A soil moisture sample is collected by maintaining low pressure on the cup for a few hours.

The waste lysimeters are arranged in two circular sets of six, with an access caisson in the center of each set. Soil sampling ports are located in the walls of the access caisson and each lysimeter. The ports permit soil samples to be removed horizontally from each lysimeter at depths of 12 and 22 inches below the waste. Samples would be obtained with a soil auger or coring tool. Because removal of soil samples would perturb the experiment, such samples will be taken rarely, and only when needed to understand the experiment.

Waste Forms. Five types of low-level waste forms, described in Table 1, were procured by BNL from power reactor sites and shipped to SRL. The waste forms are the size of 55-gallon drums and weigh 478 to 715 pounds each. The forms were removed from their metal drums

TABLE 1. Special Waste Forms for SRP Lysimeter Tests

| <u>Reactor</u> | <u>Waste<sup>a</sup></u> | <u>Binder</u>       | <u>Radioactivity,<br/>mCi</u> | <u>Number of<br/>Lysimeters</u> |
|----------------|--------------------------|---------------------|-------------------------------|---------------------------------|
| Dresden        | CW                       | Cement <sup>b</sup> | 272-340                       | 2                               |
| Dresden        | CW + R                   | Cement <sup>b</sup> | 748-816                       | 2                               |
| Dresden        | CW                       | Polymer             | 184-587                       | 3                               |
| Dresden        | CW + R                   | Polymer             | 964                           | 1                               |
| Beaver Valley  | BAC                      | Cement <sup>c</sup> | 2.3-2.5                       | 2                               |

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- a. CW = concentrate waste, 25 wt% Na<sub>2</sub>SO<sub>4</sub> evaporator bottoms  
 R = ion exchange resin  
 BAC = boric acid concentrate, 12 wt% solution  
 b. Portland cement  
 c. Masonry cement

and buried bare. A total of 10 waste forms were emplaced in 10 waste lysimeters, and one control lysimeter was established, leaving 3 lysimeters as spares for possible future use.

BNL is determining the source term for each waste form by analyzing samples of the waste streams from which the forms were made and then calculating the radionuclide inventory in each waste form. Radionuclides known to be present in all of the waste forms are Mn-54, Co-60, Zn-65, Cs-134, Cs-137, Ce-144, and probably Sr-90. In addition, the Beaver Valley waste contains small amounts of Co-57, Co-58, and Sb-125. Preliminary analyses by BNL indicate that Co-60 is the predominant radionuclide at this time, followed by Cs-137.

Lysimeter Operation. The special waste form lysimeters have operated for six months in the unsaturated mode. The first two months were required to establish proper operation, before percolate rainwater began to accumulate in the lysimeter sumps. Once established, the lysimeters have continued to function correctly. Percolate water may be collected and sampled, or discharged to a drain pit.

Sampling Program. Percolate water from the special waste form lysimeters is sampled monthly and analyzed for gross alpha and gross beta/gamma radioactivity. So far, the waste and control lysimeters have shown no radioactivity above background: <3 pCi/L gross alpha and <60 pCi/L gross beta/gamma.

Soil moisture samples are taken quarterly and analyzed for gross alpha, gross beta/gamma, calcium, sodium, and sulfate. June 1982 sampling results showed no significant differences between control and waste lysimeters.

Soil core samples will be taken only when soil moisture or percolate water shows significant radionuclide concentrations. The lysimeters will be cored after three years of operation, even if there has been no evidence of radionuclide migration. Cored lysimeters probably can continue operation with minimal perturbation if clean soil is packed into the void formed by removal of the core sample.

Recently, a more sensitive method of analysis for gamma emitters was developed. Larger volumes of percolate water, up to 20 liters, are concentrated on mixed-bed ion-exchange resin, and then analyzed by high-resolution gamma-ray spectrometry. The special waste form lysimeters are now being equipped with ion-exchange columns so that preconcentration of activity in percolate water can be accomplished in the field. This change will increase sensitivity for radionuclide detection by a factor of about 100 from the gross counting methods. Also being developed is an improved procedure for analysis of Sr-90, which is not determined directly by gamma counting.

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The information contained in this article was developed during the course of work under contract No. DE-AC09-76SR00001 with the U. S. Department of Energy.

WASTE FORM DEVELOPMENT 

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## ABSTRACT

In this program, contemporary solidification agents are being investigated relative to their applications to major fuel cycle and non-fuel cycle low-level waste (LLW) streams. Work is being conducted to determine the range of conditions under which these solidification agents can be applied to specific LLW streams. These studies are directed primarily towards defining operating parameters for both improved solidification of "problem" wastes and solidification of "new" LLW streams generated from advanced volume reduction technologies. Work is being conducted to measure relevant waste form properties. These data will be compiled and evaluated to demonstrate compliance with waste form performance and shallow land burial acceptance criteria and transportation requirements (both as they exist and as they are modified with time).

The work conducted under this program in FY 1982 and planned FY 1983 efforts are discussed.

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PROGRAM SCOPE AND OBJECTIVES

The objective of the Department of Energy's National Low-Level Waste Management Program (NLLWMP) is to "provide an acceptable Low-Level Waste Management System by 1988" which will enable disposal "of materials that have been declared as low-level waste (LLW) in a manner which will protect public health and safety in the short and long terms." The Waste Form Development program is an integral part of the NLLWMP's plan to develop technology applicable to the aggregate of LLW streams from generation to disposal as expressed in Milestone B, "Develop Technology for Waste Treatment, Handling and Packaging for Shallow Land Burial Site Disposal." Data from this program also provide input into other NLLWMP Milestones.



Low-level wastes at nuclear facilities have traditionally been solidified using portland cement. Urea-formaldehyde has also been used for LLW solidification while bitumen (asphalt) and thermosetting polymers will be applied to domestic wastes in the near future. However, operational difficulties have been observed with each of these solidification agents. Such difficulties include incompatibility with waste constituents inhibiting solidification, free standing water, premature setting and fires. Some specific wastes, so-called "problem" wastes, have proven difficult to solidify with one or more of the contemporary agents or are solidified at low efficiencies. Existing technologies may not be directly applicable to the solidification of "new" wastes which are beginning to be generated using advanced volume reduction technologies in an effort to reduce waste disposal costs. In addition, consideration must be given to the application of any additional agents which may be introduced in the near future for the solidification of low-level wastes. The scope of this work includes both major fuel cycle and non-fuel cycle LLW streams.

This program will identify and evaluate potential agents and processes for the improved solidification of low-level wastes and the solidification of "new" LLW streams generated from advanced volume reduction technologies. The data developed will provide input into DOE efforts to develop a handbook for LLW treatment and disposal by shallow land burial. It will also provide the basis to demonstrate compliance with waste form performance and shallow land burial acceptance criteria and transportation requirements (both current and as they change with time).

#### SELECTION OF SOLIDIFICATION AGENTS FOR FURTHER STUDY

A survey was conducted in FY 1981 to identify and review those agents and processes that could be used for the solidification of LLW. A report entitled A Survey of Agents and Techniques Applicable to the Solidification of Low-Level Radioactive Wastes was issued at the end of FY 1981. Potential LLW solidification agents reviewed fell into several generic classes including hydraulic cements, thermoplastics, thermosetting polymers, glasses, ceramics, mineralization processes and composite processes. The information compiled in this effort was used to select agents and processes for further study. Such agents should potentially be capable of solidifying (either with or without modification) a wide variety of LLW streams, with particular emphasis on "problem" wastes and "new" wastes as previously described. Also, it is desirable if a particular agent chosen is compatible with existing installed waste solidification equipment so as to minimize the need for major capital expenditures. For these reasons, work in FY 1982 concentrated on the use of various hydraulic cements and thermosetting polymers for improved solidification of LLW.

## WASTE COMPOSITION CHARACTERIZATION

In support of waste form development studies, efforts have been made to identify specific LLW streams and determine their compositions. Table 1 lists generic types and sources of LLW. The specific wastes produced within a given class vary considerably in regards to composition, activity content and concentration. While wet wastes obviously require solidification, some dry wastes, particularly incinerator ash and dry salts, must also be solidified because they are easily dispersible. In addition, some wet wastes, ion exchange resins in particular, have been disposed in a dewatered form. However, regulations and burial site operating licenses are moving to require solidification of these wastes.

The emphasis of FY 1982 waste form development studies was the solidification of "problem" wastes and "new" wastes. The "problem" wastes studied included ion exchange resins, oils and organic liquids and specific aqueous concentrates. Studies of "new" wastes included incinerator ash and dry nitrate salt waste.

## FORMULATION DEVELOPMENT STUDIES

Work has been conducted to determine appropriate operating parameters for the solidification of specific LLW streams with various solidification agents. This includes verification of the compatibility of solidification agents with various waste types, identification of waste stream constituents which impede or impair solidification and determination of appropriate compositional limits. FY 1982 formulation development work included the waste-solidification agent combinations listed in Table 2. Work was conducted to investigate the solidification of ion exchange resins, incinerator ash, oil wastes, organic liquids and nitrate salt and concentrate wastes. Solidification agents applied included hydraulic cements (portland type I and III cements and high alumina cement), hydraulic cements with additives, polymer modified gypsum cement (Envirostone) and thermosetting polymers (primarily vinyl ester-styrene).

Formulation acceptability criteria employed for this work require waste forms which are monolithic free-standing solids, homogeneous (on a macroscopic scale) and contain no drainable free standing liquid. These criteria have been applied conservatively in order to provide reasonable assurance that satisfactory solidification will occur for full-scale waste forms using formulations that were initially developed in the laboratory. In addition, waste forms are also required to pass a two week water immersion test without loss of mechanical integrity.

Table 1. Sources and Types of Low-Level Radioactive Wastes

| SOURCES                       | TYPES OF WASTE GENERATED |                     |                  |                     |                           |                 |                            |                              |                                              |      |                             |                        |            |
|-------------------------------|--------------------------|---------------------|------------------|---------------------|---------------------------|-----------------|----------------------------|------------------------------|----------------------------------------------|------|-----------------------------|------------------------|------------|
|                               | DRY WASTES               |                     |                  |                     | WET WASTES                |                 |                            |                              |                                              |      |                             |                        |            |
|                               | COMBUSTIBLE              |                     | NONCOMBUSTIBLE   |                     | FILTER<br>CAR-<br>TRIDGES | SPENT<br>RESINS | SLURRIES<br>AND<br>SLUDGES | AQUEOUS<br>CONCEN-<br>TRATES | SPECIAL<br>AQUEOUS<br>SOLUTIONS <sup>a</sup> | OILS | OTHER<br>ORGANIC<br>LIQUIDS | MEMBRANES <sup>b</sup> | BIOLOGICAL |
|                               | COMPAC-<br>TIBLE         | NONCOM-<br>PACTIBLE | COMPAC-<br>TIBLE | NONCOM-<br>PACTIBLE |                           |                 |                            |                              |                                              |      |                             |                        |            |
| GOVERNMENT DEFENSE            | •                        | •                   | •                | •                   | •                         | •               | •                          | •                            | •                                            | •    | •                           | •                      |            |
| D & D <sup>c</sup>            | •                        | •                   | •                | •                   |                           | •               |                            |                              | •                                            |      |                             |                        |            |
| RD & D <sup>d</sup>           | •                        | •                   | •                | •                   |                           | •               | •                          | •                            | •                                            | •    | •                           | •                      | •          |
| COMMERCIAL FUEL CYCLE MINING  | •                        | •                   | •                | •                   |                           |                 | •                          | •                            |                                              |      |                             |                        |            |
| MILLING                       | •                        | •                   | •                | •                   |                           |                 | •                          | •                            |                                              |      |                             |                        |            |
| UF <sub>6</sub> PRODUCTION    | •                        | •                   | •                | •                   |                           |                 | •                          |                              |                                              |      |                             |                        |            |
| ENRICHMENT                    | •                        | •                   | •                | •                   |                           |                 | •                          |                              |                                              | •    | •                           |                        |            |
| FUEL FABRICATION              | •                        | •                   | •                | •                   |                           |                 | •                          |                              |                                              | •    |                             |                        |            |
| POWER PLANTS                  | •                        | •                   | •                | •                   | •                         | •               | •                          | •                            | •                                            | •    |                             | •                      |            |
| SPENT FUEL STORAGE            | •                        | •                   | •                | •                   |                           | •               | •                          |                              |                                              |      |                             |                        |            |
| D & D                         | •                        | •                   | •                | •                   |                           | •               |                            |                              | •                                            |      |                             |                        |            |
| NONFUEL CYCLE MEDICAL         | •                        |                     | •                | •                   |                           |                 |                            |                              |                                              |      | •                           |                        | •          |
| PHARMACEUTICAL <sup>e</sup>   | •                        |                     |                  | •                   |                           |                 |                            |                              |                                              |      | •                           |                        | •          |
| UNIVERSITIES                  | •                        |                     | •                | •                   |                           | •               |                            |                              |                                              |      | •                           |                        | •          |
| OTHER INDUSTRIES <sup>e</sup> | •                        |                     |                  | •                   |                           | •               |                            |                              |                                              |      |                             |                        |            |

<sup>a</sup> Decontamination, pickling, etching, electropolishing, etc. solutions.

<sup>b</sup> Membranes from processes such as ultrafiltration (UF) and reverse osmosis (RO).

<sup>c</sup> Decontamination and decommissioning (D & D) operations.

<sup>d</sup> Research, development, and demonstration (RD & D) programs.

<sup>e</sup> Data on these wastes are incomplete and difficult to obtain.

Table 2. Formulation Development Studies in FY 1982

| <u>Waste Type</u>    | <u>Hydraulic Cements<br/>(w. w/o additives)</u> | <u>Polymer Modified<br/>Gypsum Cement</u> | <u>Thermosetting Polymers</u> |
|----------------------|-------------------------------------------------|-------------------------------------------|-------------------------------|
| Ion exchange resins  | X                                               | X                                         | X                             |
| Incinerator          | X                                               | X                                         | X                             |
| Oils                 | X                                               | X                                         |                               |
| Organic Liquids      | X                                               | X                                         |                               |
| Nitrate Concentrates | X                                               |                                           | X                             |
| Nitrate Salt Waste   | X                                               |                                           | X                             |

## ION EXCHANGE RESIN WASTES

Formulations have been developed for the solidification of ion exchange resin wastes in portland cements, high alumina cement, polymer modified gypsum cement and vinyl ester-styrene. These studies employed

cation, anion and mixed bed Nuclear Grade Amberlite<sup>(a)</sup> "gel" type synthetic ion exchange resins. Resins were employed in both loaded and unloaded conditions. Formulations are expressed as ternary phase diagrams in terms of wt% dry resin, water and solidification agent. Dry resin weight percentages are used in this work in order to provide formulation data in a consistent manner that is universally applicable. Handling practices for ion exchange resin wastes are diverse and as a result, the percentage of water in so-called dewatered and slurry resin wastes may vary considerably.

### Hydraulic Cements

Previous work and experience with the use of hydraulic cements (portland cement) as a solidification agent for ion exchange resin wastes indicated instances of waste form expansion which resulted in low product integrity and/or waste form disintegration. This behavior has been shown to be a function of both resin type and resin content in the waste form. A large number of formulations were prepared and tested to determine acceptable formulations and to investigate the effects of cement type, resin type, resin loading and water content. Ternary compositional phase diagrams were developed for the solidification of ion exchange resin waste with portland cements and high alumina cement.

Anion, cation and mixed bed resins yielded unique regions of acceptable formulations. Anion resin formulations indicated a wide region of free standing water but initially acceptable formulations were not susceptible to degradation in the water immersion test.

Although less restricted by free standing water problems, the performance of cation resin formulations in portland cements was limited by excessive swelling and deterioration of mechanical integrity both after curing and in immersion testing. Based upon these results, the swelling and cracking of the ion exchange resin waste forms studied can be directly attributed to the cation form of the resin.

The compositional phase diagram for the solidification of mixed bed resin in portland type III cement shows a resultant envelope of acceptable formulations similar to that observed for cation resin. While the diagram does show some of the characteristics of anion resin solidification, the behavior of mixed bed resin is dominated by its cation resin component. However, waste form degradation in immersion was not observed for high alumina cement waste forms containing mixed bed resin.

(a) Rohm and Haas Co., Philadelphia, PA

A much larger range of acceptable formulations was observed for the solidification of loaded mixed bed resins in portland cement. Fewer of these formulations failed by the free standing water and immersion test criteria than with unloaded mixed bed resins.

Studies indicated that portland type III cement formulations were capable of incorporating larger quantities of water than portland type I cement and pass free standing water and immersion criteria. The minimum water requirements for workability (mixability limit) were similar for both cement types. Type III cement also appears capable of incorporating larger quantities of ion exchange resin in acceptable formulations. High alumina cement was observed to be capable of incorporating both larger quantities of water and high waste loading while meeting acceptance criteria.

Maximum resin loadings providing acceptable waste forms are listed in Table 3.

Limited work was also conducted to investigate resin pretreatment and the use of additives to cement systems to improve resin loading and product properties. These studies included use of water extendible polyester, sodium silicates, and furfuryl alcohol. Some improvements were noted.

Table 3. Maximum Resin Loadings for Cement/Resin Combinations Satisfying Acceptance Criteria

| <u>Cement Type</u> | <u>Resin Type</u> <sup>(a)</sup> | <u>Dry Resin Wt%</u> | <u>Cement Wt%</u> | <u>Water Wt%</u> |
|--------------------|----------------------------------|----------------------|-------------------|------------------|
| portland III       | loaded M.B.                      | 25                   | 22                | 53               |
| high alumina       | M.B.                             | 23                   | 35                | 42               |
| portland III       | M.B.                             | 13                   | 52                | 35               |
| portland I         | M.B.                             | 11                   | 58                | 31               |
| portland III       | cation                           | 12                   | 59                | 29               |
| portland III       | anion                            | 20                   | 31                | 49               |

(a) M.B. = mixed bed

### Polymer Modified Gypsum Cement

Formulations were developed for the solidification of mixed bed ion exchange resin waste in polymer modified gypsum cement (Enviro-stone). This agent is capable of incorporating as much as 25 wt% dry unloaded mixed bed ion exchange resin. Waste form swelling did occur during immersion testing which was pronounced for those formulations containing greater than 10 wt% dry resin. It was also noted that surface hardness decreased as a result of immersion testing. Although remaining intact for the duration of the two week immersion test, those formulations with high resin contents (>20 wt%) tended to become brittle and have poor mechanical strength when removed from the water.

The ability of the matrix to swell, thus conforming to resin bead volume increases, may account for the absence of cracking and crumbling behavior in immersion.

### Vinyl Ester-Styrene

Vinyl ester-styrene<sup>(a)</sup>, a proprietary thermosetting polymer was studied for the solidification of unloaded mixed bed dewatered and slurry resin waste streams. In this work, the resin slurry waste consisted of 20 wt% dry resin and 80 wt% water. The maximum dry resin content obtained for the solidification of slurry waste was 18.4 wt%. As much as 28.6 wt% dry resin could be incorporated if excess and interstitial water was first removed from the waste to form "damp" resin (42 wt% dry resin, 58 wt% water). The limiting constraint in solidifying resin slurry wastes was the amount of water that could be incorporated in the water-monomer emulsion during mixing. Maximum loadings of damp resin were limited by the viscosity of the mixture and the volume of binder available to provide mechanical integrity of the waste form.

### INCINERATOR ASH WASTE

Solidification studies were conducted with portland type I cement, high alumina cement, polyester-styrene and vinyl ester-styrene. This work utilized an incinerator ash produced by the rotary kiln incinerator developed at the Rockwell International Rocky Flats plant. This incinerator was used to burn a non-radioactive waste feed whose composition is representative of the combustible low-level wastes generated at the plant. This waste feed composition is listed in Table 4.

The rotary kiln ash has a bulk density of approximately 0.23 g/cm<sup>3</sup>, has a visual appearance similar to cigarette ash and is gray-white in color but exhibits color gradations and has carbon present as a distinct particulate. In the presence of water the ash exhibits a pH of approximately 6.

<sup>(a)</sup> Dow Chemical Company, Midland, MI

Table 4. Rotary Kiln Feed Compositions

| <u>Constituent</u> | <u>Weight %</u> |
|--------------------|-----------------|
| Paper              | 40.0            |
| Polyethylene       | 22.8            |
| Neoprene           | 18.8            |
| Kerosene           | 9.5             |
| Polyvinyl Chloride | 7.9             |
| Tributyl Phosphate | 1.0             |

#### Hydraulic Cements

Portland type I cement and high alumina cement were used to solidify the rotary kiln incinerator ash. Since the ash itself is dry, water must be added to this system for cement hydration. The amount of water added is important since sufficient water must be present to form a workable and pourable cement mix. However, the addition of an excessive amount of water can result in the presence of free standing water after solidification. While it is also desirable to incorporate as much incinerator ash as possible into the final waste form, sufficient cement must be included in the formulation to hold the waste form together producing a monolithic mass. Also, the cement-waste mixture must have a sufficiently long set time to allow complete mixing and transfer from the mixing equipment to the container in which the mixture is cast.

It was determined that the ash has considerable water absorption capacity. The ash was difficult to mix directly with a cement-water mixture, unless sufficient water was first added to satisfy the ash's absorptive requirements. Conversely, adequate workability was only achieved when cement powder was added to wetted ash with sufficient water present to satisfy both ash absorption and cement workability requirements. However, the water required to saturate the ash is weakly held since when sufficient water is added, a minimum of mixing typically results in an excessively wet mixture in which some water separates to the top.

Prior ball milling of the ash was found to minimize these tendencies. Ball milling decreases the ash particle size and changes its geometry, lowers ash water absorption requirements and results in a more homogeneous ash waste with a bulk density of approximately  $0.37 \text{ g/cm}^3$ .



The increased ash bulk density also facilitates fabrication of the waste form and appears to permit an increase in the amount of ash waste that can be incorporated.

A ternary compositional phase diagram was developed for the solidification of rotary kiln incinerator ash in portland type I cement. Portland type I cement can incorporate as much as 50 wt% ash in the solidified waste form. This formulation contains relatively little cement, but is sufficient to form a waste form of reasonable mechanical integrity. High alumina cement was also capable to incorporating up to 50 wt% rotary kiln ash.

#### Polyester-Styrene

Polyester-styrene resin (LB-183-13<sup>(a)</sup>) containing 50 wt% styrene was used to solidify the rotary kiln ash. Methylethyl ketone peroxide (1.2 wt% resin) was used as a catalyst and cobalt naphthenate (0.5 wt% resin) as a promoter in order to provide a room temperature polymerization. Ball-milled rotary kiln ash could be incorporated in the polyester-styrene matrix at up to 55 wt%. A maximum ash loading of 45 wt% was achieved with the as-received ash.

Work was conducted with a polyester-styrene resin containing 20 wt% styrene. This resin was considerably more viscous and difficult to mix. However, a maximum loading of 50 wt% was achieved with the ball-milled ash.

#### Vinyl Ester-Styrene

Difficulties were encountered during solidification of the rotary kiln incinerator ash with the proprietary vinyl ester-styrene<sup>(b)</sup> resin. An incomplete polymerization producing a "rubbery" solid was often observed. This is believed to result from an interaction of the catalyst and/or promoter used with components in the ash. Increases in the catalyst and promoter addition and changes in the order in which they were used improved the degree of polymerization. However, completely polymerized products were not reliably produced.

### OIL AND ORGANIC LIQUID WASTES

Work was conducted to investigate the applicability of selected solidification media for oil and organic liquid wastes. The solidification studied included various cement and cement with additive(s) systems.

(a) U.S.S. Chemical Co., Linden, NJ

(b) Dow Chemical Co., Midland, MI

Oil waste was simulated by use of a hydrocarbon vacuum pump oil.. Solidification of a liquid organic waste mixture ("NEN") generated by a radio-pharmaceutical manufacturer was also investigated. The composition of the "NEN" simulated organic liquid waste is given in Table 5. The individual solidification agents applied to the solidification of oil waste and "NEN" organic liquid waste are listed in Table 6.

TABLE 5. COMPOSITION OF "NEN" SIMULATED ORGANIC LIQUID WASTE STREAM

| Simulated Organic Liquid Waste Stream<br>Composition |      |                                             |
|------------------------------------------------------|------|---------------------------------------------|
| Component                                            | Vol% | Component<br>Formula                        |
| Methanol                                             | 22   | $\text{CH}_3\text{OH}$                      |
| Ethanol                                              | 22   | $\text{C}_2\text{H}_5\text{OH}$             |
| Acetone                                              | 9    | $\text{CH}_3\text{COCH}_3$                  |
| Isopropanol                                          | 5    | $\text{CH}_3\text{CHOHCH}_3$                |
| Diethyl ether                                        | 9    | $\text{C}_2\text{H}_5\text{OC}_2\text{H}_5$ |
| Ethyl acetate                                        | 4    | $\text{C}_2\text{H}_5\text{COOCH}_3$        |
| n-Hexane                                             | 9    | $\text{CH}_3(\text{CH}_2)_4\text{CH}_3$     |
| Benzene                                              | 4    | $\text{C}_6\text{H}_6$                      |
| Toluene                                              | 4    | $\text{C}_6\text{H}_5\text{CH}_3$           |
| Acetonitrile                                         | 4    | $\text{CH}_3\text{CN}$                      |
| 1,2-Ethylene dichloride                              | 4    | $\text{C}_2\text{H}_4\text{Cl}_2$           |
| Chloroform                                           | 4    | $\text{CHCl}_3$                             |
|                                                      | 100  |                                             |

Table 6. SOLIDIFICATION MEDIA INVESTIGATED

|                                                     | <u>Immobilization of<br/>Oil</u> | <u>Immobilization<br/>of "NEN"</u> |
|-----------------------------------------------------|----------------------------------|------------------------------------|
| Portland Type I Cement                              | X                                |                                    |
| Vermiculite and<br>Portland Type I Cement           | X                                |                                    |
| Nutek 380-Cement                                    | X                                | X                                  |
| Emulsifier, Portland Type I<br>Cement, and Silicate | X                                |                                    |
| DCM Cement                                          | X                                |                                    |
| Environstone Cement                                 | X                                | X                                  |

The results obtained in this study indicate that there are several options available for the solidification of oil or organic liquid LLWs at low waste loadings ( $\leq 10$  volume-%). For higher waste loadings, the choices are more limited and there is greater probability of obtaining unsatisfactory solids. This study concentrated on high waste loading formulations which are assumed to be more representative of actual LLW generators' practices. Based on the criteria applied in this study, acceptable solids were obtained (i.e., selected formulations yielded free-standing monolithic solids with  $< 1$  vol-% free liquid, which passed a water immersion test). However, there are indications that these formulations may not be completely satisfactory on the basis of other or evolving LLW acceptance criteria.

In the case of oil, volumetric waste loadings of the order of 30-40 volume-% may be realized with specimens of reasonable physical integrity. For "NEN", however, acceptable formulations are generally below about 10 volume-%. Especially for organic liquid wastes, alternative treatments such as incineration may be preferable to direct immobilization.

A topical report on the solidification of oil and organic liquid wastes was issued in July 1982.

#### NITRATE SALT WASTES

The solidification of nitrate salt concentrate wastes (containing up to 85 wt% sodium nitrate) and dry sodium nitrate wastes in portland type I cement, high alumina cement and vinyl ester-styrene was studied.

Ternary compositional phase diagrams were developed for each of these solidification agents. Portland type I cement was able to incorporate up to 68 wt% sodium nitrate in solidified waste forms, while the maximum quantity of sodium nitrate was 64 wt% for high alumina cement waste forms. In addition, the portland cement forms were capable of incorporating a larger quantity of water in the waste. Vinyl ester-styrene waste forms were limited to a maximum of approximately 57 wt% sodium nitrate for nitrate concentrate wastes. However, this matrix could accommodate a much larger range of water from nitrate concentrate wastes. Up to 80 wt% dry sodium nitrate could be incorporated into vinyl ester-styrene waste forms.

#### WASTE FORM PROPERTY EVALUATION STUDIES

The waste form formulations developed in this program for the solidification of various wastes were tested to determine their characteristics relative to desired waste form properties. Properties studied include leachability, mechanical properties (compressive strength, impact strength), radiation stability, and thermal properties. This data will be described in detail in the Waste Form Development Program annual report.

#### TECHNICAL ASSISTANCE

This subtask has provided assistance to the LLWMP lead contractor and/or lead laboratory as appropriate. Technical assistance efforts in FY 1982 included participation in working committees for the development of low-level waste acceptance criteria and environmental release criteria development.

#### FY 1983 EFFORTS

In FY 1983, the emphasis of this program will be the development of waste forms produced using improved agents and processes not commercially employed for low-level wastes. Such agents may include polymer concretes, low melting glasses, specially processed cements, sulfur compounds, and mineralization processes. Process development studies will be made with selected agents to ascertain the effects of processing parameters. Waste form property evaluation studies will be conducted to develop physiochemical data for improved waste forms. One or more of the improved solidification processes will be demonstrated on a scale approximate to that required in actual operation and the properties of the process demonstration product will be evaluated and compared to data obtained in previous studies for laboratory scale specimens. Technical assistance will be performed as required by the LLWMP.

## ASSAY OF LONG-LIVED RADIONUCLIDES FROM REACTORS

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## ABSTRACT

There is no comprehensive data base for nuclide concentrations of low-level wastes placed in shallow-land burial sites. One is needed for assessing long-term radioactivity burden at these sites. A data base was built in a previous program for gamma-ray-emitting nuclides and for transuranic nuclides in approximately three-hundred samples from twenty-five operating power plants. It does not include other long-lived nuclides except for fifty results from  $^{90}\text{Sr}$  analyses and ten from  $^{14}\text{C}$  analyses. However, a portion of each sample was archived and documentation exists on their origins.

Science Applications, Inc. has a program sponsored by the U.S. Nuclear Regulatory Commission (NRC) to analyze one-hundred fifty of these samples in order to extend the data base to include the important nuclides  $^3\text{H}$ ,  $^{14}\text{C}$ ,  $^{59}\text{Ni}$ ,  $^{63}\text{Ni}$ ,  $^{94}\text{Nb}$ ,  $^{129}\text{I}$ ,  $^{99}\text{Tc}$  and expand it for  $^{90}\text{Sr}$ . Radiochemical analyses would utilize proven, verified, and documented procedures. We will analyze ten samples for  $^{135}\text{Cs}$  using mass spectrometry. Samples will include resin and reactor coolant insolubles. Statistical correlation techniques will be applied to the resultant expanded data base to search for indirect methods to analyze for the difficult-to-analyze nuclides.

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INTRODUCTION

The Nuclear Regulatory Commission (NRC) will soon establish concentration limits for the long-lived radionuclides to be buried in low-level radioactive waste sites (10 CFR Part 61). Many of these nuclides do not emit measurable gamma-rays in their decay, and are measured only through complicated and time-consuming radiochemical and nuclear counting processes. Little data exist upon which to base estimates of concentrations of these nuclides in waste. In

most cases, it is not known whether typical wastes have concentrations for these nuclides that approach the limits.

An EPRI funded study (1), begun in 1976 and ended in 1979, measured concentrations of transuranics, gamma-ray-emitting nuclides and radionuclides in samples of solid waste. The study included samples collected over a period of forty-two months from twenty-five plants: eleven boiling water reactors (BWR) and fourteen pressurized-water reactors (PWR). The program collected over three-hundred samples of the principal radwaste solids of resins, filter sludge and evaporator bottoms, and of insolubles in reactor coolant. Statistical averaging over the several plants, the length of the program, and the many samples collected addressed the questions of homogeneity and representativeness of any particular sample. The study was successful. It produced the most comprehensive data base of documented samples currently available for the quantitative assessment of low-level waste-disposal issues. Furthermore, an indirect method of estimating plutonium concentrations resulted from the study; it is based on assaying the more readily measured gamma-emitting nuclide  $^{144}\text{Ce}$ . The technique is already being used by several plants.

Science Applications, Inc. has a new study program to assist the industry in implementing the waste classification system in an effective and defensible way. The purposes of the study are:

1. To provide a data base of typical concentrations of the long-lived nuclides in representative radioactive waste from nuclear power plants.
2. To identify indirect measurements that can estimate the concentrations of the difficult-to-analyze nuclides in waste.

A portion of each sample from the EPRI program was archived. The samples are documented as to source, plant and time, and many are of an adequate quantity to analyze for additional radionuclides. These samples will be analyzed further for  $^3\text{H}$ ,  $^{14}\text{C}$ ,  $^{59}\text{Ni}$ ,  $^{63}\text{Ni}$ ,  $^{94}\text{Nb}$ ,  $^{99}\text{Tc}$  and  $^{129}\text{I}$ . Since only fifty of the samples were previously analyzed for  $^{90}\text{Sr}$ , we will analyze more of the samples for this nuclide.

## EXPERIMENTAL MEASUREMENTS

### SELECTION OF EXPERIMENTAL MEASUREMENTS SAMPLES

The samples to be assayed for additional nuclides are 52 samples of reactor coolant insolubles, 33 samples of filter sludges, 35 samples of spent ion-exchange resins, and 30 samples of evaporator bottoms. The samples came from eight BWR and seven PWR sites.

The criteria for choosing these samples were the following:

1. Extensive analyses were done on the sample. These analyses included the actinides, the gamma-ray-emitting nuclides, and, in some cases, strontium.
2. Enough of the sample remains and its condition is satisfactory so that representative analyses can be performed.
3. The origin of the sample is definite, and the sample is of a principal source of waste.

The program includes the following elements:

1. Survey the literature to determine effective analytic procedures, and establish an efficient analytical program that will utilize some of our established analytical procedures and the required new procedures.
2. Establish quality control and quality assurance programs for the project. These would be in addition to our existing quality assurance program and will include verification with standards, internal reproducibility studies, and a sample splitting round-robin program with another laboratory.
3. Determine whether the previously analyzed samples are valid for this program by reassaying eight of them for uranium, transuranic and gamma-ray-emitting nuclides. These results will be compared with (a) the earlier results and (b) the results of assays for these radionuclides in eight new waste samples chosen by SAI.
4. Perform analyses for tritium,  $^{14}\text{C}$ ,  $^{59}\text{Ni}$  and  $^{63}\text{Ni}$ ,  $^{90}\text{Sr}$ ,  $^{94}\text{Nb}$ ,  $^{99}\text{Tc}$ , and  $^{129}\text{I}$  on 150 samples of wastes from both BWR and PWR power plants. Ten archived samples will be assayed for  $^{135}\text{Cs}$ .
5. Twenty new samples chosen by the NRC, ten each from a BWR and a PWR will be assayed completely and added to the base.
6. Establish a data base of results that will identify the plant sample origin, sampling date, nuclide concentrations and uncertainties.
7. Perform computer-aided searches for concentration correlations between the nuclides of interest and the gamma-ray-emitting nuclides.

As was pointed out in the original study (1), the sample representativeness is a serious problem in radwaste sampling. The sampling program of that study was designed to reduce this problem through the time-averaging inherent in a forty-two month study. Hence, the samples are thought to be statistically representative.

#### Sample Preparation

The regulations in 10 CFR Part 61 impose the restrictions in terms of an activity per unit volume. Radiochemical techniques are more readily applied in terms of activity per unit weight. We will

obtain the analytical results of the new samples in terms of both units. The procedure for sample preparation is the following:

- o Slurry the sample as received into a container of standard size and known tare using a metered amount of water.
- o Let the solids settle; weigh and record the net weight of the solids (subtracting the weight of the liquid overburden) and the volume of solids (height) to obtain the sample wet weight and density.
- o Evaporate to dryness carefully at low temperature to assure the retention of the volatile nuclides such as  $^{129}\text{I}$ .
- o Weigh the sample and measure the volume to obtain dry sample weight and density.
- o Perform the gamma-ray analysis on the entire sample.
- o Take weighed aliquots of the dried sample for chemical analyses.
- o Archive part of the sample for possible reanalysis.

#### Analytical Techniques

Methods Selection. SAI has an analytical program with an efficient method to analyze for all the desired nuclides. A flow chart is shown in Figure 1. It is preliminary pending completion of the testing of non-routine procedures. Individual radiochemical procedures for the nuclides are either SAI methods now in use or were selected from a literature survey. The procedures to be used have been submitted for peer review by two other laboratories.

Methods Testing. Before beginning assays of the waste samples obtained for the program, we are testing analytical procedures for radionuclides that we do not regularly use. For this purpose we are using spiked, simulated waste samples, and previously archived reactor-waste samples. The simulated sample is homogenized, divided, part of it spiked, and both fractions analyzed. The entire final analytical flow path is to be subsequently tested on simulated and real samples.

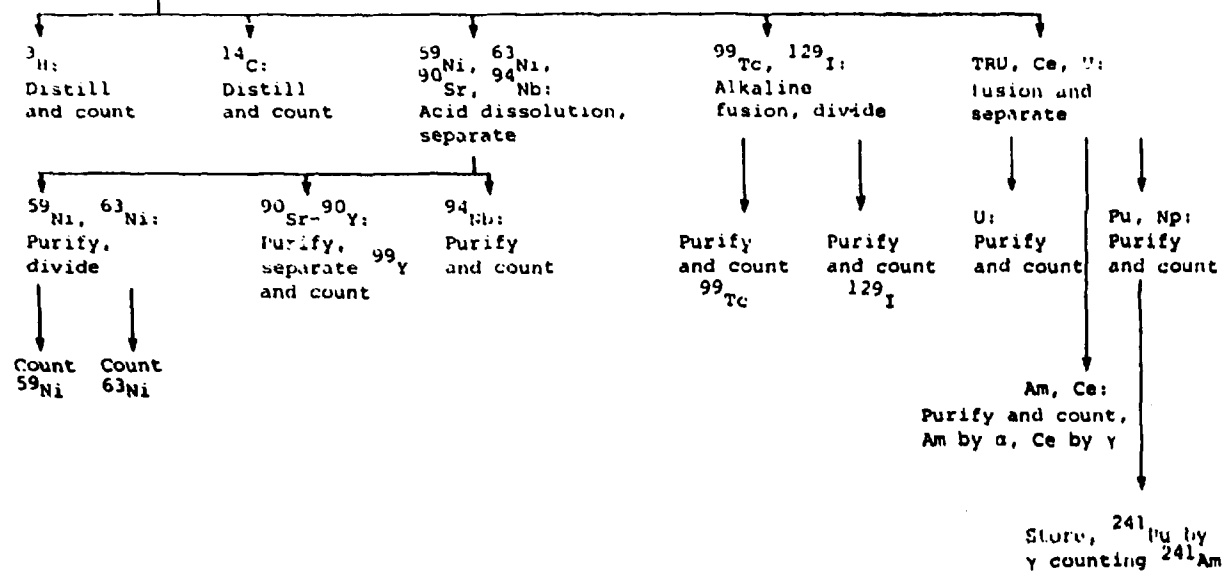
SAI Gamma-Ray Spectrometer System and Data Reduction. We will analyze new samples prepared for gamma-ray analysis with 4096-channel Ge(Li) gamma-ray spectrometers. The spectrometer system was calibrated for detection efficiencies using a technique based on the use of NBS standards (2). Each spectrometer has a structure for positioning samples precisely. In addition, gamma-ray analyses will be done on the electrodeposited plutonium fractions after a suitable delay time, and on the separated trivalent-actinide fractions. Plutonium plates will be counted to measure the growth of  $^{241}\text{Am}$  in order to calculate the amount of  $^{241}\text{Pu}$  present at the time of separation. The americium-curium solutions will be counted before electrodeposition to measure the lanthanides present. Cerium and europium isotopes can usually be seen.



Figure 1

## Schematic of the Sample Chemical Analysis Program

Receive and log sample  
Determine physical properties  
Perform Ge γ-ray spectrometry  
Subsample by mass (dry)



Report results in  $\mu\text{Ci/g}$  dry weight and  $\mu\text{Ci/cm}^3$  wet weight

Gamma-ray data reduction is performed by an automated analysis utilizing a computer-based analyzer system. The results are subsequently evaluated by the experimenter. The spectrometer computer system automatically searches the acquired pulse-height spectrum for peaks, calculates the energy, the gross and net counts, and identifies the radionuclide giving rise to each peak. From this information, the concentration in  $\mu\text{Ci/unit weight}$  for each nuclide identified in the sample is determined and corrected for radioactive decay to the time of the sampling. The system subtracts the background radiation contribution for the identified nuclides. For those radionuclides not identified from the nuclide table, calculations are performed of the minimum detectable limits.

Results of Analyses. The results of the measurements will be reported as radioactivity concentrations in terms of  $\mu\text{Ci}\cdot\text{g}^{-1}$  dry weight for sludges, resins, and evaporator bottoms, and in  $\mu\text{Ci}\cdot\text{ml}^{-1}$  of filtered liquid for the filters of the reactor coolant samples. The counting uncertainty (Poisson error) will be given for each analysis. Estimates will be provided of systematic uncertainties associated with each type of analysis. For new samples, which would be solids, the measurement results will also be expressed in  $\mu\text{Ci}\cdot\text{cm}^{-3}$ , wet weight.

#### IDENTIFICATION OF NUCLIDE CORRELATIONS AND INDIRECT MEASUREMENT TECHNIQUES

##### ESTABLISHMENT OF COMPUTER DATA BASE

We are creating a data base on our PDP 11/34 computer. The base will include analyses data from this program as well as any other relevant data such as those from commercial analyses and from the previous EPRI study. These data are to be put on magnetic tape and made available to the NRC or its designates. Data in the base will be from analyses of samples obtained from the plants that are listed in Table 1. An example of the contents is given in Table 2.

##### CORRELATION TECHNIQUES

We believe that the best technique to estimate concentrations of the difficult nuclides is through the use of empirically determined scaling factors. Other approaches are hampered by difficulties in the source terms to the particular radwaste system, in the transport mechanism and in the radwaste operation itself. The existence of these difficulties is corroborated (1) by our previous work in radwaste sample analysis.

Table 1

## PLANTS LISTED IN THE DATA BASE

Samples Analyzed For TRU, Gamma-Ray Emitters and Strontium

| <u>BWR</u>           | <u>PWR</u>           |
|----------------------|----------------------|
| Big Rock Point *     | Calvert Cliffs 1,2 * |
| Browns Ferry 1,2,3 * | Connecticut Yankee * |
| Brunswick 1,2 Δ      | R. E. Ginna *        |
| Dresden 1,2,3        | Kewaunee             |
| Duane Arnold Δ       | Maine Yankee *       |
| J. A. Fitzpatrick    | Millstone 2          |
| LaCrosse *           | North Anna 1,2       |
| Millstone 1 *        | Oconee 1,2,3 *       |
| Monticello *         | Point Beach 1,2 *    |
| Oyster Creek *       | Prairie Island 1,2   |
| Pilgrim *            | Robinson 2           |
| Quad Cities 1,2 *    | Surry 1,2            |
| Vermont Yankee *     | TMI-2 Δ              |
|                      | Trojan Δ             |
|                      | Yankee Rowe *        |
|                      | Zion 1,2 *           |
| <hr/> 19             | <hr/> 24             |

\* Some samples from these plants will be analyzed for all of the long-lived nuclides.

Δ Some samples from these plants have been analyzed for  $^{63}\text{Ni}$ .

Table 2

## Radwaste Nuclide Data Base Structure

| PLANT NAME     | PILGRIM         | QUAD CITY       | OSTER CREEK     |
|----------------|-----------------|-----------------|-----------------|
| UNIT NUMBER    | 0               | 1               | 0               |
| PLANT TYPE     | SWR             | BWR             | BWR             |
| PLANT LOCATION | SEACOAST        | INLAND          | SEACOAST        |
| START-UP DATE  | 01-DEC-72       | 01-AUG-72       | 01-DEC-68       |
| SAMPLE TYPE    | REACTOR COOLANT | REACTOR COOLANT | REACTOR COOLANT |
| SAMPLING DATE  | 08-OCT-76       | 14-MAR-78       | 15-MAR-77       |
| SAMPLE NUMBER  | T050            | T040            | T094            |
| DRY SPEC GRAV  | 1.00E+00        | 1.00E+00        | 1.00E+00        |
| WET SPEC GRAV  | 1.80E+00        | 1.80E+00        | 1.80E+00        |

| CONCENTRATIONS IN MICROCURIES/GRAM |          |          |          |          |          |          |
|------------------------------------|----------|----------|----------|----------|----------|----------|
|                                    | VALUE    | UNC.     | VALUE    | UNC.     | VALUE    | UNC.     |
| 3-H                                |          |          |          |          |          |          |
| 14-C                               |          |          |          |          |          |          |
| 54-MN                              | 4.10E-04 | 1.00E-05 | 7.37E-06 | 3.70E-07 | 3.40E-05 | 2.00E-06 |
| 58-CO                              | 1.80E-04 | 1.00E-05 | 2.77E-05 | 1.40E-06 | 3.10E-06 | 2.00E-07 |
| 60-CO                              | 1.21E-03 | 6.00E-05 | 7.00E-05 | 3.50E-06 | 3.10E-05 | 2.00E-06 |
| 59-NI                              |          |          |          |          |          |          |
| 60-NI                              |          |          |          |          |          |          |
| 90-SR                              |          |          |          |          |          |          |
| 90-ZR                              | 7.20E-05 | 8.00E-06 | 1.31E-05 | 7.60E-06 | 6.00E-06 | 9.00E-07 |
| 94-NB                              |          |          |          |          |          |          |
| 92-TC                              |          |          |          |          |          |          |
| 105-RU                             | 6.60E-06 | 1.90E-06 |          |          | 6.60E-07 | 2.00E-07 |
| 110M AG                            |          |          |          |          |          |          |
| 125-SB                             | 1.80E-06 | 1.20E-06 |          |          |          |          |
| 127-I                              |          |          |          |          |          |          |
| 134-CS                             |          |          | 1.16E-06 | 2.10E-07 | 3.20E-07 | 1.00E-07 |
| 135-CS                             |          |          |          |          |          |          |
| 137-CS                             | 1.60E-06 | 1.00E-06 | <8.9E-08 |          | 2.40E-07 | 9.00E-08 |
| 141-CE                             | 1.21E-04 | 6.00E-06 | 8.33E-06 | 4.20E-07 | 7.90E-06 | 4.00E-07 |
| 144-CE                             | 9.10E-05 | 5.00E-06 | 5.44E-06 | 3.70E-07 | 4.30E-06 | 1.50E-06 |
| 152-EU                             |          |          |          |          |          |          |
| 154-EU                             | 9.20E-07 | 3.00E-07 |          |          |          |          |
| 155-EU                             | 5.80E-07 | 2.00E-07 |          |          |          |          |
| 235-U                              | 1.40E-09 |          |          |          | <7.0E-11 |          |
| 238-U                              | 1.40E-09 |          |          |          | <7.0E-11 |          |
| 238-FU                             | 9.10E-08 | 1.00E-08 | 1.12E-08 | 6.00E-10 | 2.10E-08 | 2.00E-09 |
| 239-FU                             | 1.20E-07 | 2.00E-08 | 4.90E-09 | 3.00E-10 | 2.20E-08 | 2.00E-09 |
| 241-FU                             |          |          |          |          |          |          |
| 241-AM                             | 1.10E-07 | 3.00E-08 | 4.30E-09 | 6.00E-10 | 1.40E-09 | 2.00E-10 |
| 247-CM                             | 1.00E-06 | 2.00E-07 | 1.80E-07 | 2.00E-08 | 2.30E-07 | 2.00E-08 |
| 244-CM                             | 3.70E-08 | 5.00E-09 | 8.80E-09 | 1.00E-09 | 8.10E-09 | 1.00E-09 |

The mathematical technique proposed to search for correlations is to fit the data to linear or polynomial functions of the concentrations of the two correlating nuclides. Accepted methods of non-linear least squares will be used to calculate the correlation coefficients. The results will be scanned for true correlations. SAI proposes to do these computations on the computer data base using search and computational algorithms.

#### POSSIBLE NUCLIDE CORRELATIONS

We will look for correlations with the nuclides described below. If usable correlations are found, they can serve to provide indirect measurements for the nuclides that are difficult to measure. These correlations would serve as scaling factors for estimating burial burdens of these nuclides in wastes. Table 3 shows the nuclides for which correlations are desired and the nuclide or technique that seems at present to offer the best chance of success.

#### QUALITY CONTROL

We are testing all of the chemical procedures with spiked simulated radioactive waste samples prior to analyzing real samples. Thereafter, we will analyze quality-control samples concurrently with actual samples by measuring a reagent blank and a spiked blank for every twenty sample analyses. Radioactive material for these spikes will be standard reference material from various suppliers and will be traceable. We will also spike real but previously analyzed radioactive samples so that the total analysis procedure can be verified. For every twenty sample analyses, we will analyze duplicate (split) samples in our own facility to determine the analytic reproducibility. SAI will also have a sample-splitting round-robin program with another laboratory.

#### REFERENCE

1. J. E. Cline, K. L. Wright and J. W. Hollcroft. Activity Levels of Transuranic Nuclides in Low-Level Solid Wastes From U.S. Power Reactors, Electric Power Research Institute, EPRI-NP-1494, August 1980.
2. J. E. Cline, A Technique of Gamma-Ray Detector Absolute Efficiency Calibration for Extended Sources, CONF-780421, 185, 1979, ANS Topical Conference at Mayaguez, Puerto Rico.

Table 3

## POTENTIAL INDIRECT ANALYSES

| <u>Nuclide</u>                                | <u>Possible Indirect Nuclide</u>                                    |
|-----------------------------------------------|---------------------------------------------------------------------|
| Tritium                                       | Reactor Coolant Analyses for tritium, sample water content          |
| $^{14}\text{C}$                               | Unknown                                                             |
| $^{59}\text{Ni}$ , $^{63}\text{Ni}$           | $^{58}\text{Co}$ , another insoluble activation product from nickel |
| $^{90}\text{Sr}$                              | $^{137}\text{Cs}$ if soluble; $^{144}\text{Ce}$ if insoluble        |
| $^{99}\text{Tc}$                              | $^{137}\text{Cs}$ if soluble, $^{144}\text{Ce}$ if insoluble        |
| $^{129}\text{I}$                              | $^{137}\text{Cs}$                                                   |
| $^{135}\text{Cs}$                             | $^{137}\text{Cs}$                                                   |
| $^{235,238}\text{U}$                          | $^{144}\text{Ce}$                                                   |
| $^{239,240}\text{Pu}$                         | $^{144}\text{Ce}^*$                                                 |
| $^{241}\text{Pu}$                             | Plant specific, unknown                                             |
| $^{241}\text{Am}$ , $^{242,243,244}\text{Cm}$ | Plant specific, unknown                                             |

\* Demonstrated previously.

# **Radionuclide Correlations in Radioactive Wastes**

**An Experimental Program**



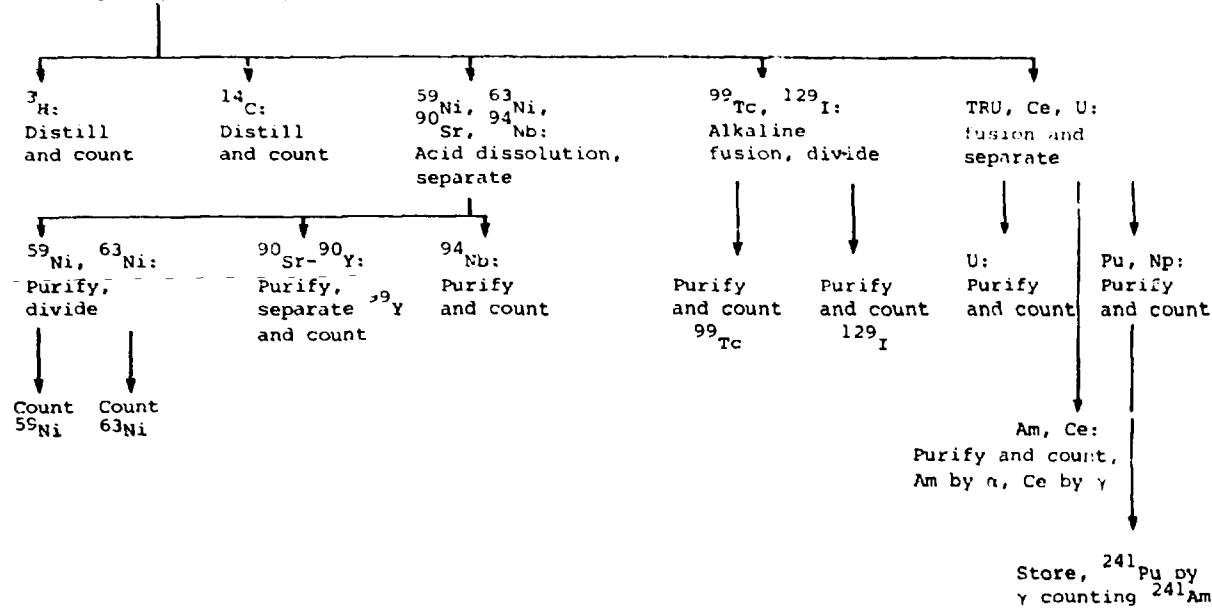
## **Features of the Program**

- **Sampling**
- **Analyses**
- **Data Base**
- **Nuclide Correlations**
- **Quality Control/Assurance**



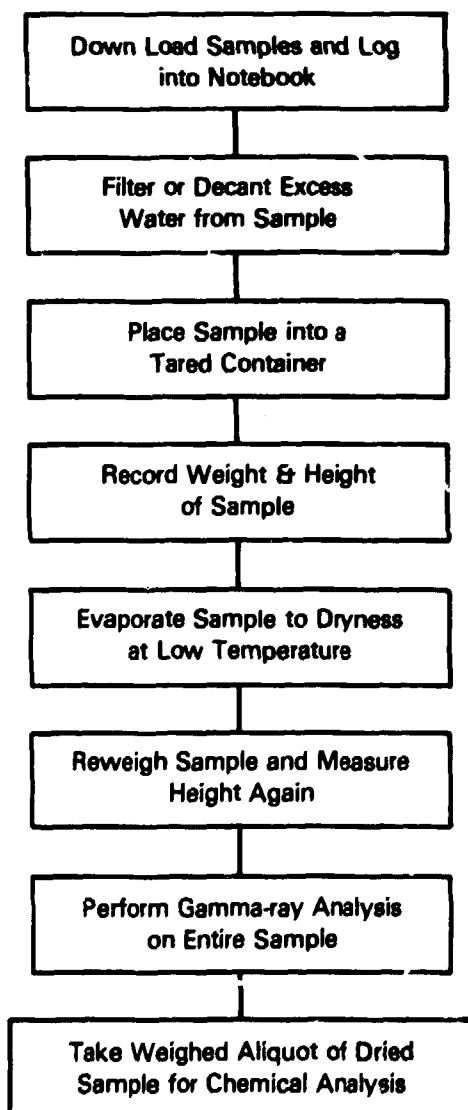
# Schematic of the Sample Chemical Analysis Program

Receive and log sample  
Determine physical properties  
Perform Ge  $\gamma$ -ray spectrometry  
Subsample by mass (dry)



Report results in  $\mu\text{Ci/g}$  dry weight and  $\mu\text{Ci/cm}^3$  wet weight

## Sample Preparation



## Sensitivities and Requirements

| Isotope              | Sensitivity<br>$\mu\text{Ci}/\text{cm}^3$ | NRC (10 CFR Part 61)<br>$\mu\text{Ci}/\text{cm}^3$ |
|----------------------|-------------------------------------------|----------------------------------------------------|
| $^3\text{H}$         | $2 \times 10^{-6}$                        | 40                                                 |
| $^{14}\text{C}$      | $1 \times 10^{-6}$                        | 0.8                                                |
| $^{53}\text{Ni}$     | $3 \times 10^{-4}$                        | 2.2                                                |
| $^{63}\text{Ni}$     | $1 \times 10^{-5}$                        | 3.5                                                |
| $^{94}\text{Nb}$     | $6 \times 10^{-5}$                        | 0.002                                              |
| $^{90}\text{Sr}$     | $6 \times 10^{-6}$                        | 0.04                                               |
| $^{99}\text{Tc}$     | $2 \times 10^{-5}$                        | 0.3                                                |
| $^{129}\text{I}$     | $5 \times 10^{-5}$                        | 0.008                                              |
| $^{235,238}\text{U}$ | $2 \times 10^{-7}$                        | 0.04                                               |
| TRU                  | $2 \times 10^{-7}$                        | 0.001                                              |
| $^{241}\text{Pu}$    | $6 \times 10^{-6}$                        | 0.35                                               |

## **Elements of Quality Assurance Program**

- **Testing of Analytical Methods**
- **Spiked Blanks**
- **Instrumentation Calibration**
- **Replicate Analyses of Samples**
- **Participation in Previous Round-robins**
- **Participation in Planned Round-robins**
- **QA Documentation**

## **Samples in the Data Base**

|                                               |            |
|-----------------------------------------------|------------|
| ● Original EPRI Program                       | 200        |
| ● Commercial Samples<br>(30 Operating Plants) | 250        |
| Total                                         | <u>450</u> |

## Plants Listed in the Data Base

### Samples Analyzed for TRU, Gamma-ray Emitters and Strontium:

#### BWR

Big Rock Point\*  
 Brown's Ferry 1,2,3\*  
 Brunswick 1,2<sup>†</sup>  
 Dresden 1,2,3  
 Duane Arnold<sup>†</sup>  
 J.A. Fitzpatrick  
 La Crosse\*  
 Millstone 1\*  
 Monticello\*  
 Oyster Creek\*  
 Pilgrim\*  
 Quad Cities 1,2\*  
 Vermont Yankee\*

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\*Some samples from these plants will be analyzed for all of the long-lived nuclides.

<sup>†</sup>Some samples from these plants have been analyzed for <sup>63</sup>Ni.

#### PWR

Calvert Cliffs 1,2\*  
 Connecticut Yankee\*  
 R.E. Ginna\*  
 Kewaunee  
 Maine Yankee\*  
 Millstone 2  
 North Anna 1,2  
 Oconee 1,2,3\*  
 Point Beach 1,2\*  
 Prairie Island 1,2  
 Robinson 2  
 Surry 1,2  
 TMI 2<sup>†</sup>  
 Trojan<sup>†</sup>  
 Yankee Rowe\*  
 Zion 1,2\*

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## Radwaste Nuclide Data Base Structure

|                 |                 |                 |                 |
|-----------------|-----------------|-----------------|-----------------|
| PLANT NAME      | PILGRIM         | QUAD CITIES     | OYSTER CREEK    |
| UNIT NUMBER     | 0               | 1               | 0               |
| PLANT TYPE      | BWR             | BWR             | BWR             |
| PLANT LOCATION  | SEACOAST        | INLAND          | SEACOAST        |
| START-UP DATE   | 01-DEC-72       | 01-AUG-72       | 01-DEC-69       |
| SAMPLE TYPE     | REACTOR COOLANT | REACTOR COOLANT | REACTOR COOLANT |
| SAMPLING DATE   | 08-OCT-76       | 14-MAR-78       | 15-MAR-77       |
| SAMPLE NUMBER   | T050            | T04B            | T094            |
| DRY SPEC GRAV.  | 1.00E+00        | 1.00E+00        | 1.00E+00        |
| WET SPEC. GRAV. | 1.80E+00        | 1.80E+00        | 1.80E+00        |

### CONCENTRATIONS IN MICROCURIES/GRAM

|         | VALUE    | UNC.     | VALUE    | UNC.     | VALUE    | UNC.     |
|---------|----------|----------|----------|----------|----------|----------|
| 3-H     |          |          |          |          |          |          |
| 14-C    |          |          |          |          |          |          |
| 54-MN   | 4.10E-04 | 1.00E-05 | 7.37E-06 | 3.70E-07 | 3.40E-05 | 2.00E-06 |
| 58-CO   | 1.80E-04 | 1.00E-05 | 2.79E-05 | 1.40E-06 | 3.10E-06 | 2.00E-07 |
| 60-CO   | 1.21E-03 | 6.00E-05 | 7.00E-05 | 3.50E-06 | 3.10E-05 | 2.00E-06 |
| 59-NI   |          |          |          |          |          |          |
| 63-NI   |          |          |          |          |          |          |
| 90-SR   |          |          |          |          |          |          |
| 95-ZR   | 7.20E-05 | 8.00E-06 | 1.31E-05 | 2.60E-06 | 4.00E-06 | 9.00E-07 |
| 94-NB   |          |          |          |          |          |          |
| 99-TC   |          |          |          |          |          |          |
| 106-RU  | 6.60E-06 | 1.90E-06 |          |          | 6.60E-07 | 2.00E-07 |
| 110M-AG |          |          |          |          |          |          |
| 125-SB  | 1.80E-06 | 1.20E-06 |          |          |          |          |
| 129-I   |          |          |          |          |          |          |
| 134-CS  |          |          | 1.16E-06 | 2.10E-07 | 3.20E-07 | 1.00E-07 |
| 135-CS  |          |          |          |          |          |          |
| 137-CS  | 1.60E-04 | 1.00E-06 | <8.9E-08 |          | 2.40E-07 | 9.00E-08 |
| 141-CE  | 1.21E-04 | 6.00E-06 | 8.33E-06 | 4.20E-07 | 7.90E-06 | 4.00E-07 |
| 144-CE  | 9.10E-05 | 5.00E-06 | 5.44E-06 | 3.90E-07 | 4.30E-06 | 1.50E-06 |
| 152-EU  |          |          |          |          |          |          |
| 154-EU  | 9.20E-07 | 3.00E-07 |          |          |          |          |
| 155-EU  | 5.80E-07 | 2.00E-07 |          |          |          |          |
| 235-U   | <4.0E-09 |          |          |          | <7.0E-11 |          |
| 238-U   | <4.0E-09 |          |          |          | <7.0E-11 |          |
| 238-PU  | 9.10E-08 | 1.00E-08 | 1.12E-08 | 6.00E-10 | 7.10E-08 | 2.00E-09 |
| 239-PU  | 1.20E-07 | 2.00E-08 | 4.90E-09 | 3.00E-10 | 2.20E-08 | 2.00E-09 |
| 241-PU  |          |          |          |          |          |          |
| 241-AM  | 1.10E-07 | 3.00E-08 | 4.30E-09 | 6.00E-10 | 1.40E-09 | 2.00E-10 |
| 242-CM  | 1.00E-06 | 2.00E-07 | 1.80E-07 | 2.00E-08 | 2.30E-07 | 2.00E-08 |
| 244-CM  | 3.70E-08 | 5.00E-09 | 8.80E-09 | 1.00E-09 | 8.10E-09 | 1.00E-09 |

## Potential Indirect Analyses

|                                                                   |                                                                     |
|-------------------------------------------------------------------|---------------------------------------------------------------------|
| Tritium .....                                                     | Reactor Coolant Analyses for tritium, sample water content          |
| $^{14}\text{C}$ .....                                             | Unknown                                                             |
| $^{59}\text{Ni}$ , $^{63}\text{Ni}$ .....                         | $^{58}\text{Co}$ , another insoluble activation product from nickel |
| $^{90}\text{Sr}$ .....                                            | $^{137}\text{Cs}$ if soluble, $^{144}\text{Ce}$ if insoluble        |
| $^{99}\text{Tc}$ .....                                            | $^{137}\text{Cs}$ if soluble, $^{144}\text{Ce}$ if insoluble        |
| $^{129}\text{I}$ .....                                            | $^{137}\text{Cs}$                                                   |
| $^{135}\text{Cs}$ .....                                           | $^{137}\text{Cs}$                                                   |
| $^{235}$ , $^{238}\text{U}$ .....                                 | $^{144}\text{Ce}$                                                   |
| $^{239}$ , $^{240}\text{Pu}$ .....                                | $^{144}\text{Ce}$                                                   |
| $^{241}\text{Pu}$ .....                                           | Plant specific, unknown                                             |
| $^{241}\text{Am}$ , $^{242}$ , $^{243}$ , $^{244}\text{Cm}$ ..... | Plant specific, unknown                                             |



## Relative Concentrations of TRU

| PWR Plant    | Sample             | $^{23}\text{Pu}$ | $^{239+240}\text{Pu}$ | $^{241}\text{Am}$ | $^{242}\text{Cm}$ | $^{244}\text{Cm}$ |
|--------------|--------------------|------------------|-----------------------|-------------------|-------------------|-------------------|
| Point Beach  | Coolant            | 0.12             | 0.16                  | 0.04              | 0.64              | 0.05              |
|              | Evaporator Bottoms | 0.18             | 0.21                  | 0.10              | 0.42              | 0.09              |
|              | Filter             | 0.18             | 0.28                  | 0.05              | 0.44              | 0.05              |
|              | Crud               | 0.15             | 0.20                  | 0.20              | 0.41              | 0.06              |
| Yankee Rowe  | Coolant            | 0.17             | 0.39                  | 0.05              | 0.36              | 0.008             |
|              | Filter             | 0.12             | 0.27                  | 0.08              | 0.50              | 0.012             |
|              | Evaporator Bottoms | 0.46             | 0.14                  | 0.12              | 0.18              | 0.03              |
|              | Crud               | 0.20             | 0.44                  | 0.04              | 0.30              | 0.005             |
| Zion         | Coolant            | 0.21             | 0.16                  | 0.03              | 0.59              | 0.011             |
|              | Evaporator Bottoms | 0.22             | 0.28                  | 0.16              | 0.31              | 0.013             |
|              | Resin              | 0.47             | 0.18                  | 0.14              | 0.20              | 0.02              |
| Conn. Yankee | Coolant            | 0.39             | 0.15                  | 0.03              | 0.38              | 0.05              |
|              | Resin              | 0.11             | 0.04                  | 0.05              | 0.75              | 0.07              |
|              | Evaporator Bottoms | 0.28             | 0.11                  | 0.08              | 0.35              | 0.18              |
|              | Crud               | 0.15             | 0.04                  | 0.012             | 0.74              | 0.05              |

## MONITORING AND PHYSICAL CHARACTERIZATION OF UNSATURATED ZONE TRANSPORT

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### ABSTRACT

The goal of the Unsaturated Zone project has been to develop and document the predictive capability for the transport of radionuclides and water. The project has progressed to the point where a critical evaluation of unsaturated zone models can be made. Transport parameters have been measured and experimental data sets have been collected that can be used in comparative modeling studies. These modeling studies, when completed, will provide information on how unsaturated zone transport can be modeled in the simplest manner, while retaining sufficient accuracy.

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### INTRODUCTION

This project has as its goal, the development and documentation of a predictive capability for unsaturated transport of radionuclides and water. Specific topics of research include: 1) the evaluation of monitoring methods; 2) the measurement of transport parameters used in modeling; 3) the construction of experimental data sets describing water and radionuclide transport; and 4) the use of these data to perform comparative evaluations of transport models.

Data collection has included both laboratory and field experiments. Field data has been collected at the Buried Waste Test Facility (BWTF) since late 1978. The BWTF (Figure 1) consists of six 8m deep caissons and two 1.5m deep weighing lysimeters. Water content, water potential, and soil temperature are monitored continuously together with atmospheric parameters to provide data necessary to predict the overall water balance of the facility. Irrigation of half the lysimeters is provided to evaluate the effect of enhanced precipitation on deep drainage and evaporation. Four of the six 8m caissons are used to conduct tracer tests with tritium and cobalt-60 that has been complexed with EDTA. These tracer tests have been designed to provide data on field scale hydrodynamic dispersion, as well as the effect of organic complexants on the transport of radionuclides.

# BURIED WASTE TEST FACILITY

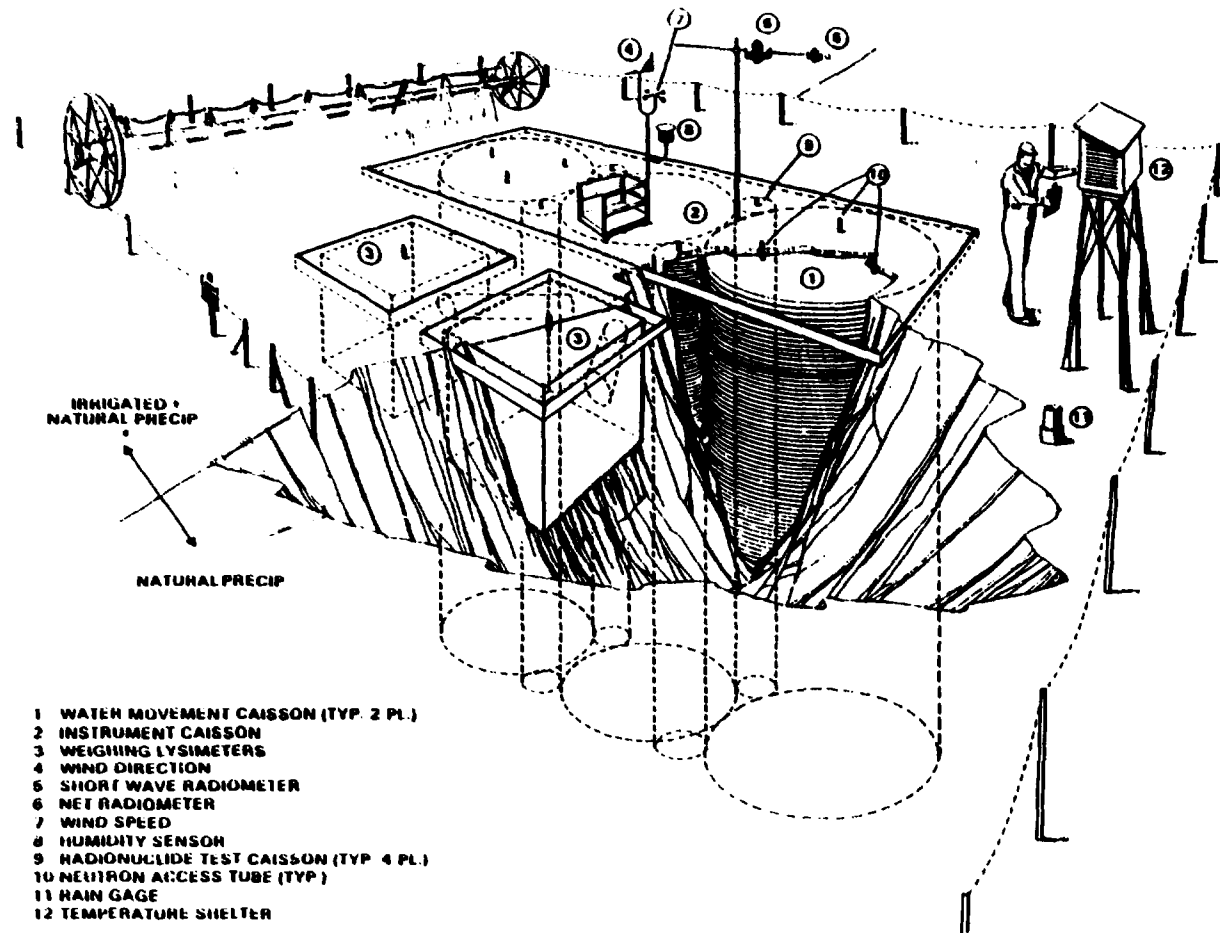


Figure 1. Buried waste test facility.

## PROGRESS TO DATE

As of the end of FY 82, work on the first three research topics described above has been completed. These are the evaluation of monitoring techniques, the measurement of transport parameters and the collection of experimental data sets describing water and radionuclide transport.

### Monitoring methods

An evaluation of the monitoring methods necessary to study unsaturated transport revealed two areas that needed attention. The laboratory measurement of tritium in soil samples, and the measurement of soil water potential. A comparison was made of two tritium measurement techniques. The leach method, where water is added to the contaminated sample and the tritium is leached out of the soil, and a method where the soil sample is vaporized in a "sample oxidizer" and the tritium vapor is condensed and collected. Both techniques provide samples for liquid scintillation analysis. The comparison revealed that both methods are comparable in terms of accuracy and precision; however, the oxidizer technique was able to detect lower concentrations of tritium than the leach method. The oxidizer method is also much faster. The primary disadvantage is the initial capital expense to buy the sample oxidizer equipment (approximately \$16K).

The measurement of soil water potential in low-level waste management activities was evaluated during FY 82 and a report issued (1). Two areas were identified where the measurement of water potential was needed in waste management. These were site characterization studies, and the long-term monitoring program that will follow the operational phase of a burial site. An evaluation of the instrumentation available revealed that two different types of sensors were needed for the two applications. For the site characterization studies the accuracy of tensiometers is required, while for the long-term monitoring programs the stability and reliability of gypsum or fiberglass blocks is preferred.

### Measurement of transport parameters

Figure 2 shows a listing of the specific transport parameters being investigated as well as the data sets being collected in laboratory studies and at the BWTF. The measurement of water and heat transport parameters has been completed this year with the evaluation of the non-isothermal water vapor diffusivity. Data reported last year (2) showed that the hydraulic conductivity could be evaluated consistently by several methods. The successful measurement of thermal conductivity as a function of water content was reported in Cass et al (3). These measurements formed the basis of the evaluation of the non-isothermal vapor diffusivity reported in Cass et al (4). A key component in this diffusivity term is the water vapor enhancement factor shown in Figure 3. The experimental data shown in Figure 3 show that the two theoretical

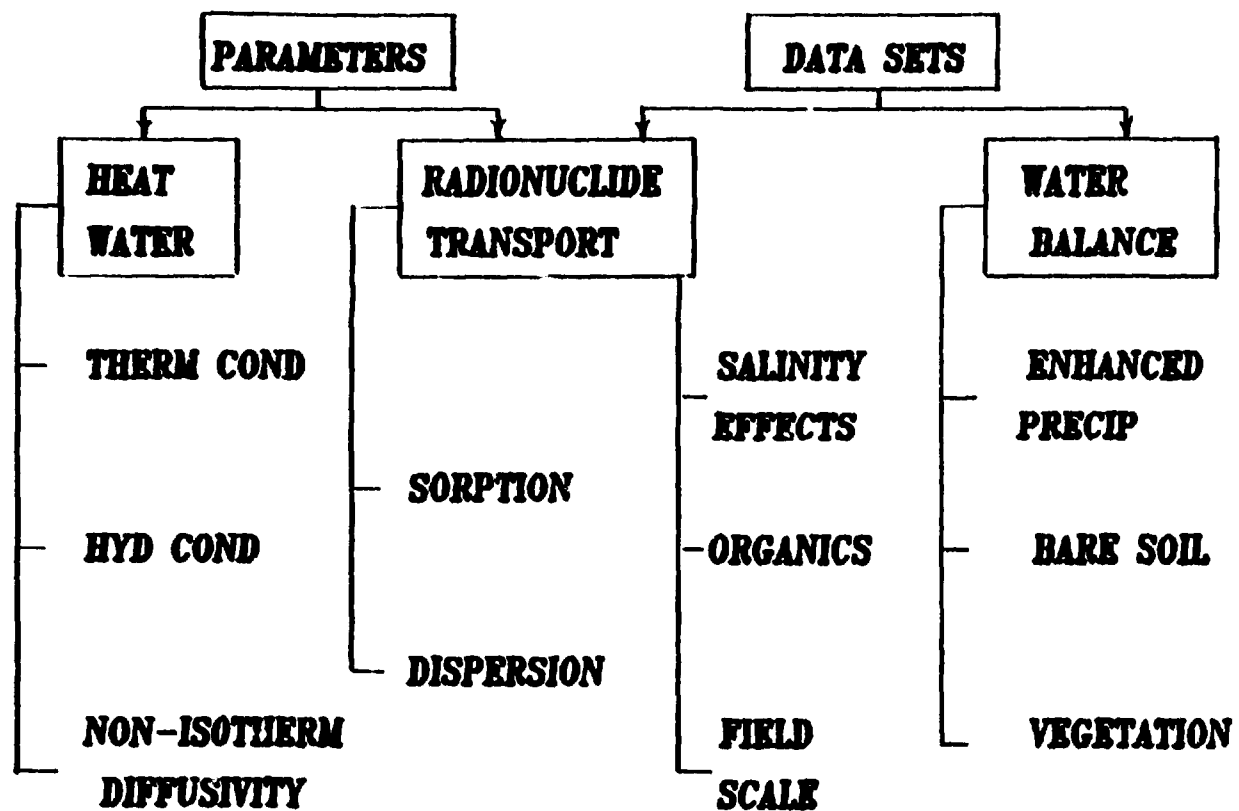


Figure 2. List of transport parameters and experimental data sets being investigated.

# **WATER VAPOR FLOW ENHANCEMENT FACTORS FOR BWTF SOIL MATERIAL AT 35°C**

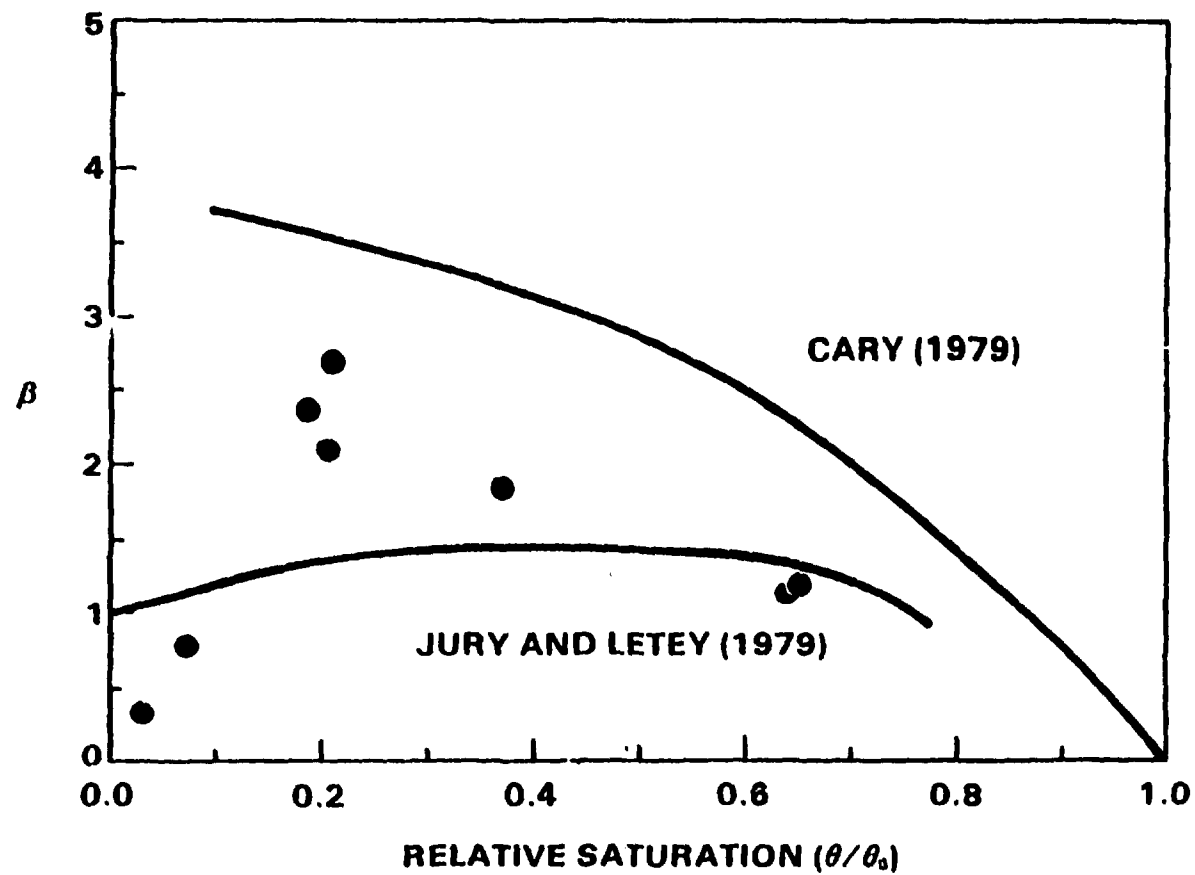


Figure 3. Water vapor flow enhancement factor ( $\beta$ ).

models available in the literature do not properly describe the dependence of the enhancement factor on water content. The measurement of this parameter is essential for the evaluation of the non-isothermal evaporation algorithm being evaluated as part of the water balance modeling.

Radionuclide transport parameters investigated include sorption as measured by the distribution coefficient ( $K_d$ ), and the hydrodynamic dispersion coefficient. The major contribution of the project to the understanding of sorption was the measurement of  $K_d$  in unsaturated soils. The  $K_d$  of various fission products was shown not to change down to water contents of 10% (5). The work with strontium-85 was reported last year (2) and revealed the effect of soil salinity on sorption. Data was presented which illustrated the "snowplow" effect.

#### Experimental data sets

Figure 3 lists the experimental data sets being constructed to be used in the comparative modeling studies. Laboratory data include the unsaturated column studies used in the  $K_d$  evaluation described above, as well as the strontium-85 data illustrating the snowplow effect. Other laboratory data include the transport of cobalt-60 with and without the organic complexant EDTA.

Field scale tracer tests have also been conducted at the BWTF. These tests provide data sets describing both large scale hydrodynamic dispersion, and the effects of EDTA on the transport of cobalt-60 in a more realistic natural environment. These tracer tests were also conducted at ambient and enhanced levels of precipitation. Tracer tests were completed in FY 82.

Water balance data has been collected at the BWTF since 1978. Data that show the changes in water content over yearly cycles, as well as the annual evaporation and drainage totals are available. Data such as that shown in Figure 4 will be used to compare water balance models (primarily evaporation algorithms) as to their validity. Figure 4 shows the steady increase in water content at the bottom of the five foot lysimeter. The increase in water content from 4 to 25% is accompanied by an increase in water potential from -0.1 bar to essentially zero.

#### WORK TO BE COMPLETED

The data collection at the BWTF has been with unvegetated lysimeters. The final data set to be collected will be of water balances under vegetated surfaces. The use of water by desert plants will be monitored to provide data on plant water uptake necessary to model transpiration at burial sites.

Existing data sets will be fully documented and summarized for use in the comparative modeling studies. This will result in a complete data

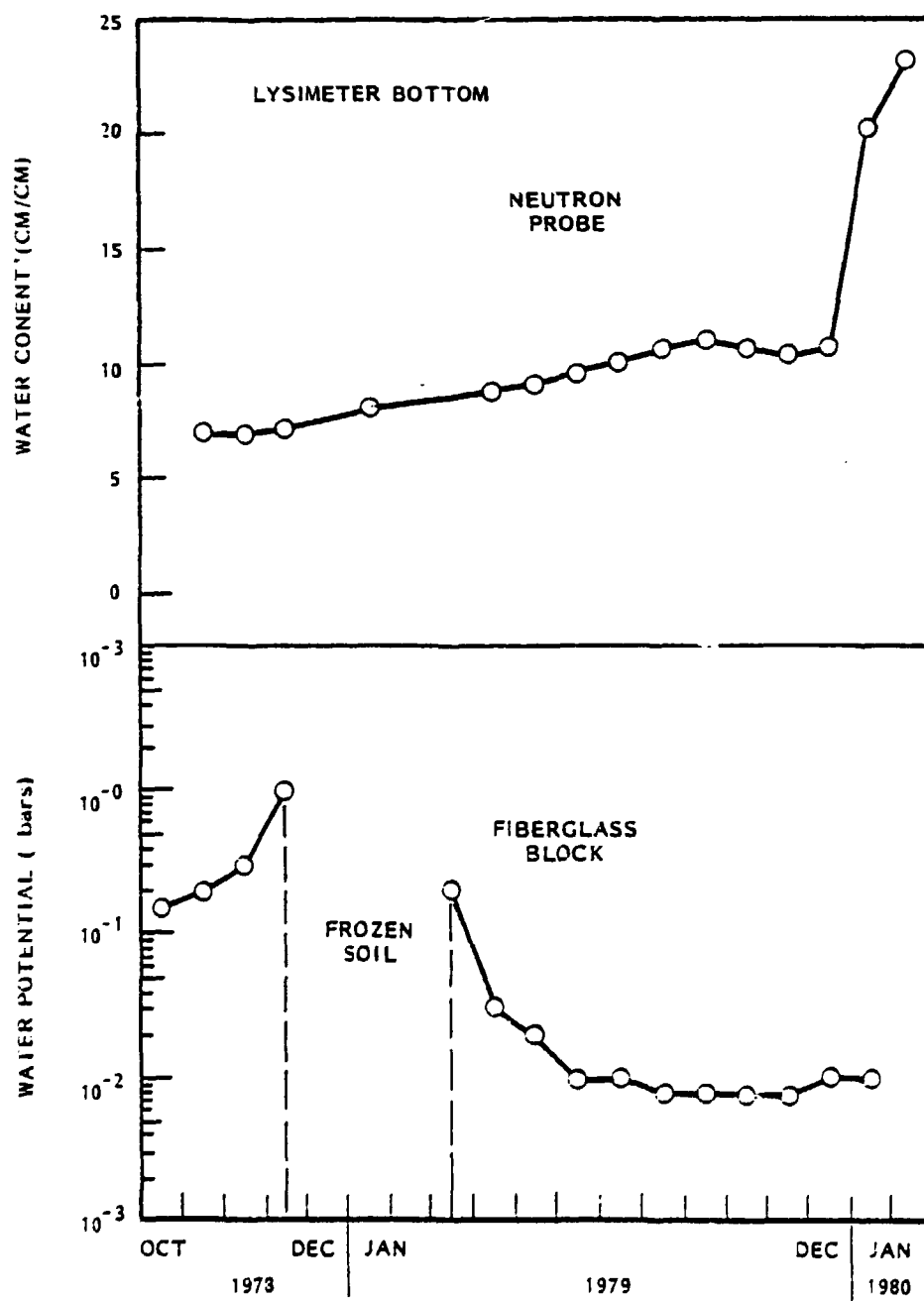


Figure 4. Water content and water potential changes with time in bottom of five foot unirrigated lysimeter.



base available for the evaluation of unsaturated zone transport models. Preliminary modeling has already been done with the strontium-85 snowplow data. This modeling compared the convective-dispersive and the mobile-immobile water models. This will be expanded to include the random walk type models (6). These models will also be used extensively to analyze the scale dependence of the dispersion process. The small scale laboratory experiments will be compared to the large scale field tests in order to develop a relationship for the dispersion coefficient that takes into account the experimental scale. If a relationship that provides adequate predictive capability cannot be found, then alternative models must be found. These include the stochastic models that explicitly account for the spatial variability of the soil hydraulic properties (7).

The model comparison dealing with profile water balances will concentrate on describing evaporation. This seems to be the major area of uncertainty remaining in the modeling of unsaturated water transport. Empirical algorithms, as well as the more theoretical non-isothermal approaches, will be compared (Figure 5). Their validity will be judged on their ability to describe data such as that found in Figure 4.

#### SUMMARY

Work on this project has progressed to the point where a critical evaluation of unsaturated zone transport modeling may be made. Transport parameters have been characterized and experimental data sets have been collected that will allow fair tests of available models. Processes such as evaporation and hydrodynamic dispersion will be modeled in various ways to show which method is the simplest yet still provides the necessary accuracy needed in low-level waste management.

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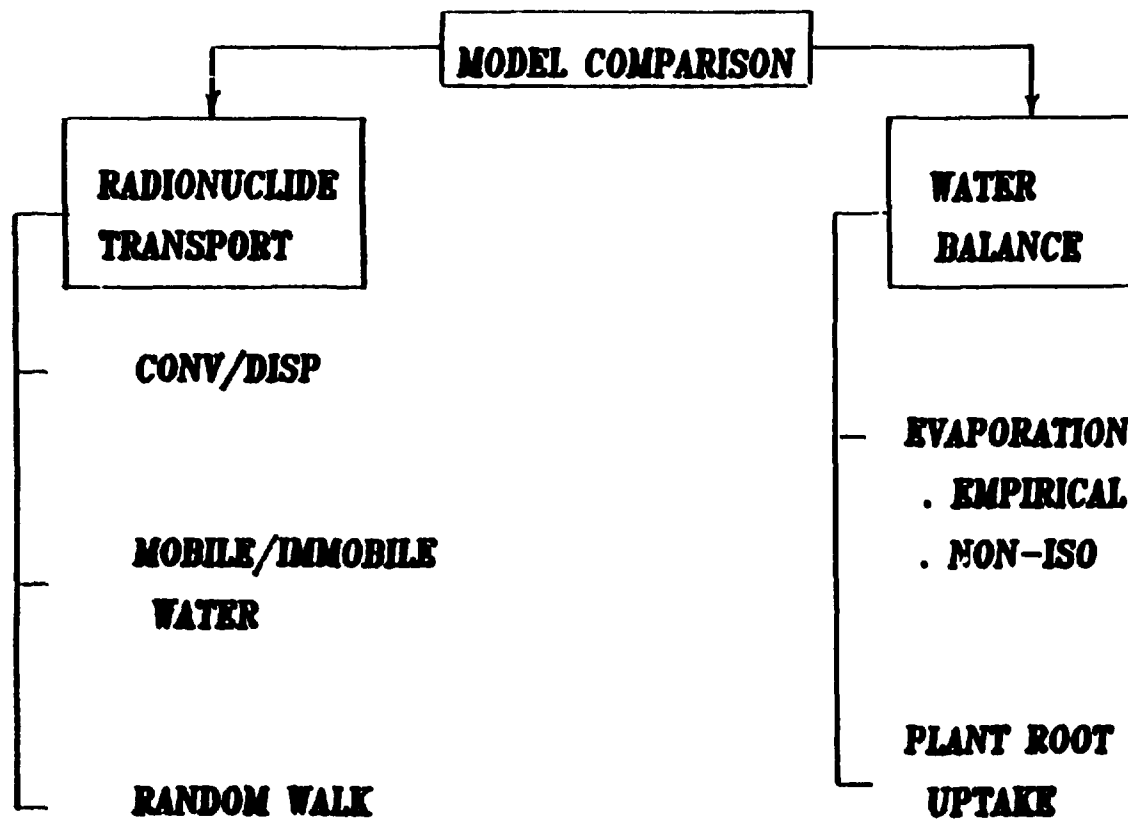


Figure 5. List of models to be compared during modeling portion of project.

Laboratory. Richland, Washington.

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SUMMARY OF THE UNSATURATED HYDROLOGIC FLOW AND  
TRANSPORT MODELING PROJECT CONDUCTED FOR NRC



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ABSTRACT

The objective of this technical assistance project sponsored by NRC was to summarize the present state-of-the-art for unsaturated flow and transport modeling. It involved two primary efforts: 1) preparing an indepth summary and evaluation of available unsaturated flow and transport models potentially useful to NRC and 2) organizing and convening a symposium on unsaturated modeling to elucidate the present strengths and weaknesses in unsaturated flow and transport analysis.

The combination of model reviews and symposium discussion was designed to consider the present state-of-the-art in unsaturated modeling as part of a continuum of the past, present, and needed future developments in partially saturated flow and transport technology. The emphasis of this effort was on providing assessment capability to evaluate low-level waste management alternatives and disposal facilities.

The two phases of this technical assistance project for NRC are essentially complete and further work under this specific study is not expected to extend beyond the current fiscal year. The model review and summary results are presented in a technical report by C. A. Oster (1) which was submitted to NRC for review and publication. The symposium proceedings have been submitted for publication and will be released this month (2). These two extensive documents are only summarized here and one pertinent transport issue considered in a cursory way as an important point of discussion of the symposium proceedings.

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SUMMARY REVIEW OF UNSATURATED FLOW  
AND TRANSPORT MODELS

The report by C. A. Oster describes the results of an extensive literature search along with telephone and mail correspondence conducted to identify computer models suitable for studies of waste facilities

located in the unsaturated zone. Fifty-five existing models were identified and briefly summarized as potentially useful. Ten of these models were selected for further examination. Each of the ten selected models are described in considerably more detail.

The report contains a statement of the ground-water flow-contaminant transport problem, a discussion of those methods used to reduce the physical problem to a computer model, and a brief discussion about the data requirements of these models. The procedure used to select the ten codes for further discussion is given, along with a list of these models.

Finally, the Appendices contain the data about the fifty-five codes examined. Specifically, Appendix D contains the detailed discussion of each of the ten selected codes. Included in each discussion are items which a potential user requires to determine whether the code is suitable for his applications. Appendix E contains brief summary information about each of the fifty-five codes. Included in the summaries are identification data, authors, pertinent references, and model type.

#### SUMMARY OF SYMPOSIUM PROCEEDINGS

The NUREG/CD-00030 document records the proceedings of the symposium on flow and transport processes in partially saturated ground-water systems, conducted at the Battelle Seattle Research Center on March 23-24, 1982. The Symposium was sponsored by the U.S. Nuclear Regulatory Commission for the purpose of assessing the state-of-the-art flow and transport modeling for use in licensing low-level nuclear waste repositories in partially saturated zones.

The first day of the Symposium centered around research in flow through partially saturated systems. Papers were presented with the opportunity for questions following each presentation. In addition, after all the talks, a formal panel discussion was held. During this discussion written questions were addressed to the day's speakers. The second day of the Symposium was devoted to solute and contaminant transport in partially saturated media in an identical format.

The papers presented and the discussion on the second day demonstrated the reevaluation and gradual changing approach occurring within the subsurface transport modeling field. The gradual transition is best considered from a historical perspective. The traditional convective or advective dispersion equation introduced in the late 1940's and early 1950's is being questioned as to how well it describes fluid transport in porous materials. Field experimental studies, as well as recent theoretical analysis, suggest an improved mathematical description of fluid transport is needed. Four of the eight symposium transport papers pointed to the need for an improved description of dispersive transport.

The central issue in the reevaluation now underway in the transport area is in the conceptualization of the dispersion process in porous media flow. To understand the issue careful consideration of two processes is required, namely diffusion and dispersion. Unfortunately, too often these two processes have not been carefully enough differentiated. Admittedly, they do combine to yield the same observable results in a porous flow system. Both result in gradually spreading or reducing of the concentration of a dye spot injected into a porous flow system. However, diffusion and dispersion are grossly different in terms of the underlying cause of the process involved.

The diffusion of a tracer or solute in the fluid in a porous material is caused or results from the molecular motion of the solute involved. Since there are more of the solute atoms or molecules in the area of higher concentration the result of diffusion always involves the very gradual spreading of the solute into an ever widening volume of the porous media. Experience indicates that such a diffusion process is rather slow and the spreading is restricted to relatively short distances. It occurs regardless of the fluid being at rest in the porous media or if the fluid is moving. Diffusion is still acting any time there is a difference in solute concentration in the fluid.

In contrast, the dispersion of a tracer or solute cannot occur if there is no fluid motion. Unlike the diffusion, the dispersion process is a result of variations or changes in the fluid velocity so dispersion does not occur in a static or nonmoving fluid since there are no velocity variations. The variations of fluid velocity in the porous material which are the cause of dispersion can be discussed both on the basis of a smaller microscopic scale of velocity variations or on the larger macroscopic scale.

On the microscopic scale of individual soil pore size the velocity variations are a result of changing pore channel areas and on different flow path lengths which cause some tracer or solutes to be carried along faster by the higher fluid velocities while other tracer molecules transported by the slower fluid velocities lag behind. Movement along more tortuous paths around soil particles, sand grains, or around small rocks in the porous material further accentuate the solute spreading by dispersion. All of these causes which introduce velocity differences result in extending and broadening a spot of tracer or solute injected into the system with an associated reduction in concentrations. Such reduction of concentration and spreading in observed solute by the dispersion processes are caused by variations or differences in the fluid velocity.

Similarly, the larger scale macroscopic dispersion is associated with changes in macroscopic fluid velocities on the larger scale. The changes or variations in fluid velocities are associated with spatial changes in hydraulic conductivity and effective porosities, i.e., they are caused by the large scale heterogeneity of natural materials. The larger scale dispersion effects are associated with the larger scale fluctuations or variations in fluid velocities. If, on the other hand, there is no velocity so that fluid is static then there can be

no dispersion. Any gradual solute spreading and dilution under static conditions can only be through diffusion. When flow is occurring, the gradual spreading and solute concentration reduction as flow progresses is the combined effect of the diffusion and dispersion processes. Usually diffusion is the much smaller effect; in fact, often it may be negligible.

The physical process of diffusion is very appropriately described by the usual Fickian-type description. Such a Fickian description mathematically defines the diffusive flux as the product of a macroscopic diffusion coefficient and a gradient of the solute concentration.

Careful differentiation between the diffusion and dispersion processes during transport show them to have fundamentally different causations. Diffusion is the result of random motion of the solute molecules in the fluid and is observed to occur whenever and wherever there are differences in solute concentrations. The fluid may be either static or in motion, it makes no difference since the cause of the diffusion process is in the activity of the individual solute molecules.

Dispersion, on the other hand, occurs only when there is movement of the transporting fluid through the porous material. In a sense the solute involvement in the purely dispersion process is only as a traceable marker that is along for the ride with the transporting fluid. The strictly dispersive-induced changes in solute plume size and associated concentration reductions are only dependent upon the irregularities in the fluid velocity field.

The central issue in the overall area of ground-water transport analysis is the appropriate way to mathematically describe the dispersion process. The traditional assumption adopted in the late 1940's and early 1950's was to describe the dispersion process as a Fickian process that was completely analogous to diffusion. Under this process as a diffusion coefficient for describing the diffusion process which no one questions and then another much larger "diffusion-like coefficient" called the "dispersivity coefficient" that is assumed to describe the dispersion process. Both coefficients are multipliers of the concentration gradient. Under such assumptions the resulting equation for describing fluid-coincident solute transport in porous materials is the classical convective or advective-dispersion equation.

Four of the eight papers on transport in the NRC-sponsored symposium were either suggesting descriptions of a different mathematical form or were describing field experiments which demonstrate the difficulty in using the traditional convective-dispersive equation in describing field transport experiments. Specifically, the distinguished G. de Marsily from France stated as conclusions to his symposium paper (de Marsily, 1982, page 314):

"For several years, field experiments had shown that the classical dispersion equation was not very satisfactory for the prediction of transport in porous media.

New theoretical developments have now given some explanations of the observed phenomena, and opened areas for research.

But it is not yet clear how to apply these new concepts, and how to measure the relevant parameters in situ.

While the ideas are progressing, contaminant transport predictions should be rather conservative, and the data collected in the field should always be analyzed with the new concepts in mind, in order to detect the existence of a "time effect", and to collect evidence (if it is the case) that asymptotic behaviour has been reached."

#### SUMMARY

A number of important considerations and questions are being raised that have very important implications to the valid modeling of subsurface contaminant transport in both a saturated and partially saturated porous system. Among these considerations and questions are the following:

- Diffusion and dispersion are two different physical process that do, however, both occur during solute transport through a porous material.
- In diffusion, the solute concentration gradient is the macroscopic causal variable or is the driving mechanism of the process.
- The causal concentration gradient and a diffusion coefficient are combined to provide the well tested and accepted Fickian mathematical description of the diffusion process.
- In contrast, in dispersion the solute concentration gradient is not the causal variable or mechanism, but is only a passive observed consequence.
- In dispersion, the irregularities and variations in the transporting fluid velocities are the causal variables or driving mechanism of the process.
- Is a mathematical representation, which replaces the causal velocity variations as the driving mechanism of dispersion with the passive consequential solute concentration gradient, the best or even a satisfactory description of the dispersion process?

There is growing evidence that the assumption of a Fickian form for describing dispersion is inappropriate and of limited utility (3) (4) (5) (6). Certainly the cautions regarding subsurface transport



analysis, as pointed out by the authors and discussants at the NRC-sponsored symposium, should be carefully considered by the modeling and user community.

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## STUDY OF UNSATURATED ZONE HYDROLOGY AT MAXEY FLATS

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## ABSTRACT

The initial objective of this research was to determine the pathway(s) of water entry into the low level, radioactive material, waste burial trenches at Maxey Flats. Electrical soil moisture sensors and mini-porous cups were implanted in the trench cap of Trench 19S and in the soil profile adjacent to the trench. An organic tracer (pentafluorobenzoic acid) was used in tracing water movement. Also, tritium present in the trench waste was employed as a tracer of water movement. It was found that the principal mode of water entry into the trench was through the trench cap and that lateral movement through the soil profile is very slow. The Kentucky fescue grass that has been grown on the site takes up significant tritium from the trenches and transports the tritium to the atmosphere. A lysimeter experiment has been set up for the purpose of investigation of management countermeasures for reducing water infiltration into the trenches.

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INTRODUCTION

The problem of serious concern at Maxey Flats is the intrusion of excessive water into the waste burial trenches and the exit of that water while carrying radionuclides (1, 2, 3). The waste burial site was closed after seepage water containing radionuclides was observed in a freshly dug trench adjacent to a filled and capped trench.

This study was initiated to determine the principal pathways of water entry into the waste burial trenches and then to seek some possible countermeasures for minimizing the entry of water into the trenches. Also, investigation of transport of tritium to the atmosphere by plant root uptake of trench water and leaf transpiration is part of this study.

## EXPERIMENTAL APPROACH

A transect across trench 19S and the adjacent area was instrumented as shown in Fig. 1. Two types of instrumentation were installed in the soil profile: (1) soil moisture cells (Soiltest Co.), and (2) mini-porous cups. The soil moisture cells were installed to allow measurements of relative soil moisture content (soil moisture tension gradients). The mini-porous cups were implanted very close (5 cm) to the soil moisture cells to allow extraction of soil water in the region of the soil moisture cells. Mini-porous cups were used instead of full-sized cups so as not to influence subsequent measurements made with the soil moisture cells. The mini-cups have a volume of approximately 2 ml and the soil solution samples are taken after the soil moisture cell readings. The soil moisture cells are then not read again until at least a two-week period has passed. The porous cups allow extraction of soil under unsaturated conditions down to about 70 centibars of soil water suction (tension); therefore, they can be used for sampling saturated or moist unsaturated soils. Electrical leads from each of the moisture sensors and fine tubing (bore--1 mm) from the mini-porous cups terminated in an instrument box mounted above the soil surface and to one side of the instrument stack or "station" located vertically in the soil profile.

After the instruments were installed and the holes refilled with the soil removed from the holes, a tracer was sprayed on a swath 3 m (10 ft) wide by 23 m (75 ft) long with the swath being centered over the instrument stations. The tracer consisted of 250 g of pentafluorobenzoic acid dissolved in 8 liters of water. The purpose of the tracer was to augment information on movement of the soil water.

Samples of Kentucky fescue grass were taken from plants grown on the caps of trenches 37, 38, and 39 and from nearby areas to study the possible transport of tritium by the fescue from the trenches to the atmosphere. The tritium present in the plant transpiration stream was determined by vacuum distillation of fresh grass samples and then liquid scintillation determination of the  $^3\text{H}$  present in water collected. The tritium present in the grass as organically bound tritium or structural tritium was determined by reacting the vacuum-dried plant material with oxygen at  $900^\circ\text{C}$  and then liquid scintillation of the water obtained from the reaction.

## RESULTS AND DISCUSSION

Soil moisture cell measurements at two selected depths at station no. 1, and with the passage of time, are given in Fig. 2. Station no. 1 was located approximately 9 m away from the edge of trench 19S and in an undisturbed area between trenches. The moisture sensor located at the 0.9 m depth indicated a rather strong seasonal cycling with an overdraft of water use by the Kentucky fescue crop during the late summer

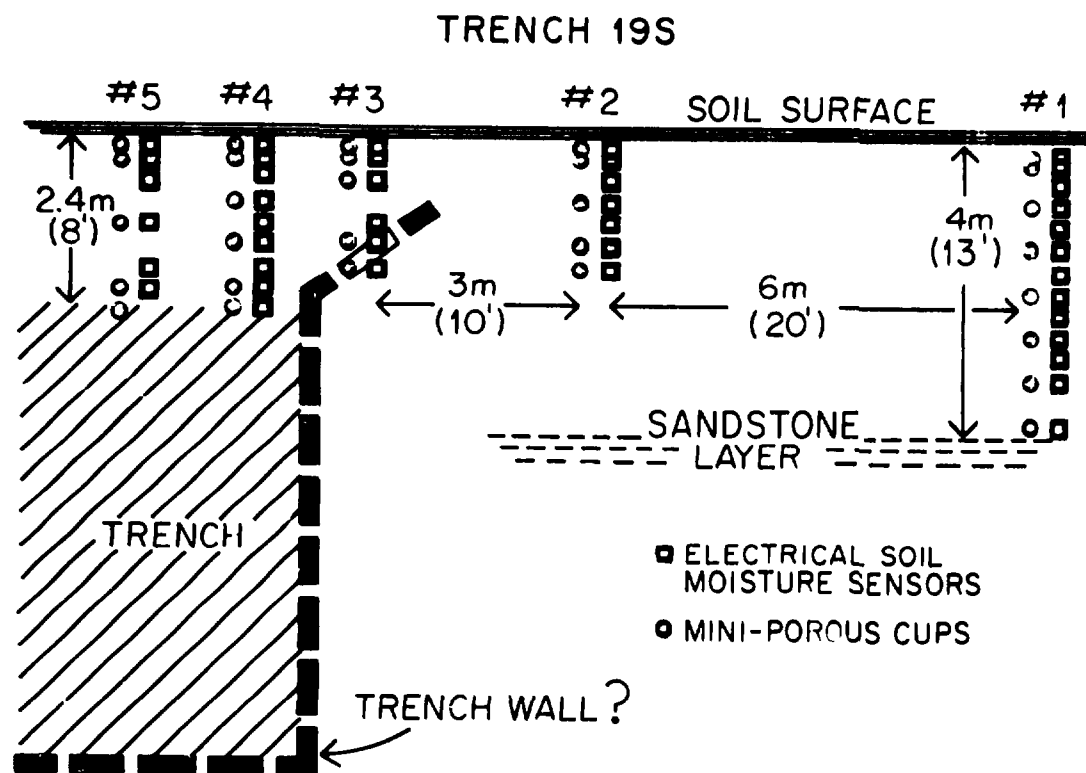


Figure 1 - Profile view of instrumental transect across trench 19S and adjacent soil. The exact location of the trench wall is not known but is known within approximately  $\pm 1$  m, or closer, from borings.

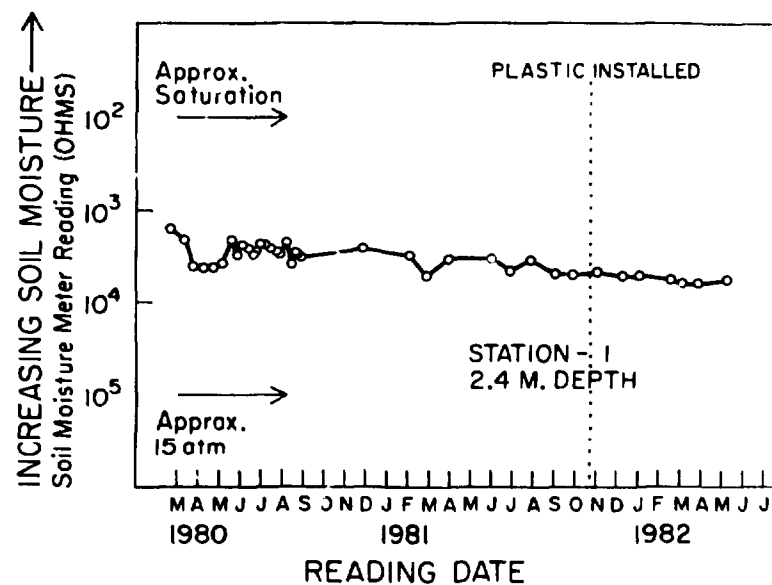
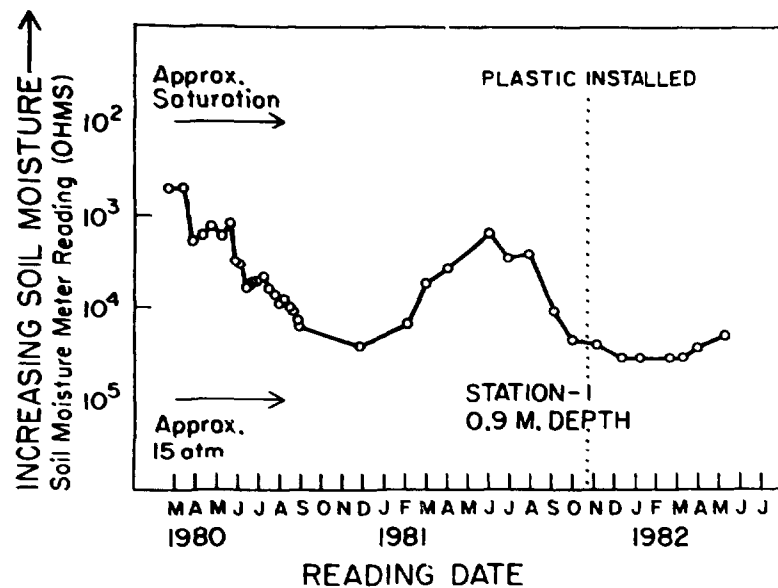


Figure 2 - Relative soil moisture content at 2 depths in soil profile, station no. 1, 9 m away from trench 19S, during the period Feb. 1980-June 1982. Strong seasonal cycling is shown at the 0.9 m depth and little seasonal cycling at the 2.4 m depth. Soil profile remained unsaturated at all times.

and fall and a replenishment of soil moisture at that depth during winter and spring by infiltration of water in excess of plant water use. At no time did the soil become saturated at that depth. At the 2.4 m depth, there was little seasonal cycling, suggesting little plant root uptake of water from that depth in the soil profile or significant rainwater percolation to that depth. During the entire period, the soil at this depth remained unsaturated.

Fig. 3 depicts relative soil moisture at four depths in the trench caps on trench 19S with the passage of time. At the 0.9 m depth, a very marked seasonal wetting and drying cycle was observed. During winter and spring, the trench cap was essentially saturated but was dried out substantially by plant extraction of water during the summer and fall periods. A similar moisture status was observed at the 1.5 m depth. At 2.4 m, little or no seasonal cycling was observed and the soil remained approximately saturated during the entire period of measurements.

From the observation given above, it is evident that no water moved from station no. 1 (located in the soil profile) to station no. 4 (located in the trench cap) as to do so, the water would have to move against a concentration gradient.

In Tables 1 and 2, the  $^3\text{H}$  concentrations in the soil solutions in stations no. 1 and no. 4 are given. The concentration of  $^3\text{H}$  in the trench cap increased markedly with increasing depth, indicating the waste in the trench below the cap was the  $^3\text{H}$  source and that  $^3\text{H}$  was migrating upward into the cap by being carried in the upward movement of water during the summer and fall periods of water depletion by the fescue crop. At station no. 1, the  $^3\text{H}$  was relatively low, and uniform with depth in the soil profile, indicating the evaporator was the source of the  $^3\text{H}$  at this location, and not the waste in the trench located 9 m away. This condition would evidence no or little movement of water from the trench to the location of station no. 1, 9 m outside the trench.

Table 3 gives the distribution of pentafluorobenzoic acid in the trench cap in January 1980. The tracer had been sprayed on the soil surface in October 1979 and as this tracer is readily water soluble and not absorbed by soil, it will tend to trace movement of "new" (rain) water into the trench cap. The data indicate that the rain water infiltrated and percolated readily through the trench cap.

More evidence of the lack of substantial lateral water movement in the top few meters of the soil profile can be seen by comparing Fig. 3 with Fig. 4. In Fig. 4 the relative soil moisture contents at three depths (3.3 m, 5.1 m, and 6.5 m) are given for the period February 1980-June 1982. From Fig. 3 it is seen that the cap was saturated for the entire period at the 2.4 m depth, while approximately 3 m away from the trench the soil profile remained quite dry at the 3.3 m depth. At the 5.1 m depth in the soil profile and 3 m away from the trench, water was apparently very slowly moving to this area from the trench. At

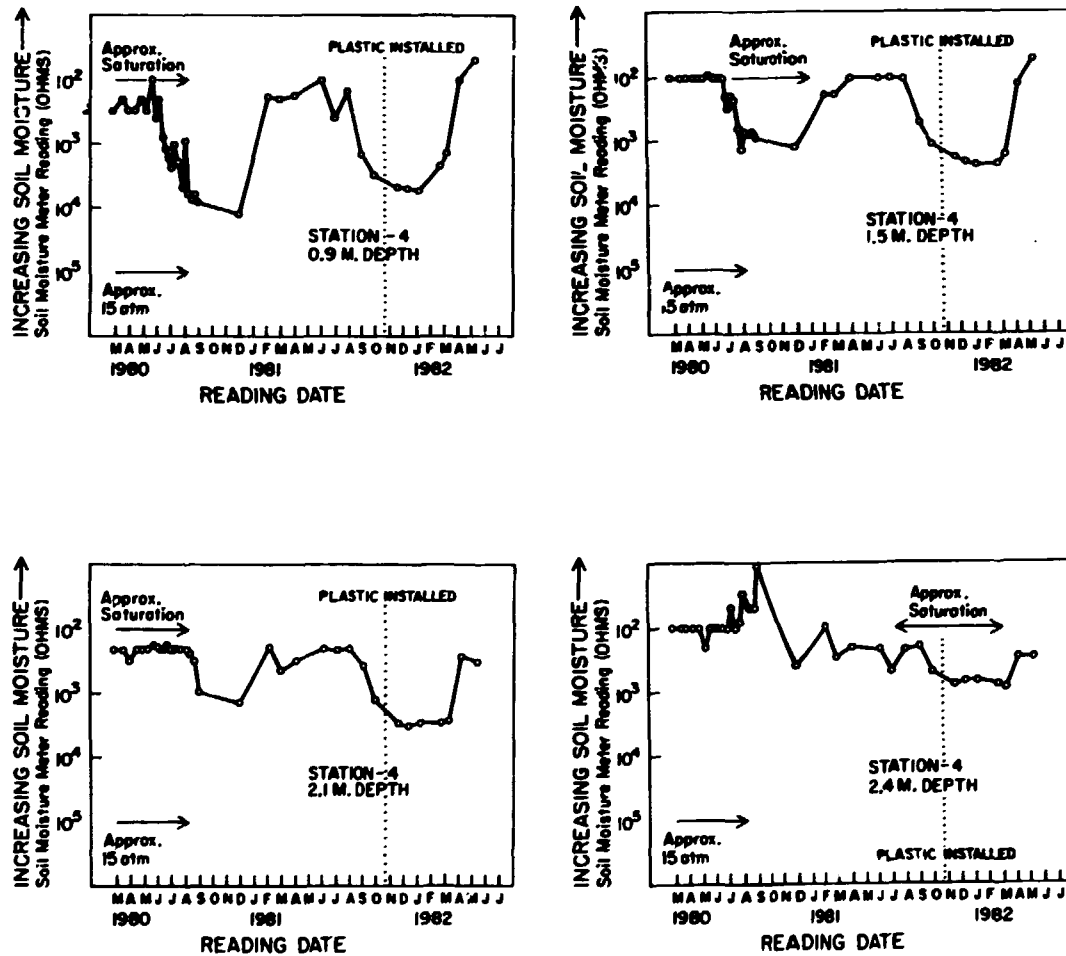


Figure 3 - Relative soil moisture contents at 4 depths in a trench cap of trench 19S, during the period Feb. 1980-June 1982. A marked seasonal cycling was observed at the 0.9 m depth and the soil was essentially saturated in the winter and spring at that depth. It was substantially dried out during the summer and fall by overdraft water extraction by the Kentucky fescue. A similar moisture pattern was observed at the 1.5 m depth. At 2.4 m into the trench cap, little or no seasonal cycling was observed and the soil remained approximately saturated during the entire period of measurements.

Table 2

$^3\text{H}$  concentration in soil solution of the trench cap, station no. 4, trench 19S. The tritium concentration in the trench cap is much higher than that found in the soil profile 9 m away from the trench (Table 1); therefore the primary source of the tritium is the trench water, not the evaporator.

| Soil depth, m | $^3\text{H}$       |
|---------------|--------------------|
|               | pCi/l              |
| 0.15          | $0.28 \times 10^6$ |
| 0.30          | 0.65 "             |
| 1.50          | 1.07 "             |
| 2.10          | 2.07 "             |
| 2.40          | 7.93 "             |

Table 1

$^3\text{H}$  concentration in soil solution, station no. 1, adjacent to trench 19S. The tritium concentration is essentially constant with depth in the soil profile and much lower than that in the trench cap.

| Soil depth, m | $^3\text{H}$       |
|---------------|--------------------|
|               | pCi/l              |
| 0.15          | $0.09 \times 10^6$ |
| 0.30          | 0.08 "             |
| 0.60          | 0.09 "             |
| 0.90          | 0.08 "             |
| 1.50          | 0.11 "             |
| 2.10          | 0.11 "             |
| 3.90          | 0.08 "             |



Table 3

Pentafluorobenzoic acid tracer concentration in soil solution of trench cap. The tracer was applied to the trench cap surface and the data indicate that rain water infiltrated and percolated readily through the trench cap.

| Soil<br>depth,<br>m | Pentafluoro-<br>benzoic acid |
|---------------------|------------------------------|
|                     | ppm                          |
| 0.15                | 14                           |
| 0.30                | 17                           |
| 0.90                | 67                           |
| 1.20                | 30                           |
| 2.10                | 161                          |
| 2.40                | 175                          |

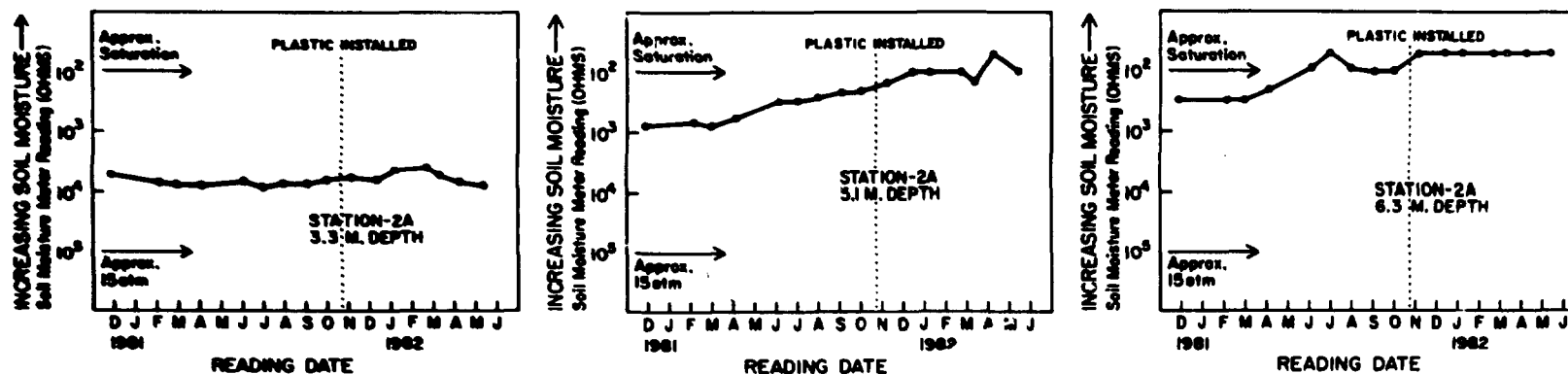


Figure 4 - Relative soil moisture contents at 3 depths in soil profile, approximately 3 m away from trench 19S during the period Feb. 1980-June 1982. It is of interest to note that the soil profile remained quite dry at the 3.3 m depth, whereas the trench was constantly saturated at the 2.1 m depth, showing little lateral water movement at this depth. At the 5.1 m depth water was apparently migrating outward from the trench and the profile was essentially saturated at the 6.3 m depth.

the 6.3 m depth the soil was essentially saturated the entire period although the moisture content increased slightly.

Prior to the application of plastic to much of the Maxey Flats disposal site in the fall of 1981, Kentucky fescue was grown on the entire site. It is of interest to determine the magnitude of the transport of  $^3\text{H}$  from the trenches to the atmosphere by plant root uptake and transpiration from the leaves.

In the winter of 1982, grass samples were taken at various locations on the site, both over trenches without plastic and in undisturbed areas. It was found that it was necessary to get as removed as possible from the evaporator depositions of tritium to minimize confusion as to the source of the tritium in the plant transpiration stream, i.e. the evaporator or the trench water as the tritium source. Data in Table 4 gives the concentration of  $^3\text{H}$  in the leaf transpiration stream from fescue grown on the caps of trenches 37, 38, 39, and nearby areas. The  $^3\text{H}$  in the transpiration stream was about seven times higher in the plants grown on the trench caps than that of the controls, but it was only about 1/1000 the concentration of the trench water. The concentration of the  $^3\text{H}$  present as structural hydrogen in the organic material was about twice as high as it would be if it were in equilibrium with the transpiration stream. At equilibrium, the organic  $^3\text{H}$  should be 60% of the  $^3\text{H}_2\text{O}$  due to the  $^3\text{H}$  concentration in  $^3\text{H}_2\text{O}$  and  $\text{C}^3\text{H}_2\text{O}$ . The plant matter is mostly cellulose; therefore the  $\text{CH}_2\text{O}/\text{H}_2\text{O}$  ratio is appropriate. The reason for the disequilibrium is probably that the plants were sampled in February 1982 during the wet, cold season and the  $\text{H}_2\text{O}$  tritium present in the plant would reflect rain-water dilution of the  $^3\text{H}$  present in the soil. The organic tritium would have been "built in" during the growth period during the warmer, drier, growing period preceding the winter.

#### WORK UNDERWAY

The instrument stations installed in the trench cap of 19S will be used to monitor changes brought about by the application of the plastic in the fall of 1981. So far, data shown in Figs. 2 and 4 do not indicate leakage past the plastic whereas the data of Fig. 3 would indicate leakage. The mini-porous cups will be sampled to investigate any redistribution of  $^3\text{H}$  above and away from the trench.

Lysimeters have been installed to enable prediction of site functions as affected by various types of site management, and possible procedures to be followed in site closing. Bulk density measurements were made on 19S and were found to be about  $1.8 \text{ g/cm}^3$ ; therefore the lysimeters were packed to that soil density. Soil fertility trials have been run at Berkeley and the soil has been found to be extremely deficient in phosphorus and nitrogen in addition to needing a lime application. The soil was not deficient in potassium or sulfur or any micronutrient. This information will be used in maximizing plant

Table 4

Tritium concentration in leaves of Kentucky fescue grass grown on trench caps and nearby field (controls) at the southeast corner of the Maxey Flats site.

| Trench number | $^3\text{H}$ concentration in plant sap <sup>†</sup> ( $^3\text{H}_2\text{O}$ ) | $^3\text{H}$ concentration in trench water ( $^3\text{H}_2\text{O}$ ) | $^3\text{H}$ concentration in leaf organic material <sup>‡</sup> ( $\text{C}^3\text{H}_2\text{O}$ ) |
|---------------|---------------------------------------------------------------------------------|-----------------------------------------------------------------------|-----------------------------------------------------------------------------------------------------|
|               | — pCi/l —                                                                       |                                                                       | — pCi/kg —                                                                                          |
| 37            | $6.9 \times 10^4$                                                               | $3.7 \times 10^7$                                                     | $9.7 \times 10^4$                                                                                   |
| 38            | 3.5 "                                                                           | 3.3 "                                                                 | 4.3 "                                                                                               |
| 39            | 10.5 "                                                                          | 10.1 "                                                                | 14.0 "                                                                                              |
| Control       | 1.1 "                                                                           | ---                                                                   | 2.0 "                                                                                               |
| Control       | 1.0 "                                                                           | ---                                                                   | 2.0 "                                                                                               |
| Control       | 0.9 "                                                                           | ---                                                                   | 1.0 "                                                                                               |

<sup>†</sup> Determined by vacuum distillation of fresh leaf material then liquid scintillation of the  $^3\text{H}_2\text{O}$  distilled off.

<sup>‡</sup> Determined by reacting vacuum-dried leaf material with  $\text{O}_2$  at  $900^\circ \text{C}$ , then liquid scintillation of the resulting  $^3\text{H}_2\text{O}$ .

growth in the lysimeters with the goal of maximizing plant transpiration and minimizing water percolation to the water table. The lysimeters will allow complete water balance studies (Fig. 5); therefore the effects of crop types and management of the crops on increasing or decreasing water entry into waste burial trenches can be evaluated. Lysimeter studies using trench water will be useful in quantifying radionuclide movement and especially  $^3\text{H}$  release to the atmosphere by evaporation of  $^3\text{H}_2\text{O}$  and plant transpiration of  $^3\text{H}_2\text{O}$ . The lysimeter  $^3\text{H}$  studies will be run in conjunction with detailed studies on the trench 38 area described earlier in this report.

#### ACKNOWLEDGEMENT

Research supported by the U.S. Nuclear Regulatory Commission, FIN No. B3042. The author expresses thanks to Dr. G. Thompson for the pentafluorobenzoic acid analyses and to Mr. J. Razor for furnishing the tritium content of trench water in trenches 37, 38, and 39.

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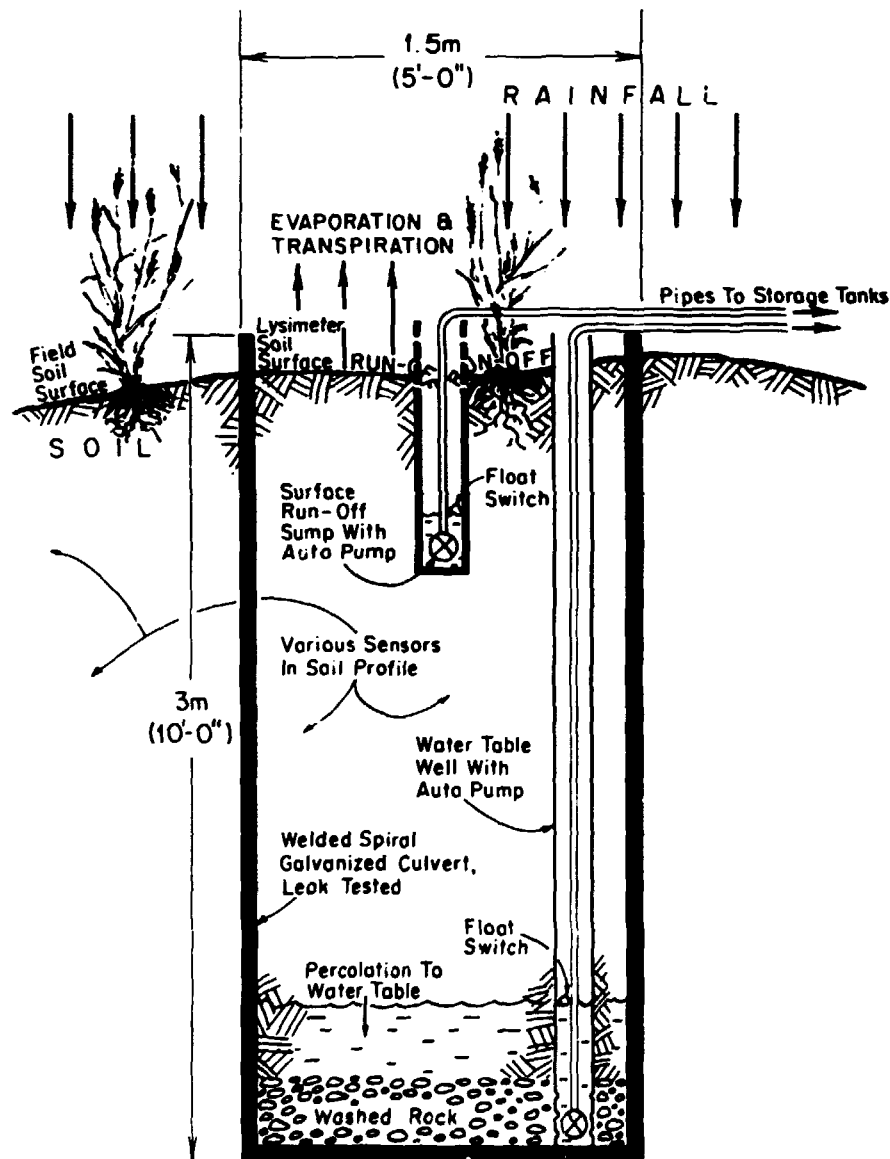


Figure 5 - Cross section of an installed lysimeter. System gives complete accountability of all rain water falling on soil surface. Amount of water striking soil surface is known from rainfall measurements. Over a sufficiently long time period, the water striking the surface has three fates: I. Surface run-off. Measured by collection in surface run-off sump and pumping to collection tank. II. Deep percolation to the water table. Measured by maintaining constant water table level by pumping to storage tank. III. Surface evaporation and plant transpiration. Surface evaporation and plant transpiration = rainfall - (surface run-off + deep percolation). Lysimeters to be used to evaluate effects of crop type, management, and also radionuclide migration studies, especially transport of  $^3\text{H}$  from water table to the atmosphere by plant uptake and transpiration.

WELL LOGGING INSTRUMENTATION FOR THE IN SITU  
DETERMINATION OF  $^{90}\text{Sr}$ , TRITIUM, AND TRANSURANICS

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ABSTRACT

The development of four instrumental methods for in situ well logging determination of  $^{90}\text{Sr}$ , tritium, and transuranic isotopes is described. Costs and detection limits are given and compared to laboratory costs and release limits. A detailed description of a technique for the determination of  $^{90}\text{Sr}$  concentrations by the measurement of bremsstrahlung radiation is given.

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INTRODUCTION

The objective of this research program is to develop appropriate well logging equipment capable of determining  $^{90}\text{Sr}$ , tritium, and transuranics at sensitivity levels that will provide useful information for the proper operation of a shallow land burial site. This DOE National Low Level Waste Management Program effort will provide technology for monitoring the environmental aspects of low level waste disposal, the effectiveness of improved shallow land burial methods, and the need for and success of corrective measures at existing low level waste disposal sites. In addition, the developed technology is also applicable to the measurement and monitoring requirements of low level waste Greater Confinement scenarios.

The need for rapid, in situ determination of radionuclide concentrations at low level waste disposal sites cannot be overstated. In order to measure the mobility and migration of radionuclides from and within a disposal site, routine monitoring of their concentrations at selected locations is imperative. The advantages of relatively inexpensive, real-time, in situ analyses over the costly laboratory analyses of grab samples, which may take months to obtain, are self-evident. In addition, in situ analyses can be performed without altering the matrix, thus ensuring the relative significance of successive results. Since the earliest application of high resolution germanium diode gamma-

ray spectrometry to the quantitative determination of radiocontaminants in deep sediments(1), the in situ measurement of gamma-ray emitting fission products has been relatively straightforward. However, the most important radionuclides, from the standpoint of potential health hazards, mobility, and long half-life, are  $^3\text{H}$ ,  $^{14}\text{C}$ ,  $^{90}\text{Sr}$ ,  $^{99}\text{Tc}$ ,  $^{129}\text{I}$ ,  $^{237}\text{Np}$ , and the transuranics. This communication will describe four separate techniques which have been developed expressly for the in situ determination of  $^3\text{H}$ ,  $^{90}\text{Sr}$ ,  $^{237}\text{Np}$ , and the transuranics.

### IN SITU DETERMINATION OF $^{90}\text{Sr}$

Several potential methods for the in situ determination of  $^{90}\text{Sr}$  have been investigated during the course of this work. The earliest experiment considered the measurement of the bremsstrahlung radiation using high resolution germanium diode gamma-ray spectrometry and demonstrated the methodology to be used(2). Some simple experiments indicated that the technique might be capable of making in situ  $^{90}\text{Sr}$  determinations at a sensitivity level on the order of 20 pCi/g in normal well logging timeframes. Subsequently, the application of Phoswich detector beta particle spectroscopy to the measurement of  $^{90}\text{Sr}$  was investigated, and a detection limit of 4 pCi/cm<sup>2</sup> was reported for surface analyses(3). However, the technology lacked the versatility of operation within a well casing, the most frequent monitoring situation expected in a low level waste disposal site. The potential for determination of  $^{90}\text{Sr}$  concentrations by in situ neutron activation analysis was also investigated in a series of laboratory experiments(4). As soon as it was demonstrated that the thermal neutron capture cross section of  $^{90}\text{Sr}$  was more than a factor of 50 lower than the previously accepted literature value, the calculations of potential sensitivity for the technique rendered it no longer competitive with the bremsstrahlung detection technique, and the experiments were discontinued. Consequently, it was decided to pursue the development of the bremsstrahlung measurement technology and to reduce the theory to a practical science. The following discussion describes the results.

Since the bremsstrahlung radiation spectra generated by  $^{90}\text{Sr}$  represents a continuum of photon energies with no distinguishing monoenergetic photopeaks, such as observed from most other fission product radionuclides, the relationship between photon intensity and  $^{90}\text{Sr}$  concentration is somewhat harder to determine. Furthermore, many other fission products likely to be present in  $^{90}\text{Sr}$  contaminated environments also emit energetic beta particles which contribute to the total bremsstrahlung spectrum and which must be accounted for prior to assignment of bremsstrahlung radiation to  $^{90}\text{Sr}$ . Hence, it was decided that the most accurate approach to data reduction would be to "strip out" each identifiable component from an unknown spectrum until only  $^{90}\text{Sr}$  bremsstrahlung radiation remained. This method assumes the compilation of a library of photon spectra from each pure radionuclide, for each detector to be used, for each monitoring well diameter and casing material, and for each matrix material to be investigated. It further



assumes that  $^{90}\text{Sr}$  is the only pure beta emitter present with sufficiently energetic beta particles to contribute to the residual bremsstrahlung spectrum at a significant level. This latter assumption is valid in virtually all practical situations. The former assumption, though valid, would require a tremendous amount of baseline data acquisition, all of which may not be necessary in practical situations.

Two different approaches were taken toward assemblage of a library of pure radionuclide reference spectra. First, a synthetic approach using a one-dimensional transport code computer simulation of anticipated spectral shapes with the matrix material, well diameter, casing material, diode response function, etc. as variables was undertaken in the hope of generating desired reference spectra for any circumstance. Second, an empirical approach to the most common field situation was chosen as a demonstration of the true capability of the technique and as a calibration point for the computer simulation method. If the computer simulation method can generate the same spectral shape as those actually measured for a series of radionuclides, then the credibility of the computer technique for extension to other detectors, wells, matrices, etc. would be greatly enhanced. At this time the computer simulation technique only required the inclusion of the detector response function before spectral shapes can be compared to the empirical data. The rest of this discussion will deal with the procedures and results of the field measurement technique, including calibration and data reduction.

Since time and funds only allowed for one practical example to be demonstrated, the most likely situation was chosen. The detector to be used was our most sensitive and versatile well logging high resolution germanium diode gamma-ray spectrometer. The matrix material was chosen to be typical Hanford sediments. The most common well encountered on the Hanford site, a 6-inch schedule 40 steel casing, was chosen for the demonstration.

Since it is virtually impossible to construct a calibration facility representing an infinite volume of homogeneously contaminated undisturbed sediments, and since it is impractical to artificially generate large volumes of contaminated sediments in order to characterize those already inadvertently contaminated zones, a different approach to the development of a calibration facility was taken. A cursory survey of the Hanford site yielded a suitably located pristine area which had never been tilled or disturbed in any other way by man. This area was designated the Natural Activities Calibration Facility and was appropriately marked to prevent inadvertent intrusion. The surface vegetation and soil were sampled using a statistically acceptable geometrical pattern, and the samples were carefully analyzed in the laboratory for all measurable radioactivities. The only radionuclides present were primordial  $^{40}\text{K}$ , uranium and thorium and their progeny, and a trace of  $^{137}\text{Cs}$  attributed to fallout. The concentrations of the primordial activities indicated the site was spatially homogeneous to an accuracy of  $\pm 8.3\%$ . A six-inch schedule 40 steel well casing was

inserted near the center of the site by means of vacuum coring, and the core material was collected in 6-inch intervals. Samples of each 6-inch layer were also analyzed in the laboratory, and the primordial radionuclide concentrations were consistent within  $\pm 5.7\%$ . The known concentrations of homogeneously distributed radionuclides were used to generate an empirical efficiency curve as a function of photon energy for our well logging system. This efficiency curve is shown in Figure 1.

The next step was to create a reference library of pure radionuclide spectra, including man-made activities, without contaminating the site. Sealed sources of known intensity were fabricated for each radionuclide of interest including the primordial activities  $^{226}\text{Ra}$  and  $^{228}\text{Th}$ . The thorium source was inserted into the sediments at the NACF by means of a thin aluminum tube driven into the ground, and the net spectrum due to the standard source was measured. The source tube was then repositioned, and the procedure was repeated until the net spectrum due to the thorium standard had precisely the same shape as the primordial  $^{228}\text{Th}$  component of the background. The comparison of spectral shape was made on the basis of relative intensity of all photopeaks in the spectrum. With the source tube in the same position, the radium standard also demonstrated the primordial spectral shape. Hence, this point source location was used to generate "replicate" library reference spectra of man-made radioactivities which represent the expected response to a uniformly contaminated zone of undisturbed sediments. Note that these secondary calibration spectra are not necessary, nor used, to determine the concentration of gamma-ray emitting radioisotopes from a well logging measurement since these calibrations are directly available from the energy-efficiency curve determined on the basis of the primordial activity concentrations. These library spectra are necessary, however, in order to "strip" an unknown field spectrum down to the  $^{90}\text{Sr}$  bremsstrahlung component. A standard  $^{90}\text{Sr}$  bremsstrahlung spectrum was also obtained from the point source location in order to compare the intensity and hence determine concentration of an unknown.

Although a library of reference spectra was now available, it was far from a simple task to strip out the various components and determine the residual  $^{90}\text{Sr}$  bremsstrahlung activity. First, the background had to be stripped from each reference spectra so that all data represented photo events from a pure radioisotopic contamination. This led to the first data handling problem. If the background is correctly subtracted from a spectrum, there should be no net activity at any energy above the highest energy photopeak produced by the radioisotope -- but, there should be a statistical fluctuation above and below zero counts. Our first modification to the computer data handling software allows the entry of negative events in a gamma-ray spectrum, a maneuver not normally encountered during gamma-ray spectral accumulation. However, if the computer were allowed to truncate all negative numbers to zero, then future subtraction of reference spectra would result in an over-subtraction from an unknown spectrum. The next data handling modification was to realign a field spectrum as a function of photon energy

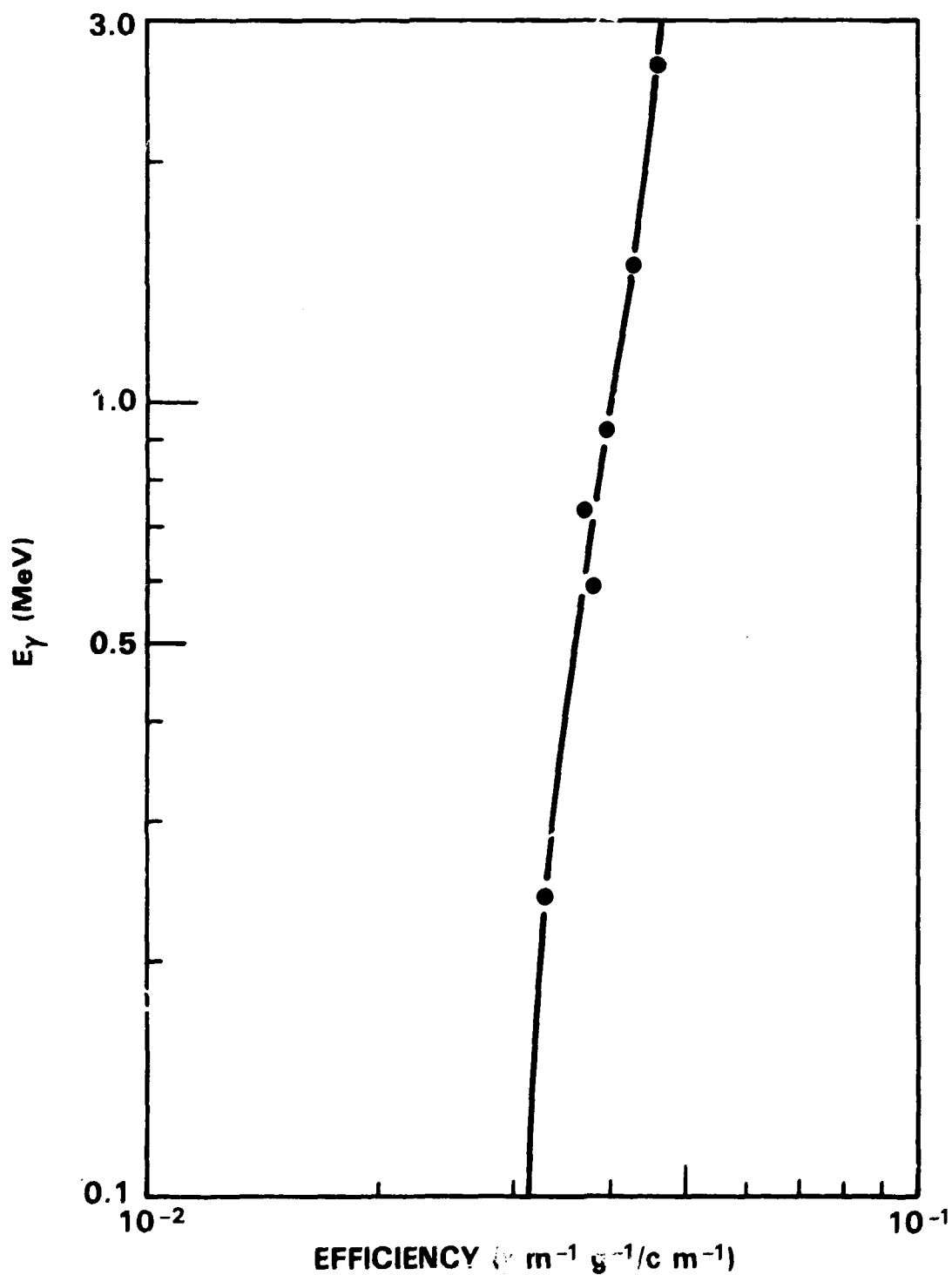


Figure 1. Efficiency of Well Logging Germanium Diode in a 6-inch Schedule 40 Steel Well Casing as a Function of Photon Energy for Contaminated Hanford Sediments.

to allow for incorrect calibration in the field or for zero or gain drift during the course of data acquisition. The difficulty in establishing and maintaining gain stability in a field environment which may involve extreme heat or cold and portable generator electric power is a nontrivial problem. Simplistically, the gain shift modification divides the field spectrum into individual counts and redistributes them into new locations based on the relative slope of a calibration line drawn through two of the prominent peaks in the field spectrum and the true calibration line drawn through the precise energy of the same photopeaks. The total number of counts is not changed in this process, but the spectrum is redrawn with all photopeaks appearing at precisely the correct energy. This data manipulation technique does not correct for loss of resolution due to field effects or gain shifts during the count, but this minor drawback has proven to be only a cosmetic effect and does not affect the data reduction accuracy.

The complete procedure for deconvoluting an unknown field spectrum starts with obtaining a background spectrum which can ordinarily be done in an uncontaminated region of the same well or from a similar well in an uncontaminated zone. After gain adjustment, the background spectrum is multiplied by the ratio of live counting times and subtracted from the unknown spectrum leaving only man-made radioisotope contributions. Each radioisotope contributing to this spectra is identified on the basis of its characteristic full energy photopeaks, and the corresponding reference spectra are called from the computer library, normalized on the basis of photopeak intensity, and stripped out of the unknown spectrum. The full energy photopeak intensities in the unknown spectra are, of course, also used to determine the concentration of each radioisotope in the matrix. The residual unknown spectrum, after stripping, represents only  $^{90}\text{Sr}$  bremsstrahlung radiation and is compared to the referenced  $^{90}\text{Sr}$  spectrum on the basis of intensity to determine concentration. Four different energy regions are normally compared in order to ensure that the residual spectrum has the correct  $^{90}\text{Sr}$  bremsstrahlung shape, and that each energy region yields the same value for the  $^{90}\text{Sr}$  concentration. In the event that no  $^{90}\text{Sr}$  is present in the matrix, the residual spectrum should appear as a straight line with only a statistical fluctuation of data above and below zero. Finally, a smoothing function has been incorporated in the software which can be applied to any spectrum for cosmetic purposes. This smoothing function neither creates nor destroys counts, but merely re-assigns neighboring channel intensities in order to remove the sharp spikes engendered by many subtractions and subtraction of peaks of the same intensity but different resolution. The only purpose of this smoothing function is to render the final product more palatable to the untrained eye which is unaccustomed to seeing negative numbers in gamma-ray spectra.

A pictorial example of the procedure is demonstrated in the following simple two-component field spectrum. Figure 2 shows the actual spectrum as acquired. A major  $^{137}\text{Cs}$  component is apparent to even the most casual observer. Just as apparent to the experienced observer is

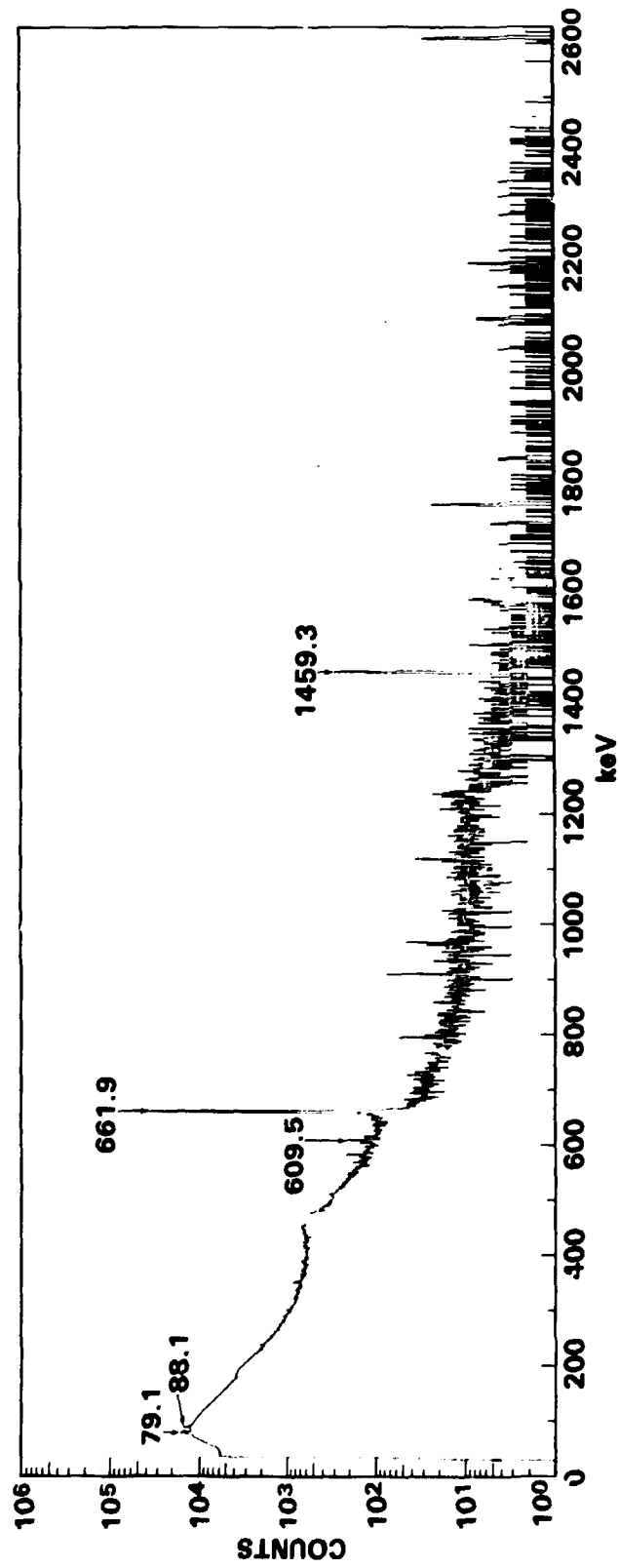


Figure 2. Gamma-ray Spectrum Obtained from a Region Contaminated with  $^{90}\text{Sr}$  and  $^{137}\text{Cs}$ .

a major  $^{90}\text{Sr}$  component, which will be demonstrated in the following sequence. First, a background spectrum obtained in a similar, but uncontaminated, environment is subtracted from the unknown spectrum after both have been "gain shifted" to the proper energies. The net spectrum is shown in Figure 3, and the  $^{137}\text{Cs}$  concentration is deduced from the intensity of the 661.649 keV photopeak. Some minor variations in resolution and gain between the unknown and background spectra are apparent at the major primordial photopeak areas. Figure 4 shows a smoothed version of Figure 3. The smoothed version is not used for continuing data reduction -- it is only a demonstration of an intermediate step. Also, since no significant data exists above 1600 keV, that portion of the spectrum will be omitted in the following figures. The net  $^{137}\text{Cs}$  reference spectrum is shown in Figure 5, with a smoothed version shown in Figure 6. This reference spectrum is normalized to the same intensity as the unknown spectrum on the basis of the 661.649 keV photopeak and is subtracted out leaving only  $^{90}\text{Sr}$  bremsstrahlung radiation as shown in Figures 7 and 8. The net  $^{90}\text{Sr}$  reference library spectrum is shown in Figures 9 and 10. Comparison of the bremsstrahlung intensity in the unknown and reference spectra yields the  $^{90}\text{Sr}$  concentration directly.

A field test of the technique was executed by logging a well in sediments known to be contaminated with  $^{90}\text{Sr}$  and  $^{137}\text{Cs}$ . Core bucket drilling techniques were used to install the well, and laboratory analyses of the core samples were used to determine the  $^{90}\text{Sr}$  and  $^{137}\text{Cs}$  concentrations in the sediments. The results from the in situ logging measurements agreed with the laboratory analyses to within  $\pm 4.3\%$ .

The detection limit for  $^{90}\text{Sr}$  in normal logging operations with this bremsstrahlung measurement technique is on the order of 10 pCi/g. This is more than adequate sensitivity since the release limit for contaminated sediments is 400 pCi/g. In routine operation, the well logging technique costs about \$15 per analysis, and the quantitative results can be available within minutes. Laboratory analyses of core samples cost more than \$500 each with typical turn-around times between one and two months. Finally, the in situ analysis provides an average  $^{90}\text{Sr}$  concentration over a sample volume on the order of 140 liters, and the core sample typically only represents a few cubic centimeters which may not necessarily represent the average, particularly in coarse or unconsolidated sediments.

The Natural Activities Calibration Facility is being maintained in order that other field assay and well logging instrumentation may be simply and inexpensively calibrated. Several surface assay detectors have already been calibrated at the site, and the same technique of data handling and reduction can be used to measure  $^{90}\text{Sr}$  concentrations on and within surfaces as well as to determine the depth of  $^{90}\text{Sr}$  activities which have been covered over with uncontaminated materials.

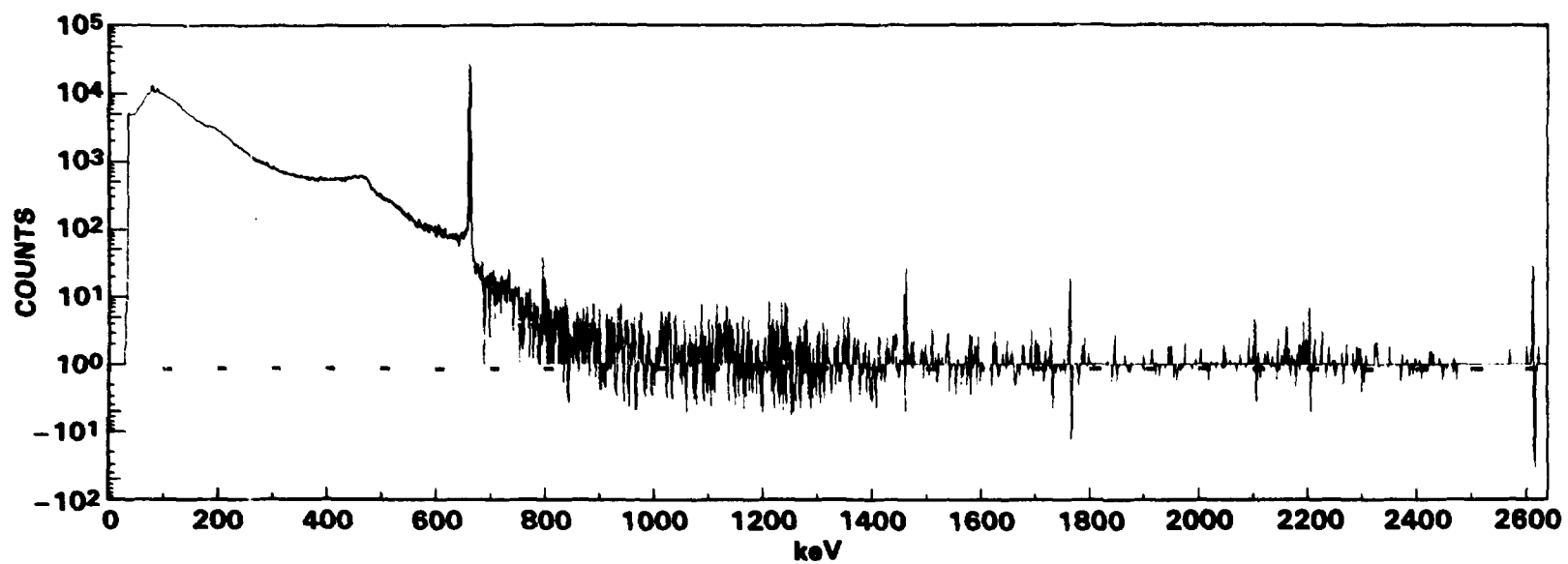


Figure 3. Gamma-ray Spectrum from Figure 2 Gain Normalized and with the Background Subtracted.

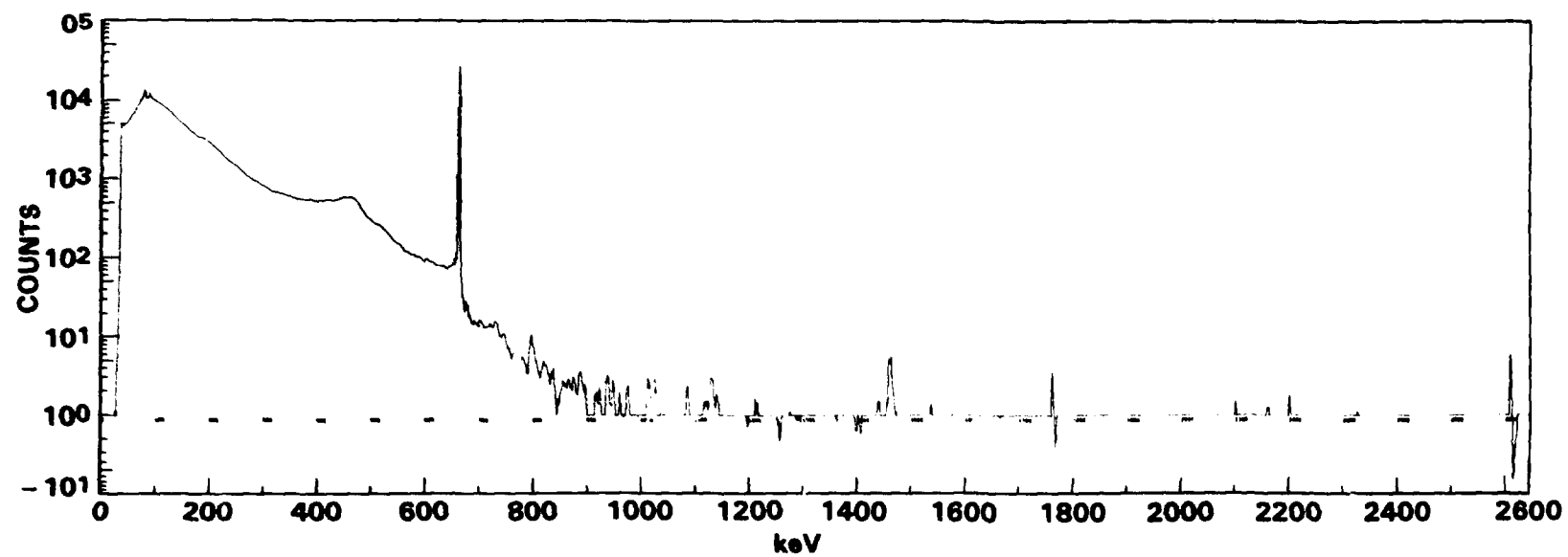


Figure 4. Smoothed Version of Figure 3.



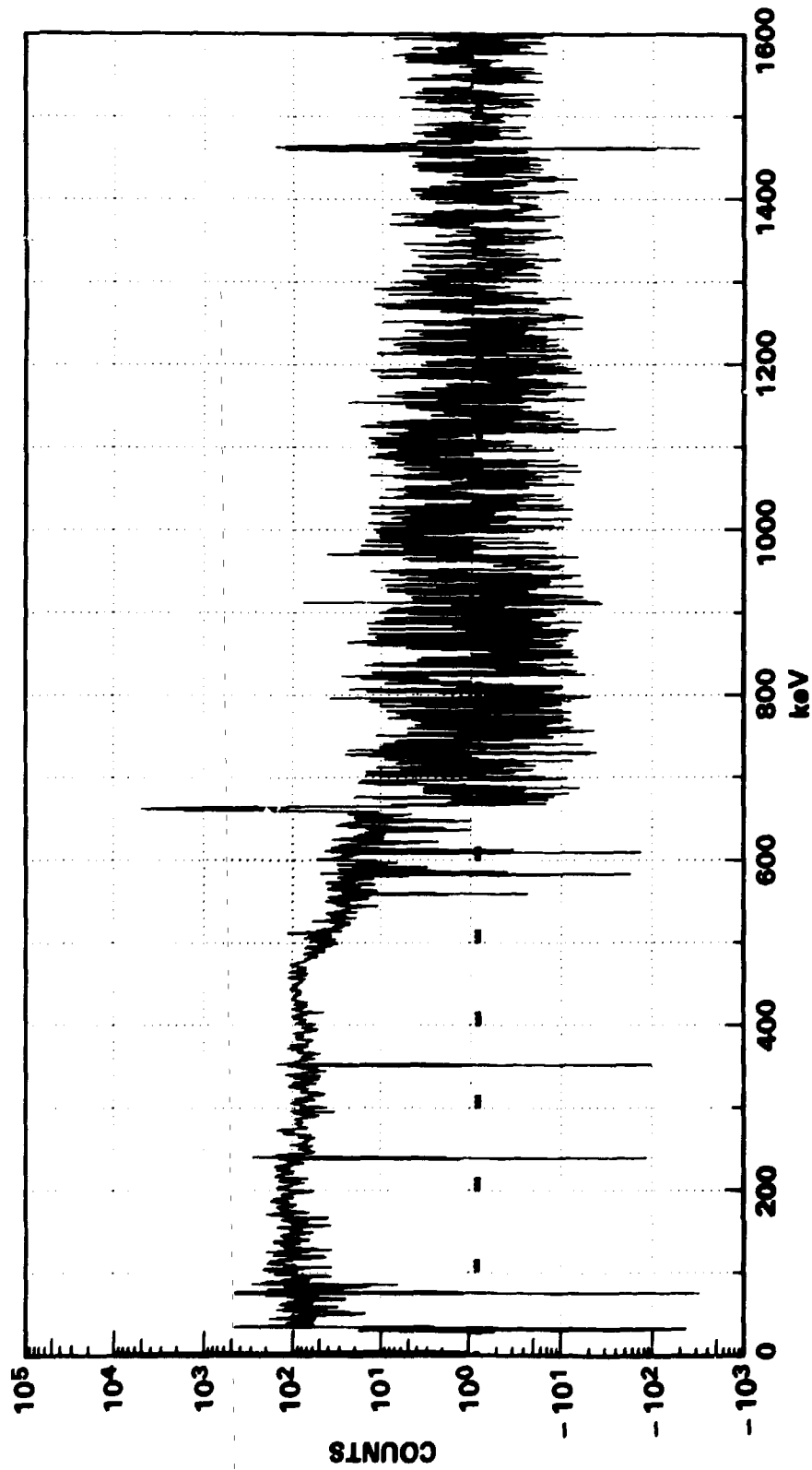


Figure 5.  $^{137}\text{Cs}$  Reference Spectrum.

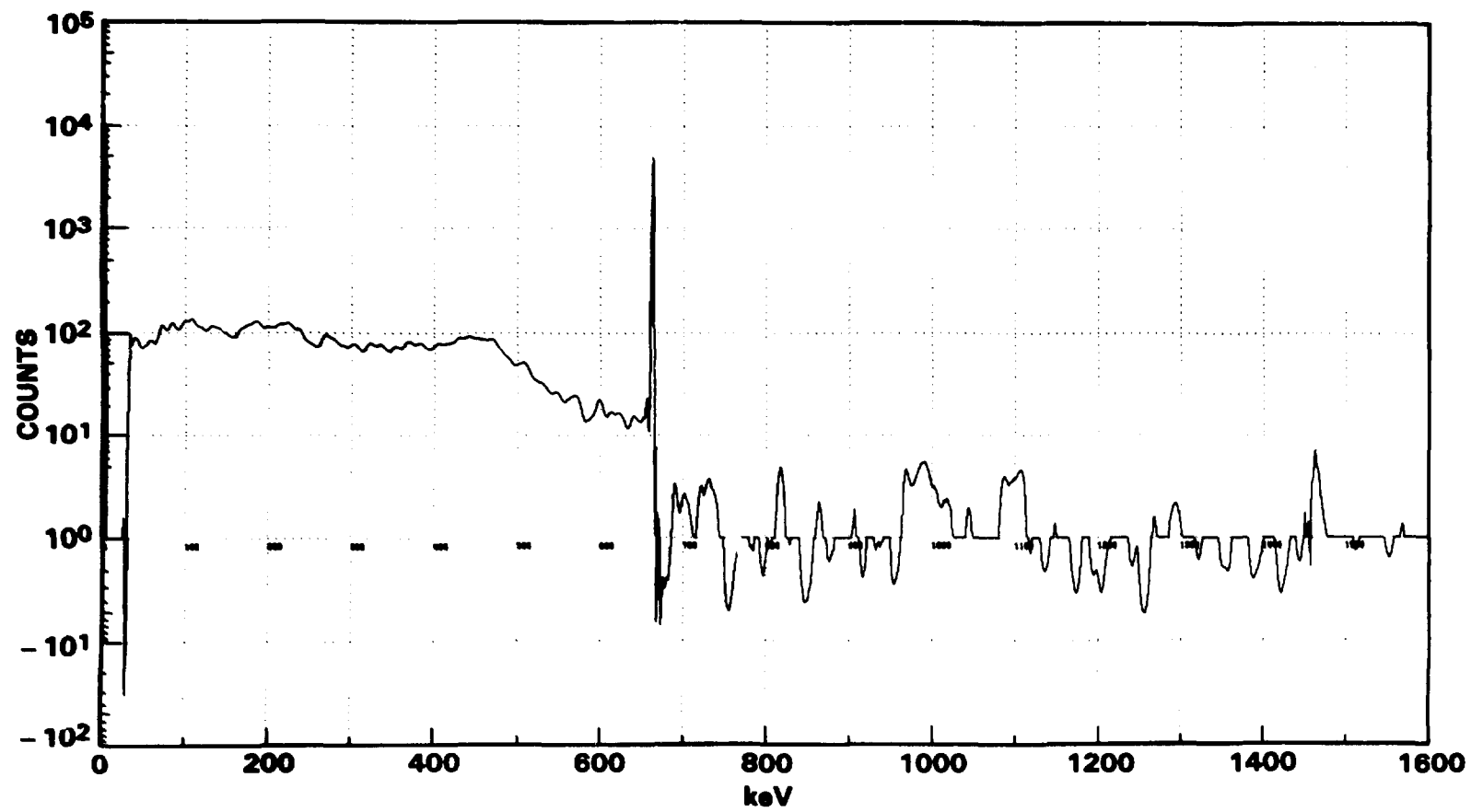


Figure 6. Smoothed Version of Figure 5.

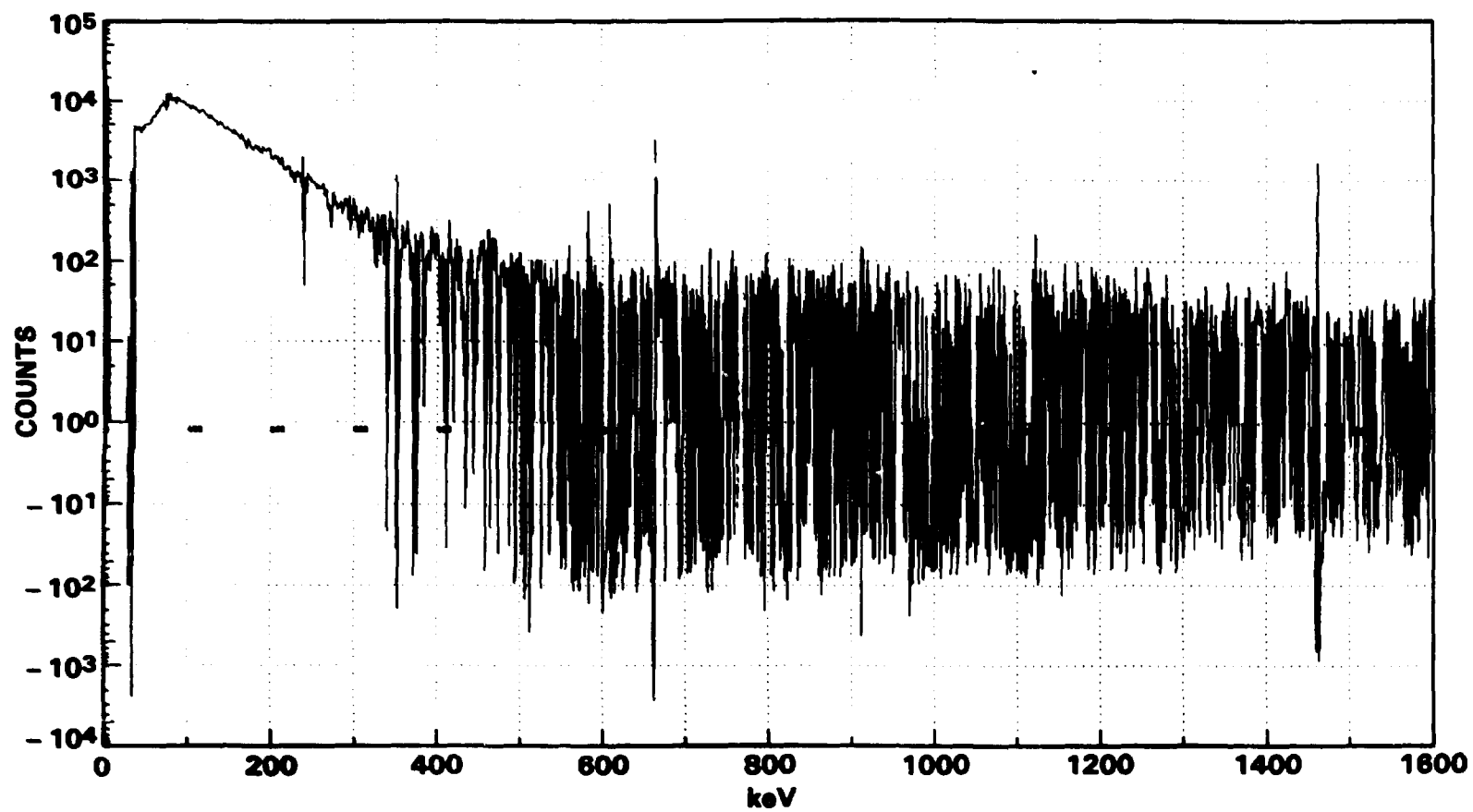


Figure 7. Gamma-ray Spectrum from Figure 3 with the  $^{137}\text{Cs}$  Component Subtracted.

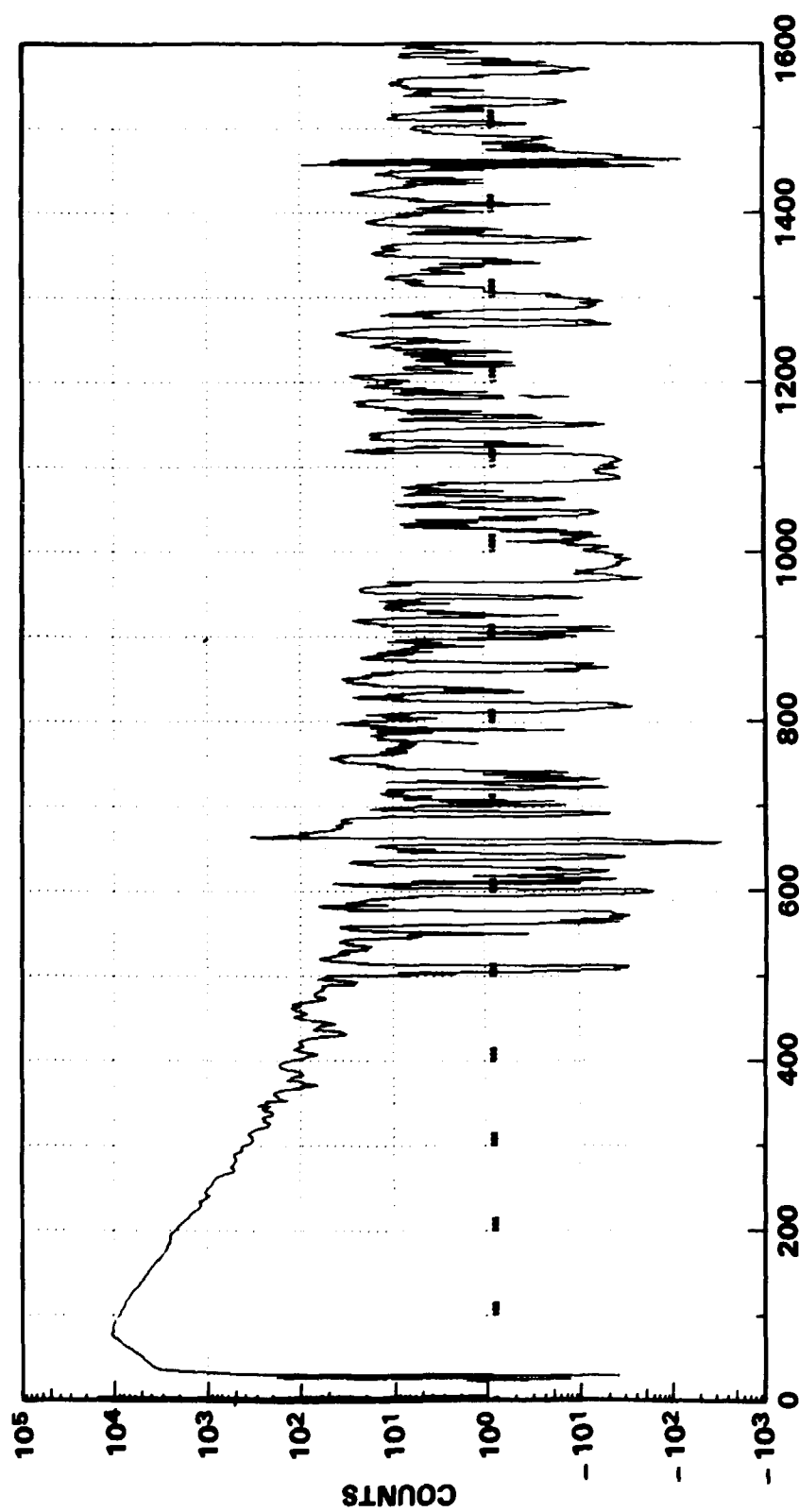


Figure 8. Smoothed Version of Figure 7.

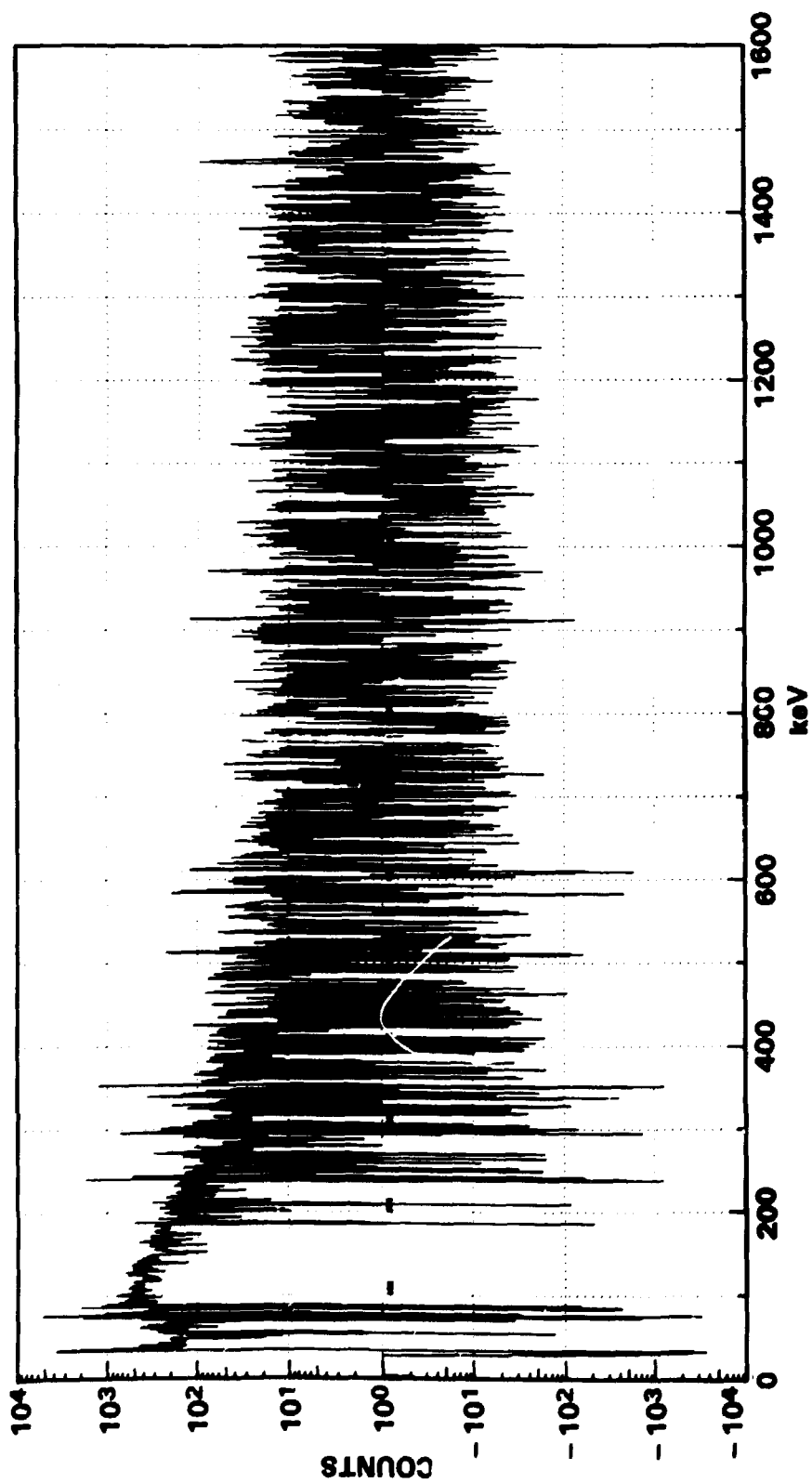


Figure 9.  $^{90}\text{Sr}$  Reference Spectrum.

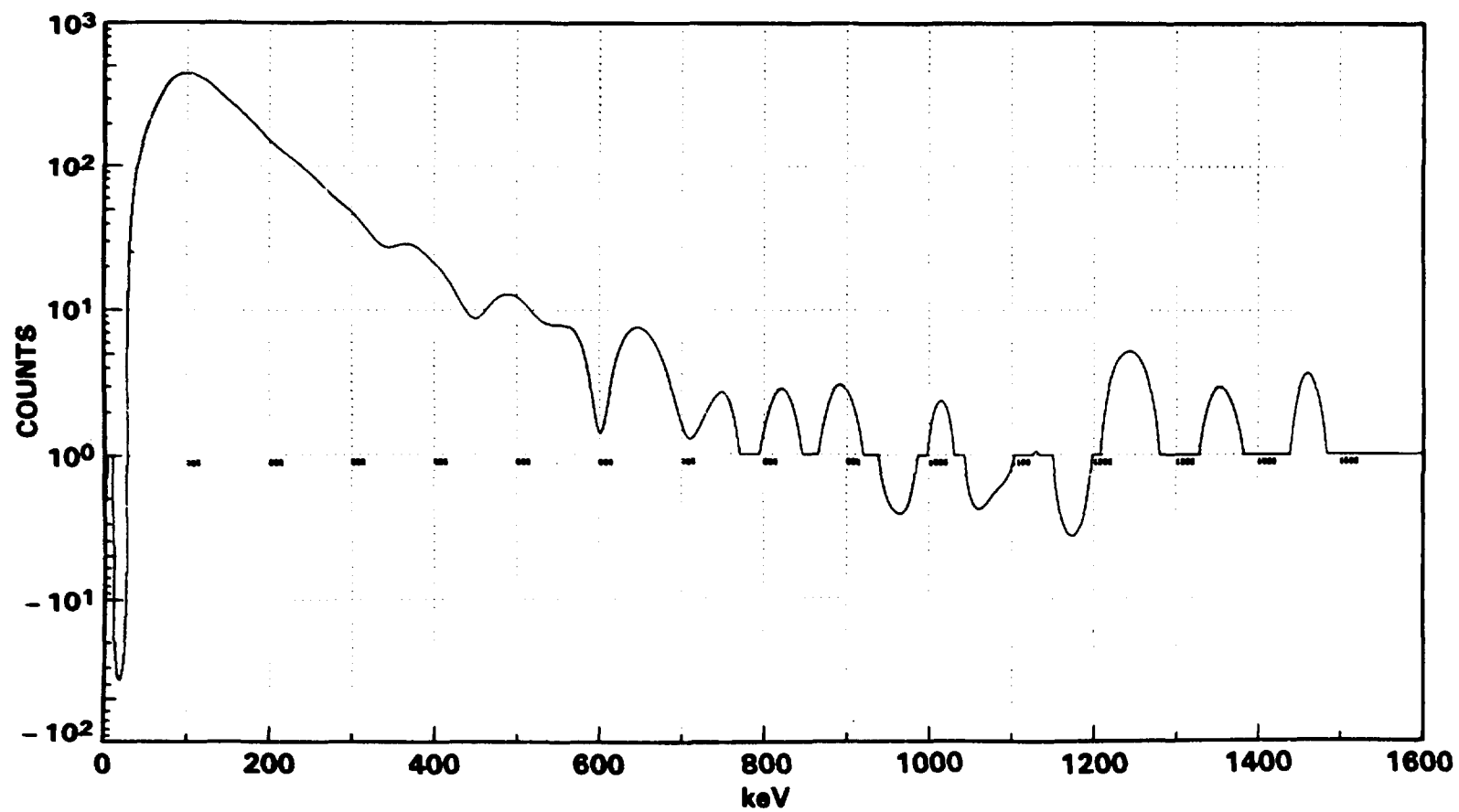


Figure 10. Smoothed Version of Figure 9.

## IN SITU DETERMINATION OF TRITIUM

The techniques considered in the course of this work for application to the in situ analysis of tritium during well logging operations included laser excitation analysis, mass spectrometric analysis, and gas proportional beta particle counting. The results of the first two techniques have been discussed in detail elsewhere<sup>(5)</sup> and will not be repeated here except to recall that the techniques both required very expensive and delicate equipment. The decision was made that unless adequate sensitivity could not be attained with the gas proportional technique, additional investigation of laser excitation and mass spectrometry was unwarranted. Consequently, development of a Nafion sampling/gas proportional counting tritium well logging device was pursued under subcontract with Professor Clarence Menninga of Calvin College, Grand Rapids, Michigan. The principle and design of the logging system was published in a manual<sup>(6)</sup> issued earlier, and an operation manual describing the as-built system and its operating parameters is currently in preparation.

Basically, the device is a flow-through instrument which samples, purifies, and analyzes sequentially. All controls are from a panel above ground, and an umbilical cable provides support, electrical, gas, water, and vacuum connections to the probe. The system can sample either water vapor or liquid water, collect the sample in a thermoelectric cold trap, passes the collected sample through a magnesium reduction furnace generating hydrogen (tritium) gas which is mixed with P-10 (90% argon - 10% methane) counting gas to prescribed pressure levels, and the tritium beta particles are counted in an anti-coincidence shielded Geiger type proportional counter using pulse height discrimination for additional background reduction. The various stages of the sampling/counting system are separated by an extensive set of solenoid valves so that different operations can be executed simultaneously. The assembled probe, including the sampling station, is 3.86 meters long and is light enough to be handled by one man. The proportional counter response function is linear over more than three orders of magnitude in tritium concentration, and the detection limit in actual logging operations has been demonstrated to be on the order of 50 pCi/ml. Considering that the MPC for tritium in an uncontrolled environment is 3000 pCi/ml, the logging device has more than adequate sensitivity. The sampling/counting system does have a memory effect of about 1% between successive samples; therefore, on those occasions when successive samples demonstrate reductions in tritium concentration of more than an order of magnitude, it is advisable to make replicate measurements to avoid introducing memory errors.

The logging device is completely assembled, laboratory tested, and available for use at this time; although field tests have not yet been conducted. It is hoped that additional funding can be obtained to perform a series of field tests and disseminate the technology to a world-wide audience through open literature publications. An expres-

sion of interest in the technology has been received from the Nuclear Regulatory Commission for application to measurements at existing commercial low level waste disposal sites.

### IN SITU DETERMINATION OF TRANSURANICS

Two separate techniques for the in situ determination of transuranics have been developed in the course of this work. The first system is a very sensitive photon detection technique capable of determining  $^{235}\text{U}$ ,  $^{238}\text{U}$ ,  $^{237}\text{Np}$ ,  $^{239}\text{Pu}$ ,  $^{241}\text{Pu}$ , and  $^{241}\text{Am}$  in normal well logging operations at sensitivities better than 10 nCi/g in environments not contaminated with large amounts of gamma-ray emitting fission products. This work has been published in the open literature(7), and the details will not be repeated here.

In the special case of matrices contaminated with high levels of gamma-ray emitting radionuclides, a passive neutron monitoring technique has been developed which is capable of determining transuranic concentrations at less than 10 nCi/g even in gamma fields of megarads/hour. This is a two-stage logging operation which is slower than the photon technique, but is the only technique able to measure low concentrations of transuranics in certain extreme environments. Basically, the method involves exposure of a metal form in the environment to be assayed and subsequent above-ground analysis of the neutron induced activation products generated in the metal. Extensive calibrations have been completed which relate the measured neutron flux to the absolute transuranic isotopic composition. A computer program has been written which requires input of the estimated (or known) relative transuranic isotopic composition, the moisture content of the matrix if greater than 16%, the diameter of the monitoring well, the matrix material (e.g., sand, salt, etc.), and the relative concentrations of fluorine, chlorine, carbon, oxygen, sodium, and aluminum in the matrix material. This chemical composition is important since ( $\alpha$ ,n) interactions on these light isotopes produce different neutron fluxes. This technique and calibration method have recently been described in an open literature publication(8) and will not be described further here.

A field test of the technique has recently been performed inside of a high level liquid waste tank. Flux measurements at eight different locations indicate the liquid is extremely homogeneous throughout the tank, and the tank has been sampled for laboratory analysis to cross-calibrate the in situ measurement technique. As soon as the laboratory analyses are completed, precise calibration factors will be generated with the computer program, and the plutonium concentrations determined by the two methods will be compared. Final adjustments to the calibration factor generation program can be made at this time if necessary. For the sake of comparison, this in situ technique costs about \$50 per analysis and requires 4-24 hours before results are available. Sampling of the liquid waste tank and laboratory analysis costs more than \$10,000, and the results are not available for more than two months.



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APPLICATION OF BOREHOLE GEOPHYSICS TO THE ACOUSTIC  
CHARACTERIZATION OF LOW PERMEABILITY ROCKS

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ABSTRACT

Down-hole acoustic measurements have important applications in the siting of proposed radioactive-waste storage facilities because of the close relationship between acoustic velocity and attenuation, and such rock properties as porosity, permeability, bulk modulus, and shear and compressive strength. Acoustic properties also are quite sensitive to the presence of fractures and shear zones that can be localized conduits for ground-water flow in otherwise impermeable rocks. In-situ acoustic measurements, however, contend with a basic trade off between depth of penetration and resolution, because high frequencies provide good sensitivity to porosity and permeability, while long acoustic wave lengths are needed for the sampling of large volumes. The U.S. Geological Survey has approached this problem by using megahertz frequencies to identify fractures, which are then sampled over larger volumes by means of lower frequency (tens of kilohertz) acoustic-waveform logs, giving depths of investigation on the order of tens of centimeters.

Initial applications of this approach were made in boreholes penetrating a homogeneous granite batholith containing isolated fractures that were clearly identified in core samples (Paillet, 1980 (1)). Numerical models of acoustic propagation in this homogeneous rock showed that existing theory was inadequate, and that recorded waveforms included modes of propagation that had never before been identified (Paillet, 1982 (2)). The improved numerical theory also identified the physical mechanisms involved in the interaction between wall rocks and borehole fluid by properly accounting for all modes of vibration (Paillet and White, 1982 (3)). An important result of this initial work was the observation that the attenuation associated with isolated fractures correlated with fracture permeability.

The initial results from acoustic studies using homogeneous granitic rocks were extended to an intensely fractured and intruded gneiss at another location on the Canadian shield (Paillet, 1981 (4)). These results indicated that: 1. Measured fracture permeability correlated with the interval-averaged value for tube-wave attenuation; 2. The method could not be applied at depths less than 50 meters, and 3. Acoustic frequencies of tens of kilohertz did not provide enough depth of penetration to resolve the character of large, intensely fractured, and altered zones.

The successful application of acoustic-waveform techniques to fractured crystalline rock indicated that the same techniques might be useful in characterizing the hydrologic properties of fractured shales. As in the case of granitic rocks, a review of existing numerical theory showed that propagation of acoustic energy along boreholes in shale was still poorly understood. Improvements in theory show that trapped fluid modes in boreholes in shale are fundamentally different from those in harder rocks. The difference is related to both the relatively high Poisson's ratio of shale relative to granite, and to the ratio of shear velocity in the shale to the acoustic velocity in the borehole fluid.

Tests of the theory for acoustic propagation in shales have been completed at a site near Fort Pierre, South Dakota. The recorded waveforms for observed trapped fluid modes and those predicted from the theory are almost the same. The test also involved filling previously dry boreholes with water. Repeated acoustic-logging runs using three different frequencies and at various times after filling the holes show that the hydration of the shale decreases the acoustic velocity with time as proposed by other investigators. Future model calculations may relate the variable source frequencies to specific radii of penetration so that the initial rate of shale hydration in the vicinity of the borehole can be determined. These results may provide important information about the in-situ porosity and permeability of unsaturated shales, along with improved characterization of secondary permeability in fractured zones.

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## MOBILITY OF ORGANIC COMPLEXES OF RADIONUCLIDES IN SOILS

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This work is a continuation of a program whose purpose is to provide some of the information necessary to the prediction of the migration of radionuclides from low-level waste disposal sites. Specifically, this program addresses the effects of organic complexing agents on the migration of radionuclides in soils; its objective is to develop and employ standard procedures for evaluating these effects. It supports Low-Level Waste Management Program Milestone C, provide documentation required to open a shallow land burial site.

Previous work on this program has been presented elsewhere (1, 2, 3). The work during FY 1982 has been described in a topical status report (4), which is scheduled for issuance in September 1982. This paper provides a summary of some of the results contained in that report.

Data were obtained for two elements that have radioactive isotopes that are important to low-level waste disposal, Ni and Co; for four organic complexants commonly used in the nuclear industry, EDTA, DTPA, oxalate, and citrate; and for two soils from sites that contain low-level waste disposal sites, Hanford, Washington, and Savannah River, South Carolina. Important variations in the effects of the complexants on radionuclide mobility were observed among the examined systems. These variations involved the rates at which reactions involving the complexes occurred as well as the extent to which the reactions occurred.

### EXPERIMENTAL METHODS AND RESULTS

The experimental approach has continued to involve measurements of radionuclide distributions between soil and solution phases in batch-type experiments, generally using a soil-to-solution ratio of 0.033 g/mL. Most of the work was done in air, but some work with Co was done in argon to prevent air oxidation to Co(III) from occurring. Measurements are made as a function of time, and the equilibrium position is approached from both directions. In approaching equilibrium from the precomplexed direction, the element and complexant are mixed and allowed to react before contact with soil is begun. In approaching equilibrium from the presorbed direction, the element is sorbed by the soil before the complexant is added. Equilibrium data are expressed as

an apparent  $K_d$ , which is defined as the quantity of element per gram of soil divided by the quantity of element per milliliter of solution

One important observation made this year is that the equilibrium effect of some complexants is reached more rapidly with Savannah River soil than with Hanford soil. An illustration of this fact is contained in Figure 1, where data in the Ni/EDTA system are compared for the two soils. With Savannah River soil, the same concentration of Ni is in solution after about 20 days in the experiment from the precomplexed direction as in the experiment from the presorbed direction. This is good evidence that a true equilibrium condition was reached with the Savannah River soil. However, with Hanford soil, a wide variation still exists between the precomplexed and the presorbed experiments after 200 days.

The reason for the marked difference in behavior with the two soils is not known, and is under investigation. A prime suspect is solution pH; the pH of solutions in contact with Savannah River soil was about 6 and that of solutions in contact with Hanford soil was about 9.

Another important observation is that the equilibrium effect of some complexants is reached more rapidly with some complexants than with others. Figure 2 presents data comparing EDTA and oxalate in the Co(II)/Hanford soil system. An equilibrium position is obtained with oxalate within 1 week, but the EDTA experiments are still a long ways from equilibrium after 6 weeks. Rapid attainment of equilibrium was observed with oxalate in all of the systems examined.

Rapid complex dissociation to allow sorption of the radionuclides by soils has also been observed with citrate complexes. Data illustrating this fact are shown in Figure 3; in the precomplexed experiment shown here the Ni concentration dropped quite rapidly for the first 5 days or so. The presorbed experiment gave some surprising results; the Ni concentration first increased rapidly and then began to fall. This suggests that the free citrate concentration is decreasing with the time of soil contact. Such a decrease could result from citrate decomposition or from complexing of citrate by a metal ion that is slowly leaching from the soil. An experiment to test these hypotheses was done under the same conditions as the precomplexed experiment except that no Ni was present. The concentration of Ca, which is the predominant metal ion in leachates of this soil, was 0.003 M after 2 days, and 0.00078 M after 14 days. This drop in Ca concentration with time is just the reverse of that predicted by the hypothesis involving complexing by soil-derived metal ions but is not inconsistent with the hypothesis that citrate is decomposing.

Generalizations regarding the rates of reactions in the systems examined are 1) reactions involving oxalate and citrate proceed more rapidly than those involving DTPA, which, in turn, proceed more rapidly than those involving EDTA, 2) reactions are more rapid in Savannah River

## COMPARISON OF SOILS IN Ni/EDTA SYSTEM

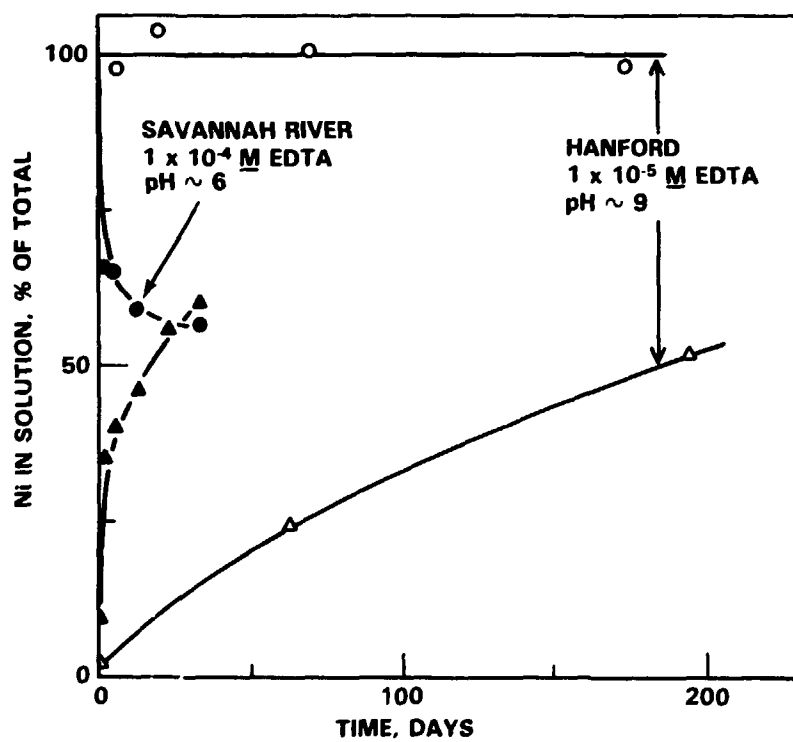


Figure 1.

### COMPARISON OF COMPLEXANTS IN Co(II)/HANFORD SOIL SYSTEM

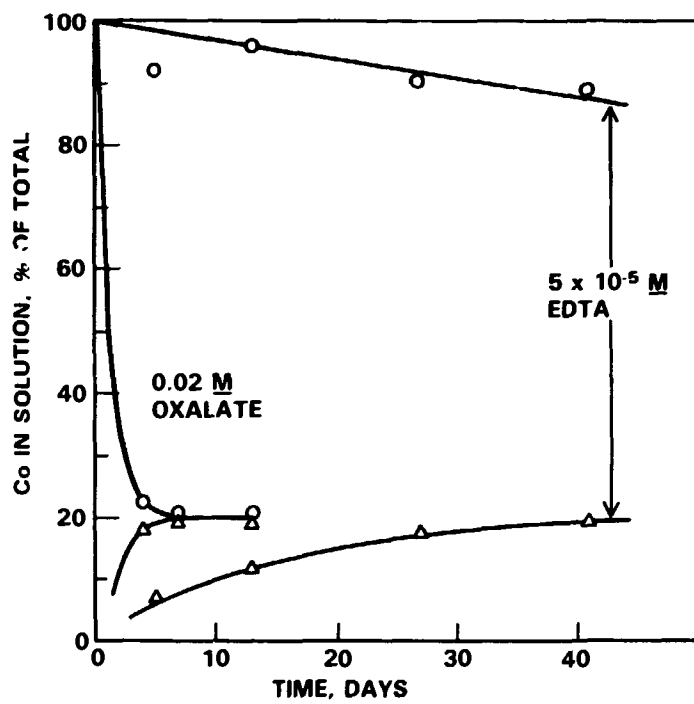


Figure 2.

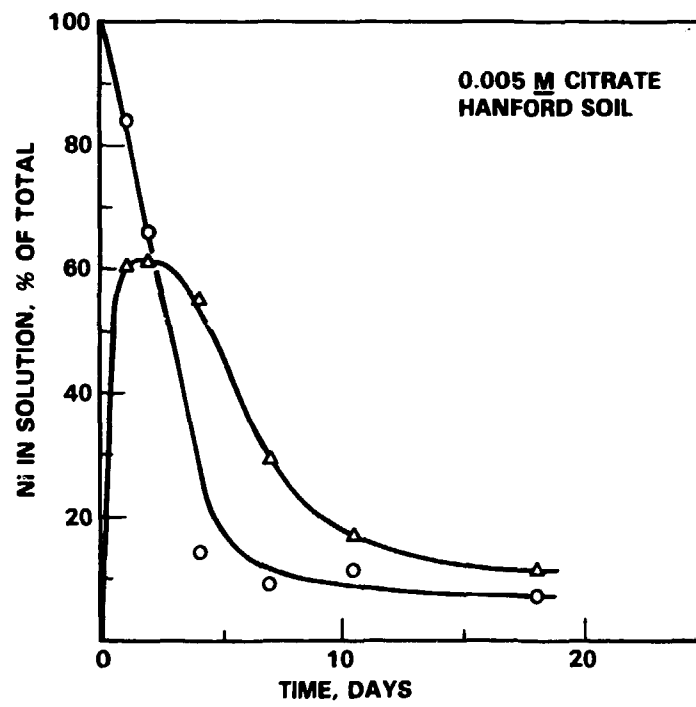
**Ni/CITRATE SYSTEM BEHAVIOR**

Figure 3.



soil systems than in Hanford soil systems, and 3) reactions involving Co are more rapid than reactions involving Ni.

The preceding discussion has emphasized the kinetic aspects of the study by comparing the rates at which various reactions occur. Enough experiments have been done in some systems to determine the thermodynamic aspects; that is, the extent to which reactions have proceeded when equilibrium has been attained. The available data are presented in Figure 4; in some cases error bars are included to represent the spread between the values obtained in precomplexed- and presorbed-type experiments at the time of writing; much smaller error bars are expected after longer time periods. The Eu/EDTA data reported earlier (1) are included in Figure 4 along with the data obtained this year.

With Hanford soil, the trend of the data agrees in most cases with that expected from the published values of the stability constants for the various complexes. That is, of the systems examined, the Co(II)/oxalate complex is the weakest and Ni/oxalate is next weakest; Ni/DTPA is the strongest complex and Co(II)/DTPA is the next strongest; however, the fact that the Eu/EDTA complex is stronger than the Co(II)/EDTA complex is not reflected in these data. The data from Savannah River soil do not reflect the fact that the Ni/DTPA complex is stronger than the Ni/EDTA complex. In the Ni/DTPA system, where data are available for both soils, a higher concentration of DTPA is necessary to give a low  $K_d$  value with Savannah River soil than with Hanford soil.

The results shown in Figure 6 were obtained at a soil-to-solution ratio of about 0.033 g/mL. A variation of  $K_d$  with the soil-to-solution ratio has been observed in some systems; it is assumed that this is due to differences in the free complexant concentration (the quantity that actually governs the equilibria), as varying amounts of the complexant are complexed by metal ions that leach from the soil. Thus, at a higher soil-to-solution ratio, a lower free complexant concentration would be obtained with a given total complexant concentration, and the lines in Figure 4 would shift to the right (a higher total complexant concentration would be required to produce a given change in the  $K_d$  value).

## DISCUSSION

The results presented above allow some qualitative conclusions regarding the effects of organic complexants on the mobilities of low-level waste radionuclides in soils to be drawn. The major conclusion is that appreciable organic-complexant caused radionuclide mobility is most likely to occur with precomplexed radionuclides in systems that reach equilibrium slowly. Since oxalate and citrate complexes of Ni and Co reach equilibrium much more rapidly than do the EDTA and DTPA complexes,

## EQUILIBRIUM DATA SUMMARY

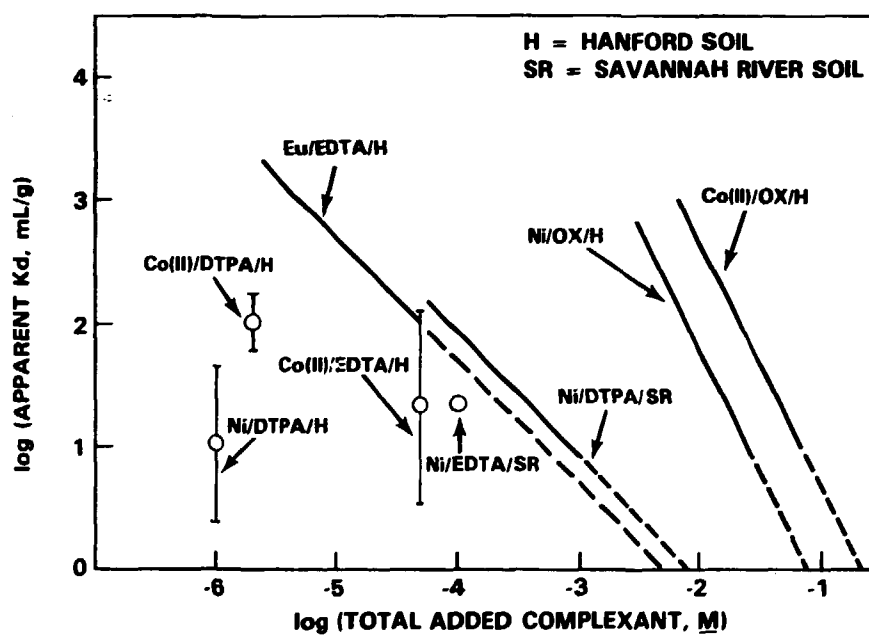


Figure 4.

they are much less likely to cause problems. Discussion of some of the factors affecting these conclusions follows.

The equilibrium distribution data in Figure 4 are extrapolated to the complexant concentrations at which the  $K_d$  values are estimated to reach a value of 1 mL/g. While such a  $K_d$  value is really not very high, it is still sufficient to provide an appreciable retardation of the radionuclide in a groundwater system. The ratio of the rate of water movement to the rate of radionuclide movement, termed the retardation factor, to the  $K_d$  is

$$\text{Retardation Factor} = 1 + \frac{(\text{bulk density})(K_d)}{(\text{effective porosity})}$$

If the bulk density is 2 g/cm<sup>3</sup> and the effective porosity is 0.2, values which are not unreasonable, a  $K_d$  value of 1 mL/g gives a retardation factor of 11. That is, the radionuclide would move through the soil 11 times more slowly than the water. Similarly, a  $K_d$  value of 10 mL/g would give radionuclide movement 100 times slower than the ground water movement for these conditions.

It can be seen in Figure 4 that the complexant concentrations required to give a  $K_d$  value as low as 1 are quite high for several systems. Thus, unless very high complexant concentrations are maintained in these systems, the mobility of these radionuclides in systems where equilibrium is obtained rapidly would not be high unless ground water flows were very high. However, if ground water flows are high, there may be appreciable dilution of the complexant, which would result in increased  $K_d$  values and thus increase the retardation factor. It is thus thought to be relatively unlikely that pronounced mobility of Ni and Co would occur in several of these complexant/soil systems in which equilibrium is obtained rapidly.

However, the systems that reach equilibrium slowly do afford the possibility of appreciable organic complexant-caused movement of Ni or Co. In such cases, data such as those presented in Figure 4 are relatively unimportant; what is important are the rates of complex dissociation and the rate at which the water moves through the soil, since the associated complexes will move as fast as the water.

The data obtained with the Hanford and Savannah River soils provide examples of different mechanisms of limiting the migration of radionuclides from waste disposal sites. With the Hanford soil/ground water system, where the Ni-EDTA complex dissociates very slowly and the Ni migrates through the soil at the same rate as the ground water, the rate of migration of precomplexed Ni from a disposal site would still be low because the low rainfall rate gives a low rate of water migration. With the Savannah River soil/ground water system, where a relatively high rainfall rate could give a high rate of water migration through the

soil, the Ni-EDTA complex dissociates more rapidly and the uncomplexed Ni can then be sorbed by the soil. Thus little of the precomplexed Ni may migrate at the same rate as the water.

Another important factor in the systems that reach equilibrium slowly is whether or not the radionuclide is precomplexed before it is exposed to the soil. In the slowly-equilibrating systems, this factor has a tremendous effect on the extent of radionuclide migration to be expected for a given distance of water migration. While the precomplexed radionuclide may migrate at the same rate as the ground water if the rate of complex dissociation is sufficiently slow; if the radionuclide is sorbed by the soil before it is exposed to the complexant, it may not travel far, even if the same quantity of complexant is present. This is because of slow desorption rates, especially at low complexant concentrations.

Another factor that would tend to reduce the likelihood of serious complexant-caused movement of radionuclides is that of complexant decomposition or degradation, which results in higher radionuclide  $K_d$  values as the free complexant concentration decreases. Such decomposition appears to have occurred in the citrate experiments discussed earlier. The environmental chemistry of several complexants has recently been summarized (5). Little quantitative data are available on the rates at which such reactions occur, but such decomposition or degradation would tend to reduce the extent of complexant-caused movement of radionuclides, at least in systems that reach equilibrium rapidly. The effect of such decomposition on slowly equilibrating systems is not known; depending on what factors control the rate of complex dissociation, complexant decomposition may not have a significant effect.

Several factors discussed here point to a conclusion regarding the effect of certain organic complexants on radionuclide mobility. The presence of oxalate or citrate in low-level waste would be much less likely to cause radionuclide mobility problems than would EDTA or DTPA. This results not only from the fact that oxalate and citrate form weaker complexes than do EDTA and DTPA, but also from the fact that oxalate and citrate do not form kinetically-inert complexes with Co and Ni, as do EDTA and DTPA under some conditions. Also the decomposition rate of citrate appears to be quite rapid.

Perhaps this conclusion regarding the relative desirabilities of having the different complexants present in low-level waste could affect decisions regarding the degree of treatment required for wastes before they can be disposed. Such decisions could, in turn, perhaps affect decisions as to which complexants are used in the nuclear industry. Such "trade-off" considerations are beyond the scope of this program, however.

## FUTURE WORK

Future work is planned to obtain more complete data in many of the systems under examination. A few additional soil and complexant systems may also be examined in the coming year. Of special interest here would be some unknown complexant mixtures present in proprietary decontaminating agents. This project is scheduled for termination in FY 1983, and a final report will be issued covering what data are available at that time.

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RADIONUCLIDE MIGRATION STUDIES  
AT THE SAVANNAH RIVER PLANT LLW BURIAL GROUND,  
A HUMID SLB SITE

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ABSTRACT

Experimental studies of the Savannah River Plant burial ground include subsurface monitoring, lysimeter tests, and soil/water chemistry. Groundwater is regularly monitored for traces of radioactivity and mercury. Only tritium has been found to any appreciable extent. Recently, Tc-99 was found in very low concentrations in the groundwater flow path. To define radionuclide migration further, 42 defense-waste lysimeters containing well-characterized waste forms have operated 2-4 years. Traces of radioactivity now appear in percolate water from some of the lysimeters. Ru-106 is the only gamma emitter released so far. Chemical studies are underway on soil/waste/water interactions in the burial ground. A comprehensive study of the composition of groundwater and trench water shows several chemical indicators of potential migration.  $K_d$ 's for Cs-137 and Sr-90 in trench waters are lower than in groundwaters.  $K_d$ 's for Tc-99 as  $TcO_4^-$  are very low. Technetium can migrate at greater than 88% of groundwater velocity.

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INTRODUCTION

Low-level waste migration studies at the Savannah River Plant (SRP) are designed to develop a fundamental understanding of the soil/waste/water system of the SRP burial ground, that will ensure its safe operation now and in the future. These studies are performed by the Savannah River Laboratory (SRL), both for technical support of ongoing waste disposal operations by SRP and for development of generic technology for the DOE Low-Level Waste Management Program (LLWMP).

The purpose of the LLWMP task is to define the migration of radionuclides from an operating shallow land burial (SLB) site in a humid

region. The results will be pertinent to the LLWMP milestone for development and documentation of technology required to open an SLB site. The work at SRL will contribute to the LLWMP technology development areas of (1) a radionuclide transport data base and (2) migration mechanisms. The SRL studies also afford the opportunity to observe the consequences of past burial practices that are no longer followed at SRP, including burial of transuranic wastes and burial of unencapsulated tritium-bearing wastes. Each of three subtasks will be discussed:

- o Subsurface monitoring
- o Lysimeter tests
- o Soil/water chemistry

The SRP low-level waste burial ground is a 195-acre plot located in an interstream area between two radiochemical separations plants. A 76-acre portion that was filled in 1972 has been the subject of intensive investigation, from which the most recent results are reported in this paper. The remaining 119-acre portion of the burial ground is the site of current operations that are expected to continue until the burial ground is filled in the early 1990's.

Trenches in the SRP burial ground are 20 feet deep, 20 feet wide, and typically 300 to 1000 feet long. The trenches are filled with waste to within 4 feet of the surface and then are backfilled with native soil.

In the closed 76-acre burial ground, the water table typically is 45 feet from the surface. Groundwater flow generally is westward across the approximately rectangular site, until near the west end where the flow turns southward to exit the burial ground. Eventually this groundwater outcrops into a tributary of Four-Mile Creek, which flows into the Savannah River.

#### SUBSURFACE MONITORING

In this subtask, the movement of radionuclides and hazardous chemicals from trenches in the SRP burial ground is monitored. The purposes of the subtask are: (1) to develop a generic methodology for characterizing a humid SLB site; (2) to identify a generic methodology for assessing the performance of a humid SLB site; (3) to verify under field conditions the results of laboratory and lysimeter tests for predicting migration of radionuclides from humid SLB sites.

The principal methods of detecting radionuclide migration from the burial ground trenches are (1) analysis of water samples from monitoring wells, (2) location of radioactivity in dry boreholes in selected trenches, and (3) analysis of soil cores. Most of the information available to date is from the first two methods, although core samples will be important later in locating slowly migrating radionuclides.

Over the years numerous monitoring wells have been placed in and around the SRP burial ground. Monitoring responsibilities are divided between the SRL and SRP organizations. Routine environmental monitoring is conducted by the SRP Health Protection Department, while monitoring for long-term research projects is carried out by the SRL Waste Disposal Technology Division. Only results of the latter studies will be discussed here.

Most of the wells are screened at the water table, but a few wells in the groundwater flow path south of the burial ground are screened at various depths below the water table. Also, several wells are screened at the bottoms of trenches to detect perched water. Of particular interest is a system of 64 water table wells located on a grid with 200-foot spacings across the 76-acre burial ground. These wells permit a statistical sampling of groundwater beneath the burial ground.

Gross Radioactivity. SRL regularly samples these and other wells, measures the water table elevation, and analyzes the groundwater samples for gross alpha, gross nonvolatile beta/gamma, and tritium activity. Only 9 of the 64 grid wells have shown any alpha or nonvolatile beta/gamma contamination, and the concentration of radioactivity in these wells has shown a diminishing trend in the past few years. Migration of activity into most of these wells was apparently caused by former equipment decontamination operations or by inadvertent solvent spills.

Tritium. Several years ago, the monitoring program revealed that the burial ground provided less than perfect confinement for tritium. A plume of leached tritium is moving with the groundwater toward an outcrop in Four-Mile Creek. In 1979 the maximum concentration of tritium in the plume was 64,000 pCi/mL, declining to 6,000 pCi/mL at the outcrop. About 850 Ci/yr of burial-ground tritium were released at the outcrop, or about 2% of annual SRP releases to surface waters.

Several measures were taken to improve tritium confinement: (1) A special set of wells and corings were drilled to define the extent of the plume and quantify the amount of tritium in it. (2) The flow path to outcrop was lengthened by about two tritium half-lives by repair of an eroded ditch and diversion of a large flow of surface water into an engineered channel, thus minimizing future erosion and premature outcrop. (3) New tritium-bearing wastes are sealed into spent melt crucibles to impede leaching.

The monitoring program allows the tritium plume to be followed with time and has shown that the repair work was successful in lengthening the tritium flow path. Tritium in the groundwater at the SRP burial ground will continue to be monitored to assure that the plume behaves as expected.

Technetium-99. The presence of the tritium plume provides a convenient means of defining the direction and velocity of flow of groundwater that has been in contact with buried waste. Any radionuclides



that migrate with velocities approaching that of groundwater also would be expected in the tritium plume. Tc-99 is one such radionuclide, of environmental concern because of its long half-life (210,000 yr) and suspected mobility as the pertechnetate anion,  $\text{TcO}_4^-$ . Tc-99 is a fission product produced in high yield, and thus should be plentiful in fission-product low-level waste.

By special analyses of samples from a well outside the burial ground but within the groundwater flow path, small amounts of Tc-99 have been detected recently. Replicate samples contained 8+5 and 22+11 pCi/L of Tc-99, compared to 0.02 pCi/L in a control sample from a well upflow from the burial ground. Measurements were by an ultra-low-level mass-spectrometric technique. Thus, the mobility of Tc-99 has been confirmed in a field monitoring test that corroborates laboratory experiments, to be discussed later. Tc-99 concentrations in about 20 selected grid wells will be determined in the coming year.

Other Radionuclides. Future emphasis of the monitoring program will be on determination of individual radionuclide concentrations in groundwater from selected wells, particularly those wells with a history of alpha and nonvolatile beta/gamma contamination. In the past year, some 20 wells were surveyed for Pu-238, Pu-239, Sr-90, and gamma emitters, of which only Cs-137 and Co-60 were detected. The data are summarized in Table 1. Careful radiochemical separations and low-level counting techniques were required to obtain these results. Other than tritium, groundwater contamination at the SRP burial ground is minimal.

TABLE 1. Radionuclide Content of 20 Groundwater Wells  
at SRP Burial Ground

| Radionuclide       | Concentration, pCi/L |       |                 | Number<br>of Wells |
|--------------------|----------------------|-------|-----------------|--------------------|
|                    | Average              | Range | Detection Limit |                    |
| Co-60              | 13                   | ---   | 8               | 1                  |
| Sr-90 <sup>a</sup> | 18                   | 7-30  | 6               | 4                  |
| Cs-137             | 12                   | 10-16 | 8               | 3                  |
| Pu-238             | 5                    | 2-17  | 1               | 8                  |
| Pu-239             | 3                    | 2-4   | 1               | 3                  |

a. In addition, one well contained 1600 pCi/L of Sr-90

Mercury. In addition to radionuclides, the grid wells have been surveyed every 1-2 years for mercury in the groundwater. Concentrations have always been less than the 2 ppb drinking water limit. In the early years of burial ground operation, about 10 tons of contaminated mercury metal was buried in trenches along with other wastes. Analyses in 1982 found 23 of 64 wells with greater than the detection limit of 0.05 ppb for total mercury, with a range of 0.06 to 1.6 ppb and an average of 0.3 ppb. Annual monitoring for mercury will continue, to ensure that unacceptable amounts do not migrate to the groundwater.

### LYSIMETER TESTS

In this subtask, radionuclide migration from well-characterized sources emplaced in lysimeters is measured. The purposes of the subtask are: (1) to verify and evaluate under controlled conditions the results of laboratory tests for predicting migration of radionuclides from a humid SLB site; (2) to correlate results with actual field measurements; (3) to develop profiles of radionuclide movement through the soil columns; (4) to develop data on leach rates and migration from typical unencapsulated defense wastes.

A field of 42 defense-waste lysimeters, 10 feet deep and 6 or 10 feet in diameter, were constructed and started up between 1978 and 1980. The experiment is designed to study leaching and migration from 8 typical SRP defense waste forms under both saturated and unsaturated conditions. Each type of waste form was emplaced in a set of 5 lysimeters: two saturated, 6-foot-diameter; two unsaturated, 6-foot-diameter; and one unsaturated, 10-foot-diameter. Two control lysimeters contain no waste and are unsaturated, 6-foot-diameter. Conditions simulate those of the burial trenches as closely as possible. Percolate water may be collected and sampled, or discharged.

The waste forms contain known amounts of Pu-238, Pu-239, fission products (FP), and activation products (AP). The waste forms are: (1) separations laboratory glove-box waste (Pu-238); (2) reactor moderator deionizer resin (FP, AP); (3) plutonium finishing line job-control waste (Pu-239); (4) SRL high-level cells waste (Pu-238, Pu-239, FP); (5) separations canyon pipe jumpers (Pu-238, Pu-239, FP); (6) separations area laboratory waste (FP); (7) separations area canyon job-control waste (FP); (8) reactor area scrap metal (AP).

Hydraulic Performance. Lysimeter hydraulic performance for several years has been analyzed statistically. The amount of percolate water pumped out of the lysimeters may be expressed as a fraction of the cumulative rainfall into the lysimeters, the remainder being lost by evapotranspiration. In 1981, the mean hydraulic performance of lysimeters that were operating properly was (28+5)% of rainfall for 15 unsaturated 6-foot-diameter lysimeters, (32+5)% for 8 unsaturated 10-foot-diameter lysimeters, and (33+4)% for 11 saturated

6-foot-diameter lysimeters. Eight other lysimeters deviated significantly from the means and probably will require corrective action to restore proper hydraulic performance.

Results. Monthly samples of percolate water have been analyzed for gross alpha and gross beta/gamma activity. By these methods, about half the samples to date have been less than detection limits: <10 pCi/L gross alpha and <100 pCi/L gross beta/gamma. In samples with measurable activity, 10-140 pCi/L gross alpha and 100-2000 pCi/L gross beta/gamma have been observed. Although the data fluctuate widely, there is fair correlation with the known contents of the lysimeters.

Recently, a more sensitive method for gamma emitters was developed. Larger volumes of percolate water, up to 20 liters, are concentrated on mixed-bed ion-exchange resin, and then analyzed by high-resolution gamma-ray spectrometry. In this way, Ru-106 at up to 300 pCi/L has been detected in some of the current samples. No other gamma emitters from the waste have been found. Information is not yet available on levels of Pu-238, Pu-239, or Sr-90 released from the lysimeters.

In the near future, the defense-waste lysimeters will be equipped with ion-exchange columns so that preconcentration of activity in percolate water can be accomplished in the field. This change will increase sensitivity for radionuclide detection by a factor of about 100 from the gross counting methods. Also in the coming year, one lysimeter will be cored in order to measure radionuclide migration distances directly.

#### SOIL/WATER CHEMISTRY

In this subtask, laboratory and field data required to understand the fundamental chemical processes of the soil/waste/water system of the SRP burial ground are developed. The purposes of the subtask are: (1) to compare and evaluate different laboratory techniques for determining distribution coefficients,  $K_d$ ; (2) to determine the  $K_d$  of specific radionuclides; (3) to correlate the chemical properties of groundwaters from the burial ground with radionuclide transport mechanisms.

Groundwaters. A detailed chemical analysis of groundwater from 63 burial ground wells has been completed. (One of the 64 grid wells was temporarily unavailable for this study.) Chemical properties are summarized in Table 2 for those wells considered "normal" or probably not affected chemically by the presence of nearby waste. However, for each chemical property measured, usually one or more wells were identified as "anomalous", which generally could be taken as evidence that the nearby waste was affecting the well. In total, about 20 different wells were identified as anomalous in one or more chemical properties. Most of these wells were not contaminated with radioactivity. Of the 9 wells known to contain traces of radioactivity, only

TABLE 2. Normal Chemical Composition  
of 63 Groundwater Wells at SRP Burial Ground

| <u>Component</u>          | <u>Average<sup>a</sup></u> | <u>Range<sup>a</sup></u> | <u>Number<br/>of Wells</u> |
|---------------------------|----------------------------|--------------------------|----------------------------|
| Sodium                    | 7                          | 2-32                     | 61                         |
| Potassium                 | 2                          | 0.2-18                   | 62                         |
| Magnesium                 | 2                          | 0-40                     | 63                         |
| Calcium                   | 9                          | 0.09-56                  | 60                         |
| Strontium                 | 0.1                        | 0-0.5                    | 61                         |
| Barium                    | 0.3                        | 0-2                      | 62                         |
| Iron                      | 0.3                        | 0-13                     | 61                         |
| Silicon                   | 4                          | 0.08-11                  | 61                         |
| Phosphorous               | 0.07                       | 0-2                      | 61                         |
| Fluoride                  | 0.09                       | 0-0.4                    | 62                         |
| Chloride                  | 6                          | 0.9-45                   | 62                         |
| Nitrate                   | 10                         | 0-28                     | 63                         |
| Sulfate                   | 3                          | 0-24                     | 62                         |
| Dissolved Oxygen          | 6                          | 1.5-9                    | 63                         |
| Carbon (inorganic)        | 11                         | 0-57                     | 61                         |
| Carbon (organic)          | 4                          | 0-45                     | 62                         |
| pH                        | 5.4                        | 2.9-7.2                  | 62                         |
| Eh (mV)                   | 519                        | 264-683                  | 62                         |
| Conductivity <sup>b</sup> | 73                         | 15-360                   | 60                         |

a. Units: ppm, except for pH, Eh, and conductivity

b. Units: micromho-cm

4 are also anomalous chemically, but 3 of these were involved in a solvent spill.

Nevertheless, chemical factors could be responsible for slow migration that has not yet been detected. Three mechanisms may be involved, and chemical indications of each are available:

(1) Complexation to form mobile species may be possible with organics, carbonate, phosphate, and fluoride. (2) Radionuclide ion exchange with soil may be less effective as other ions compete for sites, if conductivity and concentrations of various cations are high. (3) Migration may be increased by loss of important iron(III) adsorption sites in soil by reduction to soluble iron(II), as indicated by low Eh or dissolved oxygen, and high iron concentration in the groundwater. These mechanisms also offer reasonable explanations for each well in which radioactivity has been detected.

Trench Waters. A similar chemical analysis has been performed for 7 of 24 trench wells that contained enough perched water to sample. Compared to groundwaters, trench waters have higher iron and calcium concentrations, higher conductivity, and lower dissolved oxygen and Eh. Batch  $K_d$  measurements for Sr-85 and Cs-137 tracers, between trench water and typical burial ground soil, were smaller by about a factor of 3 than with groundwaters. For strontium in trench water,  $K_d$  is 5 to 8 mL/g; for cesium,  $K_d$  is about 350 mL/g.

Technetium. Laboratory studies have shown that Tc-99 is mobile in SRP burial ground soils as pertechnetate anion,  $TcO_4^-$ . Batch  $K_d$  measurements with groundwater and low clay content (10%) soils gave 0.1 to 0.3 mL/g for Tc-99, which corresponds to 41-67% of groundwater velocity. In more realistic column tests, Tc-99 moved at greater than 88% of groundwater velocity. For higher clay content (45%) soils,  $K_d$  for Tc-99 was 1.3 mL/g, or 13% of groundwater velocity. Trench water and distilled water results were similar to those with groundwater. Irreversible sorption on soil was measured with Tc-99m tracer to give an upper limit of  $2.5 \times 10^{-13}$  g of technetium sorbed per g of soil. Relatively rapid migration of Tc-99 was confirmed by the field measurements discussed earlier.

In the coming year, laboratory studies of cesium, iodine, and plutonium migration are planned.

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The information contained in this article was developed during the course of work under contract No. DE-AC09-76SR00001 with the U. S. Department of Energy.

CHEMICAL CHARACTERISTICS, MIGRATION AND FATE OF RADIONUCLIDES  
AT COMMERCIAL SHALLOW LAND BURIAL SITES

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ABSTRACT

Experimental areas established at the Maxey Flats, Kentucky shallow land burial site permit sampling to define the actual chemical species of radionuclides that migrate by subsurface routes. Concurrently, studies are conducted to identify the chemical characteristics of waste trench leachate that retard or enhance radionuclide mobility; determine the chemical changes in leachate properties that result from installation of infiltration barriers; and provide the chemical and radiochemical input parameters essential for hydrological/geochemical modeling of subsurface radionuclide transport. Plutonium exists in subsurface waters at Maxey Flats in a reduced, anionic, highly complexed form. EDTA is the major organic component in waste trench leachate and in an inert atmosphere well. The areal distributions of surface radionuclides confirm that contamination at Maxey Flats has been largely contained within the restricted area; only tritium and  $^{60}\text{Co}$  levels have been influenced by the site in the areas immediately adjacent to the site.

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INTRODUCTION

Commercial shallow land burial sites located in the eastern half of the country where rainfall rates are relatively high experience greater problems with subsidence and seepage of water into and from the burial trenches than those located in the more arid regions. Interpretation is complicated by the fact that there are several processes for radionuclide transport besides subsurface transport. For example, at both Maxey Flats and West Valley, the burial trench cap material subsided and the reworking of the areas to form new caps has resulted in localized spread of contamination. At Maxey Flats additional known sources of contamination include atmospheric transport from the evaporator plume and spillage of radioactive liquids. Consequently, it is very difficult to differentiate between environmental

contamination from these sources and radionuclide migration by subsurface aquifer transport.

One of the most serious problems to be considered during the long-term isolation of wastes is the subsurface migration of radionuclides, which depends on the chemical forms of radionuclides. The oxidation state and ionic form will influence migration, and the formation of complexes or chelates with organic ligands may greatly facilitate the migration of radionuclides(1,2,3). Determination of the precise chemical species of migrating radionuclides is essential for an accurate evaluation of the long-term stability of shallow land burial sites. It is also essential to determine those compounds and ionic species which are present or can be generated in situ and subsequently participate in the formation of the chemical species that are shown to migrate. The data are required as chemical and radiochemical input parameters for hydrological/geochemical modeling of subsurface aquifer transport.

In recognition of these potential problems and needs, a broad, multidisciplinary, multilaboratory program was initiated by the U. S. Nuclear Regulatory Commission (NRC) in 1979 at Maxey Flats, Kentucky, with objectives of determining:

- Radiochemical and chemical compositions of leachates in the burial trenches;
- Areal distributions of radionuclides on the site and the factors responsible for this distribution;
- Concentrations of radionuclides in vegetation both on and offsite and the uptake of radionuclides by typical agricultural crops;
- Atmospheric pathways for radionuclide transport and the mechanisms involved;
- Subsurface migration rates of radionuclides and the chemical, physical, biological, and hydrogeological factors which affect this migration;
- Engineering practices which influence the seepage of surface waters into the burial trenches.

Pacific Northwest Laboratory (PNL) has coordinated and conducted research at Maxey Flats to establish and carry out a unified and comprehensive study of the problems specific to that site. In a broader context, the PNL research project will develop an understanding of the processes responsible for radionuclide migration at other commercial shallow land burial sites as well. Chemical measurements of waste trench leachate and identification of chemical changes in leachate during migration will provide a basis for geochemical waste transport models.

Field measurements at shallow land burial sites provide unique opportunities to identify in actual waste burial situations the chemical and geochemical parameters which retard or enhance radionuclide mobility at shallow land burial sites. The work at Maxey Flats also responds to the Office of State Programs' request for information on the areal distribution of radionuclides and the extent of subterranean transport of radionuclides at that site. This information is needed by the Commonwealth of Kentucky to develop plans to stabilize, close and monitor the Maxey Flats site.

Specific objectives of this PNL project with respect to measurements to be made at existing sites are:

- Define the actual chemical species of the radionuclides which are migrating by subsurface routes;
- Identify the chemical characteristics of the trench leachate which retard or enhance radionuclide mobility;
- Determine the chemical changes in leachate properties which result from installation of infiltration barriers such as the flexible vinyl covers at Maxey Flats;
- Provide the chemical and radiochemical input parameters essential for hydrological/geochemical modeling of subsurface radionuclide transport;
- Determine the areal distributions of surface radionuclides, the factors responsible for this distribution, and procedures for monitoring.

## EXPERIMENTAL METHODS

### EXPERIMENTAL SLIT TRENCH AND INERT ATMOSPHERE WELLS

An experimental slit trench was constructed adjacent to an existing waste-filled trench at Maxey Flats in 1979. The experimental trench was designed to accommodate a variety of experiments by the laboratories participating in research at Maxey Flats. Sumps were installed to permit sampling for radionuclide and non-radioactive tracer studies of saturated flow. Porous cups and soil moisture cells were installed in the trench wall, cap and adjacent area for the study of unsaturated flow and moisture gradients near the experimental trench. Nonradioactive tracer studies were included to monitor groundwater movement to and from the experimental trench and to evaluate the permeability of the trench caps to water movement and to root penetration. Several small vegetation plots were established to evaluate potential cover crops, and engineering studies were included to evaluate several different trench caps and drainage systems. Details of trench construc-



tion are described in an earlier report(4) and are summarized in Table 1.

In a second and related construction effort in 1980, inert atmosphere sampling wells were installed in undisturbed ground adjacent to the experimental trench to study subsurface water that would not be influenced by the trench construction materials. The well caps were sealed, permanent sampling and gas lines installed, and the wells purged with argon. During and between sampling activities, the wells are maintained under an argon atmosphere to permit chemical separations to be made on water samples from the wells without the adverse effects of traces of oxygen on the chemical equilibria. The well design permits the determination of chemical species of dissolved radionuclides with minimal disturbance of the chemical balance between soil and groundwater.

Fourteen inert atmosphere wells were installed as part of an effort which also included the installation of sampling stations for unsaturated flow and soil solution studies near the experimental trench and in an undisturbed region of the Maxey Flats site between two old waste trench caps. Figure 1 shows the locations of these experimental installations relative to the existing waste-filled trenches. The placement of the inert atmosphere wells relative to the experimental trench and to waste trench 27 is presented in greater detail in figure 2. Details of sampling station placement for the study of unsaturated flow and moisture gradients by other laboratories are given in another report(5).

Since their installation, up to eight of the wells have been productive sampling stations, but some wells provide water only on an intermittent basis. Several of the wells, mostly on the western end of the experimental trench near sections 4 and 5, have remained dry since installation. After sampling, the water samples are shipped to PNL for analysis for  $^3\text{H}$ ,  $^{60}\text{Co}$ ,  $^{90}\text{Sr}$ ,  $^{137}\text{Cs}$ ,  $^{238}\text{U}$ ,  $^{239}\text{Pu}$ ,  $^{240}\text{Pu}$ ,  $^{241}\text{Am}$  and organic ligands. Tritium analyses are performed by liquid scintillation spectrometry and  $^{90}\text{Sr}$  is chemically separated and counted using a beta proportional counter. Gamma-emitting radionuclides are analyzed by gamma-ray spectrometry using a Ge(Li) detector. Plutonium and americium are determined using solid state detectors and alpha energy analysis. Nonradioactive species are determined by neutron activation analysis using the PNL subcritical neutron multiplier(6), and by chemical and instrumental analysis.

## IN SITU GAMMA-RAY SPECTROMETRY

The in situ gamma-ray spectrometry system is similar to systems used in previous research by Pacific Northwest Laboratory(7,8,9). It consists of a Princeton Gamma-Tech coaxial intrinsic germanium diode coupled to a Canberra Series 80 multichannel analyzer and a cassette data recorder. A portable 1.5 kW Honda generator supplies the electrical power. The detector is attached to a tripod stand and can be

Table 1  
Maxey Flats Experimental Trench Summary

|                                       | Trench Section                                          |                                                                      |                                    |                                   |                           |
|---------------------------------------|---------------------------------------------------------|----------------------------------------------------------------------|------------------------------------|-----------------------------------|---------------------------|
|                                       | T-1                                                     | T-2                                                                  | T-3                                | T-4                               | T-5                       |
| Gamma-Ray Logging                     | Partial                                                 | Yes                                                                  | -                                  | Yes                               | -                         |
| Geologic Logging*                     | -                                                       | Partial                                                              | -                                  | -                                 | -                         |
| Water In-Flow                         | Yes                                                     | Yes                                                                  | Yes                                | -                                 | -                         |
| Sumps                                 | 2                                                       | 2                                                                    | 2                                  | 1                                 | 1                         |
| Porous Cups<br>in Trench Wall         | -                                                       | -                                                                    | -                                  | Three Vertical<br>Traverses + Cap | -                         |
| Soil Moisture Cells<br>in Trench Wall | -                                                       | -                                                                    | -                                  | One Vertical<br>Traverse + Cap    | -                         |
| Cap Design                            | Local Soil/<br>Clay                                     | Local Soil/<br>Clay                                                  | Local Soil/<br>Mirafi-140/<br>Clay | Local Soil/<br>Clay               | Clay/<br>Hypalon/<br>Clay |
| Groundwater Tracer                    | NaBr                                                    | NaBr                                                                 | NaBr                               | -                                 | NaBr                      |
| Cap Tracers                           | Na Benzoate                                             | o-Fluoro-<br>benzoic Acid                                            | Na Benzoate                        | Pentafluoro-<br>benzoic Acid      | Na Benzoate               |
| Root Penetration<br>Tracers           | BaCl <sub>2</sub><br>NiSO <sub>4</sub>                  | Sr(NO <sub>3</sub> ) <sub>2</sub><br>Li <sub>2</sub> CO <sub>3</sub> |                                    |                                   |                           |
| Vegetation Plots                      | Yes                                                     | Yes                                                                  |                                    |                                   |                           |
| Drainage Ditches                      | Drain Trench Cap for Tracer Studies and Erosion Control |                                                                      |                                    |                                   |                           |

\*Limited by safety considerations.

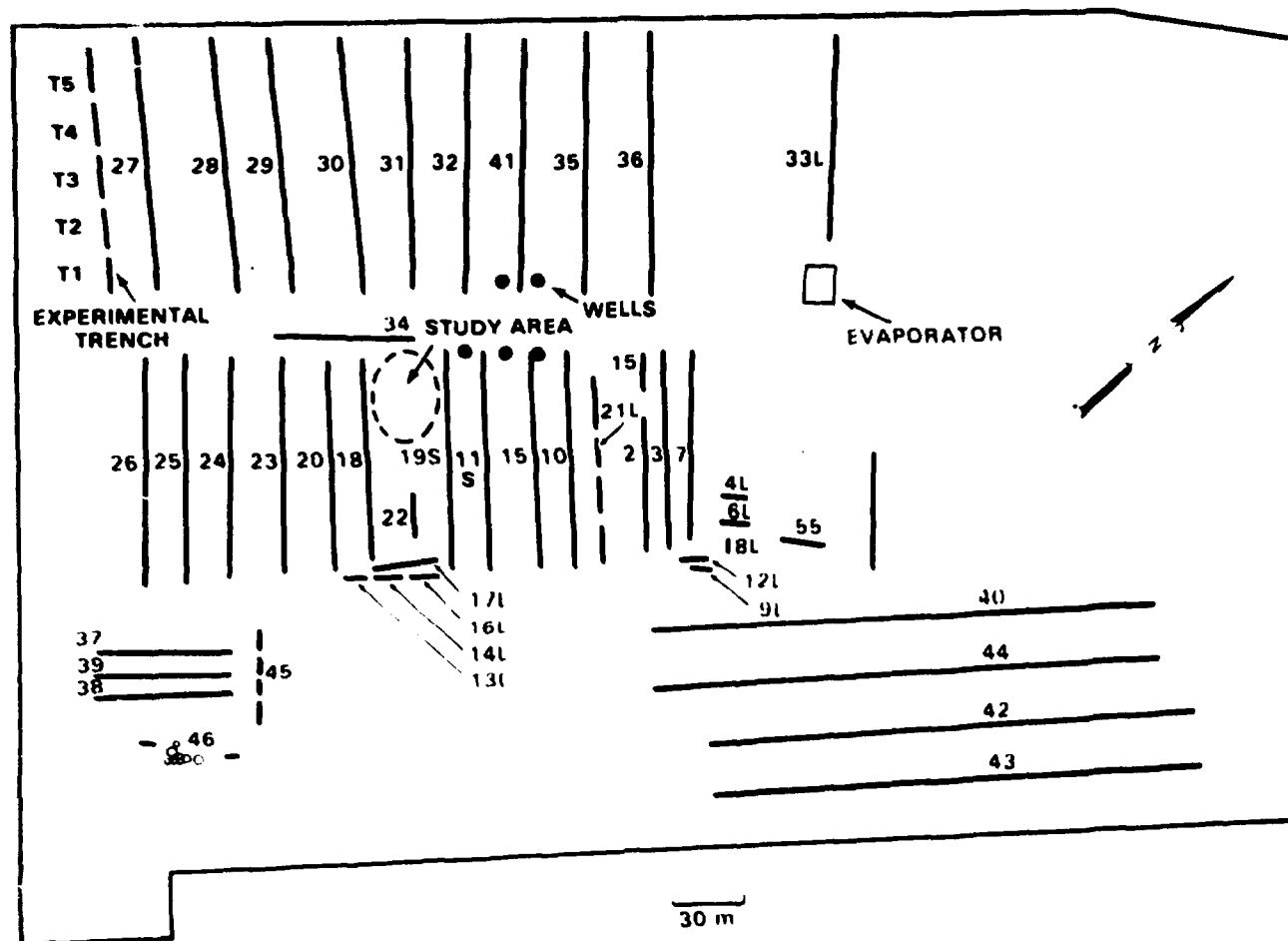


Figure 1. Maxey Flats Waste Trenches and Experimental Areas. The numbered lines show the approximate locations of the waste trenches. The experimental trench area (upper left-hand corner) is shown in greater detail in Figure 2. The study area by waste trench 19S is described in greater detail in Section V of reference 5. These experimental areas are used by several laboratories in research studies at the Maxey Flats shallow land burial site.

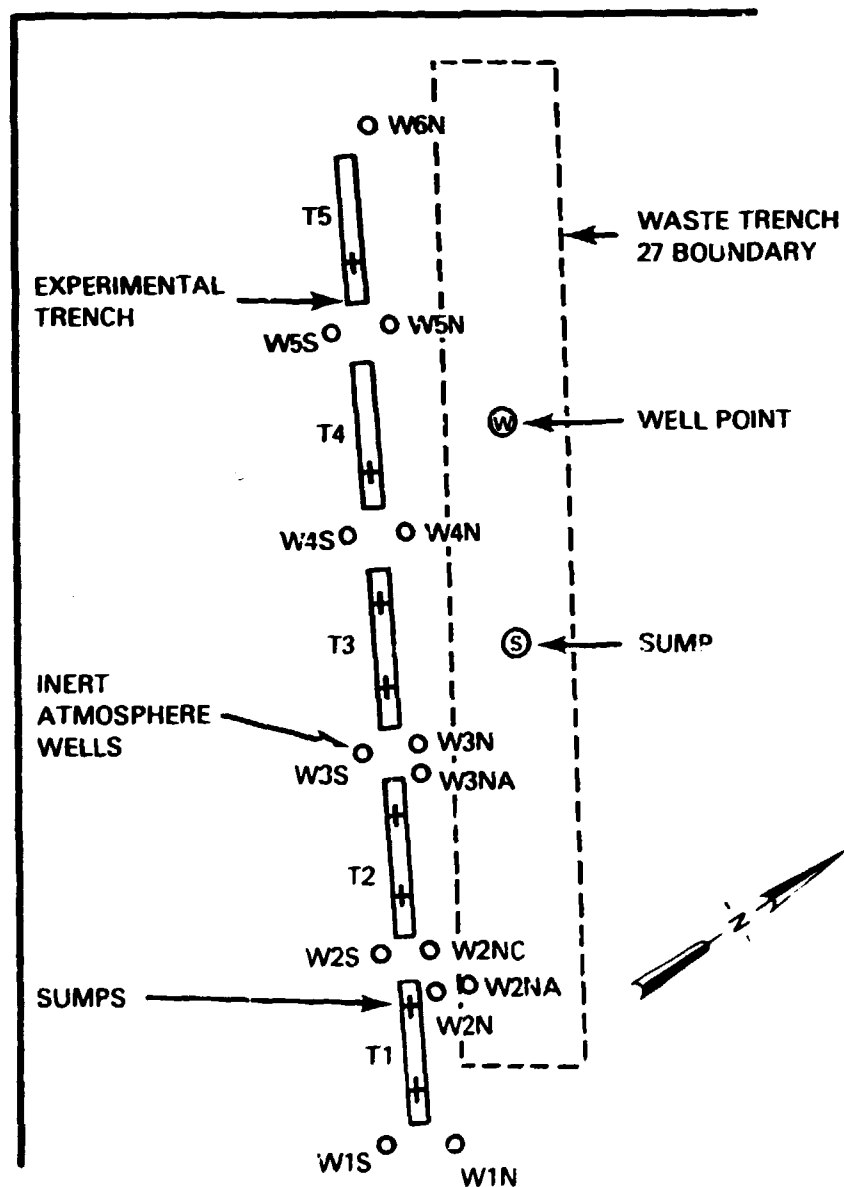


Figure 2. Maxey Flats Experimental Trench and Wells near Waste Trench 27. These facilities allow concurrent sampling of the experimental trench, inert atmosphere wells and adjoining waste trench for groundwater migration and chemical species studies. The experimental trench is used by several laboratories in research studies at the Maxey Flats shallow land burial site.

positioned from ground level to a height of 1 meter. All field measurements at the Maxey Flats site were made with the detector 1 meter above ground level. Signal and power cables 30 meters long connect the detector to the analyzer, which is transported in a vehicle. The liquid nitrogen reservoir for the detector system has sufficient capacity to permit at least 12 hours of continuous operation.

For the measurements in the drainages adjacent to the restricted area, a portable backpack gamma-ray spectrometer system was used. The system consists of a Princeton Gamma-Tech coaxial intrinsic germanium diode coupled to a Davidson Model 4106A multichannel analyzer. Data acquisition is completed by memory transfer to a magnetic tape with a portable cassette recorder. This electronic package, together with a 140 volt d.c. Globe gel/cell battery pack, is mounted on a specially adapted backpack frame and the detector is suspended from a modified photographer's tripod.

Calibration of the counting systems was made by comparing the disintegration rates of the naturally-occurring radionuclides in the soil with the count rates for the same radionuclides at each in situ counting location. The calibration factors so derived depend on the assumption that all the radionuclides are uniformly dispersed throughout the soil. By plotting the ratios of the in situ count rates to soil disintegration rates versus gamma-ray energy, calibration factors were derived for  $^{137}\text{Cs}$  and  $^{60}\text{Co}$ . All data reduction was accomplished in the laboratory using a PDP 11/44 computer and software developed for these analyses. The radionuclides measured include the naturally-occurring radionuclides  $^4\text{K}$ ,  $^{208}\text{Tl}$ ,  $^{212}\text{Pb}$ ,  $^{214}\text{Pb}$ ,  $^{212}\text{Bi}$ ,  $^{214}\text{Bi}$ , and  $^{228}\text{Ac}$ , as well as  $^{60}\text{Co}$  and  $^{137}\text{Cs}$  and other exogenous radionuclides. Thorium-228 is determined from its daughter concentrations.

Soil samples were collected from each location where a count was made and returned to the laboratory for analysis. Samples were taken at the site surface at counting locations within the restricted area. Two soil samples were taken wherever possible at each counting location in the main drainage channels leading from the site, one at the surface and a second sample 15 cm below the surface. Analysis of the soil samples for up to twenty-four gamma-emitting radionuclides (e.g.,  $^4\text{K}$ ,  $^{208}\text{Tl}$ ,  $^{212}\text{Pb}$ ,  $^{212}\text{Bi}$ ,  $^{214}\text{Bi}$ ,  $^{228}\text{Ac}$ ,  $^{228}\text{Th}$ ,  $^{60}\text{Co}$  and  $^{137}\text{Cs}$ ) was performed with a Ge(Li) coincidence/anticoincidence counting system at PNL facilities in Richland, Washington. Destructive analysis is continuing to determine the levels of  $^{90}\text{Sr}$ ,  $^{238}\text{Pu}$ ,  $^{239}\text{Pu}$ ,  $^{240}\text{Pu}$  and  $^{241}\text{Am}$  in selected soil samples using laboratory procedures similar to those developed for studies conducted for the Electric Power Research Institute (10).

## ORGANIC LIGAND ANALYSIS

Organic analyses were performed on unfractionated water samples (survey study) and on fractionated water samples following steric exclusion chromatography (see figure 3).

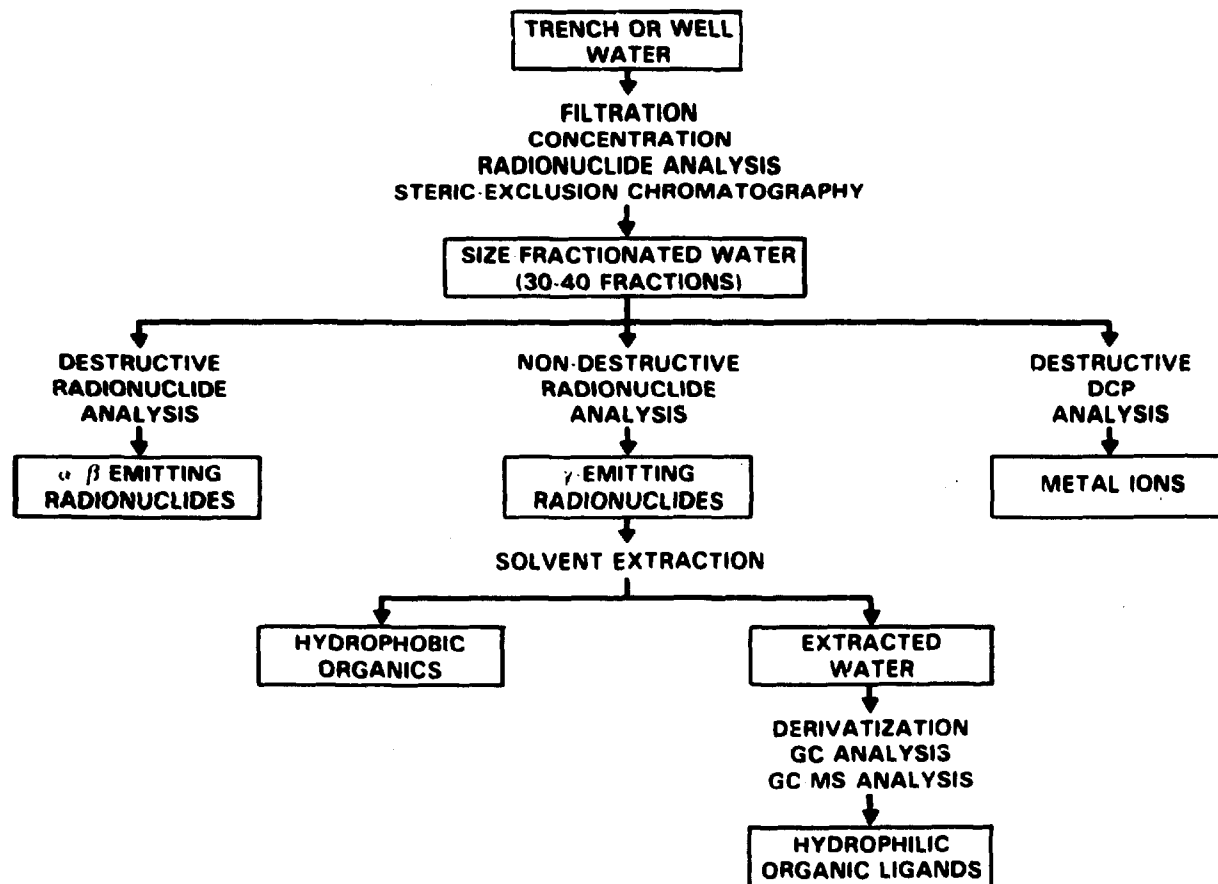


Figure 3. Trench and Well Water Analysis Scheme.

Fifty ml each of the unfractionated water samples were concentrated ten-fold by rotary evaporation prior to organic derivation. The water samples set aside for chromatographic fractionation were also concentrated ten-fold and the pH adjusted to that of the original sample ( $\sim$  pH 7.2 - pH 7.6). The solution was filtered, loaded on a Sephadex G-15 column, and chromatographed with redistilled water. Replicate runs were made, with and without Blue Dextran as a void volume marker. The column effluent was monitored by ultraviolet absorbance at 254 nm and collected using a fraction collector. Each column fraction was acidified with hydrochloric acid and divided into equal portions, to be used for destructive radionuclide analysis and for nondestructive analysis of gamma-emitting radionuclides and organic compounds.

The aliquot of the Sephadex G-15 fraction remaining after gamma-ray analysis was extracted with chloroform to remove hydrophobic organic compounds and this extract set aside for analysis by gas chromatography (GC) and combined GC-mass spectrometry (GC-MS). The aqueous fraction containing hydrophilic organic compounds was evaporated to dryness under nitrogen. The residue of each fraction was then methylated with  $\text{BF}_3$ /methanol at  $100^\circ\text{C}$  in a sealed vial. After cooling, chloroform was added, the mixture was transferred to a buffer solution (pH 7) and vortexed. Part of the chloroform layer, which contained the methylated hydrophilic organics, was then evaporated under nitrogen. The residue was redissolved in chloroform and an aliquot analyzed by GC and by GC-MS.

GC analyses were performed on a Hewlett-Packard 5880 gas chromatograph with an SE 52 column. From an initial value of  $40^\circ\text{C}$ , the column temperature was programmed at  $20^\circ\text{C}$  per minute for three minutes to  $100^\circ\text{C}$  followed by  $8^\circ\text{C}$  per minute for 25 minutes to  $300^\circ\text{C}$ , and finally maintained isothermally at  $300^\circ\text{C}$  for 10 minutes. Tetramethyl-EDTA was used as an internal and external GC standard to quantify the various organic constituents in the GC samples.

GC-MS analyses were performed on a Hewlett Packard 5985 GC-MS instrument in the electron-impact (70-eV) mode. The gas chromatograph on the 5985 instrument was equipped with a 60-m x 0.25-mm I.D. fused silica capillary column coated with 0.25- $\mu\text{m}$  of SE-54; the column was programmed from  $40^\circ\text{C}$  to  $300^\circ\text{C}$  at  $5^\circ\text{C}/\text{min}$ , where it was maintained isothermally for 8 min. A splitless injection system was used to introduce the sample onto the GC-MS instrument. The GC column was interfaced directly to the mass spectrometer. A mass range of 50 to 400 amu was scanned every 1.0 sec by computer (HP-2100MX equipped with the HP-7920 Large Disc Drive).

## RESULTS AND DISCUSSION

## EXPERIMENTAL SLIT TRENCH AND INERT ATMOSPHERE WELLS

The decision to construct an experimental trench was based on the premise that this design offered the best opportunity to intercept subsurface flow from waste trenches, either through fracture flow at sandstone-shale interfaces, or through direct flow. Indeed, water that may not have been intercepted by wells flowed into three sections of the experimental trench during construction. The initial radioactive concentrations were comparable to waste trench 27 leachate, as shown in Table 2. The radionuclide concentrations measured in these construction samples clearly show that subsurface movement of radionuclides has occurred at Maxey Flats, but the distance of migration cannot be inferred from these data.

When the experimental trench was resampled on later dates, most radionuclide concentrations showed marked decreases, suggesting that the radionuclides were present initially only in the experimental trench location due to a decrease of hydraulic pressure during construction that permitted flow to occur from the waste trench into the experimental trench area. However, it is also possible that the trench design had adversely influenced the concentrations of radionuclides in water flowing toward the trench sumps through changes in the water chemistry. The exposure of the open trench to air during construction resulted in the formation of an in-place scavenger (ferric hydroxide), which reacted synergistically with the crushed rock backfill to form an absorptive barrier for radionuclide transport. Consequently, the decision was made to install monitoring wells in undisturbed soil near the experimental trench to collect anoxic samples for chemical species determinations.

Water flowed into six of the inert atmosphere sampling wells during the post-construction period. Radionuclide and bromine concentrations from these wells are compared with waste trench leachate in Table 3. Changes in bromine concentrations that have been observed in waste trench 27 leachate since 1978 and in water from the experimental trench and inert atmosphere wells indicate that there is some communication by groundwater flow between these locations. These comparisons are made possible in part by the presence of sodium bromide (NaBr) tracer added to sections 1, 2, 3 and 5 of the experimental trench during its construction in 1979. Bromine concentrations in waste trench 27 leachate were quite low prior to the construction of the experimental trench. Following trench construction, the bromine concentration in waste trench 27 increased from 2.8 to 460 ppm, indicating that some of the NaBr tracer had moved into waste trench 27. The well point samples had low concentrations of bromine, probably because the well point was installed opposite section 4 of the experimental trench, which contains no added water. There seems to be no direct communication by groundwater flow between these two points.



Table 2

Comparison of Radionuclide Concentrations in Waste  
Trench Leachate (5-78) and Initial Experimental  
Trench Inflow (9-79)

|                          | Concentration, pCi/l |          |                  |                  |                   |
|--------------------------|----------------------|----------|------------------|------------------|-------------------|
|                          | <sup>3</sup> H       | Total Pu | <sup>90</sup> Sr | <sup>60</sup> Co | <sup>137</sup> Cs |
| Waste Trench 27 Leachate | 5.9+08               | 4.8+03   | 2.1+05           | 1.3+03           | 8.0+03            |
| Experimental Trench      |                      |          |                  |                  |                   |
| Section T1               | 3.7+08               | 3.6+01   | 6.7+03           | 3.6+01           | 6.5+01            |
| Section T2E              | 1.7+09               | 6.1+01   | 1.7+05           | 2.2+02           | 1.6+03            |
| Section T2W              | 4.2+09               | 3.4+03   | 1.6+05           | 9.7+03           | 4.6+03            |
| Section T3E              | 3.4+09               | 5.0+03   | 1.1+05           | 6.0+03           | 9.0+01            |

Table 3

Comparison of Radionuclide and Bromine Concentrations in Waste Trench Leachate (5-78 and 9-80) and Inert Atmosphere Wells (9-80)

|                                      | Radionuclide Concentration (pCi/l) |                   |                       |                  | Bromine<br>ppm |
|--------------------------------------|------------------------------------|-------------------|-----------------------|------------------|----------------|
|                                      | $^3\text{H}$                       | $^{238}\text{Pu}$ | $^{239,240}\text{Pu}$ | $^{90}\text{Sr}$ |                |
| Waste Trench 27<br>Sump (5-78)       | 5.9+08                             | 4.1+03            | 6.7+02                | 2.1+05           | 2.8+00         |
| Waste Trench 27<br>Sump (9-80)       |                                    |                   |                       |                  | 4.6+02         |
| Waste Trench 27<br>Well Point (9-80) |                                    |                   |                       |                  | 2.7+00         |
| Well W1N                             | 4.7+05                             | 3.7+01            | 6.0+00                | 1.5+02           | 1.0+00         |
| Well W2N                             | 2.9+06                             | 2.1+01            | 7.7+01                | 3.3+02           | 2.0+00         |
| Well W2NA                            | 7.7+08                             | 2.4+02            | 4.8+00                | 1.7+04           | 1.2+02         |
| Well W3N                             | 1.2+09                             | 2.4+03            | 4.4+01                | 3.4+04           | 8.2+02         |
| Well W3NA                            | 2.0+07                             | 2.3+01            | 5.0+00                | 2.4+02           | 5.2+02         |
| Well W5N                             | 1.0+03                             | 9.4+00            | 5.0+00                | 2.1+02           | 5.0+00         |

Bromine concentrations in the inert atmosphere wells could also be influenced by the presence of NaBr in sections 1, 2 and 3 of the experimental trench. Wells W3N and W3NA show this influence even more markedly than the comparison with waste trench 27, since the wells are much closer to the NaBr source. Well W2NA also indicates a communication with the NaBr source while wells W1N, W2N and W5N seem not to be connected to the NaBr source. When radionuclide concentrations are compared, only W2NA and W3N appear to be closely related to waste trench 27, possibly connecting to it through subsurface fractures.

Because of the presence of tritium in waste trench 27 leachate, a useful tracer is in place for following groundwater movement in that region. Tritium concentrations in water from the experimental trench are summarized in Figure 5. The tritium concentrations in samples from trench sections T2 and T3 have shown very marked decreases since the experimental trench was constructed, indicating that there is substantial movement of water away from the trench in this region. The tritium concentration in waste trench 27 leachate has been relatively stable during the same time; the waste trench leachate apparently did not flow toward the experimental trench to maintain the tritium concentrations near the levels that were originally observed in sections T2 and T3. It is unlikely that this degree of dilution could have come from water filtering through the trench caps, since only one of the nonradioactive tracers installed in these caps was detected in trace quantity in section 4 of the experimental trench. The tritium data, as do the bromine data, show that some water flow has occurred away from the experimental trench and toward waste trench 27 during the study period. Water from sections T2 and T3 would therefore be diluted by subsurface lateral flow of low-tritium water moving from the south toward the experimental trench.

Plutonium in waste trench 27 leachate is present in the reduced form as a strongly complexed anionic species<sup>(4)</sup>. To obtain evidence on the chemical behavior of plutonium complexes with complexing agents likely to be present in Maxey Flats trench waters, laboratory ion exchange experiments were performed with plutonium (+3) and plutonium (+4) and several candidate complexing agents (EDTA, TBP, TOA). Among the solutions studied,  $10^{-6}$  M EDTA had the greatest effect on plutonium absorption. (The EDTA concentration having the greatest effect on plutonium absorption in these experiments is about the concentration measured in waste trench 27 leachate.) The presence of a strong complexing agent like EDTA, even at these low concentrations, may actually be more important than the oxidation state of plutonium in determining whether it absorbs on a material with a high exchange capacity. These experiments support the view that plutonium is present in waste trench leachate as a strong complex, probably with EDTA.<sup>(5)</sup>

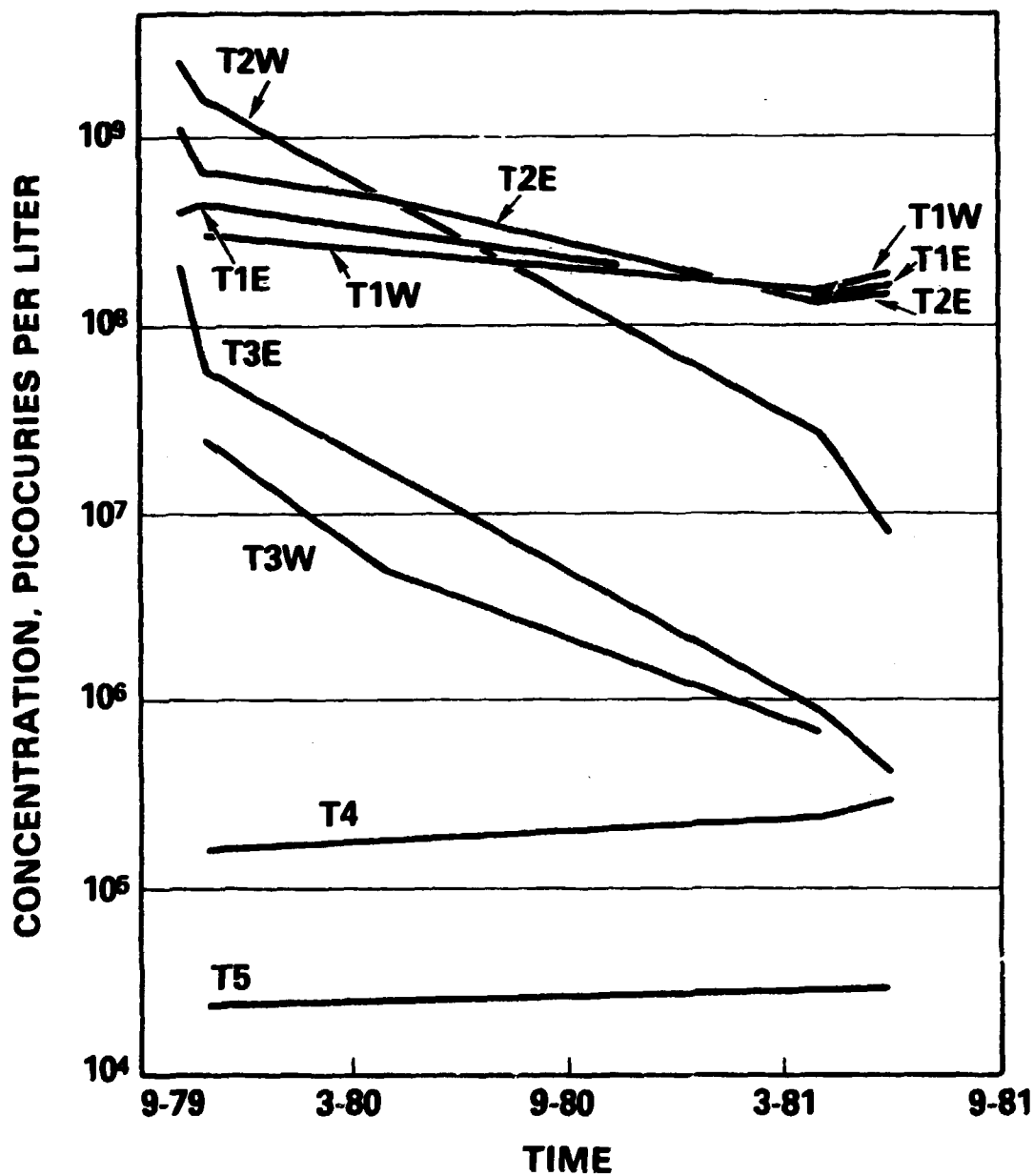


Figure 4. Tritium concentrations in the experimental trench during September 1979 through June 1981. Major changes in tritium concentrations in trench sections T2W, T3E and T5W indicate that groundwater flow during this time was generally away from the experimental trench and toward the waste burial area.

## AREAL DISTRIBUTIONS OF RADIONUCLIDES

Two general kinds of radionuclides were determined in these studies. They are endogenous (natural) radionuclides originating from the parent soil material and exogenous (man-induced) radionuclides originating from the nuclear fuel process. Cobalt-60 and  $^{137}\text{Cs}$  are exogenous radionuclides and both are constituents of global fallout as well as being components of low-level radioactive wastes stored at Maxey Flats. Potassium-40 and  $^{228}\text{Th}$  are endogenous radionuclides originating from the underlying rock strata and soil. In a typical gamma-ray spectrum, most of the peaks are due to endogenous radionuclides, chiefly natural uranium, thorium, and their daughters, plus  $^4\text{K}$ .

Very low concentrations of radionuclides were present in most soil samples. Cesium-137 and  $^{60}\text{Co}$  were the most abundant radionuclides of exogenous origin and these have their origin in global fallout, which is ubiquitously distributed over all land surfaces. The distribution of  $^{60}\text{Co}$  in surface soils within and adjacent to the restricted area at Maxey Flats in May, 1980 is plotted in Figure 5. The data have been separated into three levels of  $^{60}\text{Co}$  soil concentrations -- greater than 2 pCi  $^{60}\text{Co}$  per gram of soil, 1-2 pCi  $^{60}\text{Co}$  per gram of soil, and less than 1 pCi  $^{60}\text{Co}$  per gram of soil. A similar plot for  $^{137}\text{Cs}$  distribution in surface soils within and adjacent to the restricted area at Maxey Flats is shown in Figure 6.

In general, a distribution for any one radionuclide may not be used to predict a distribution pattern for another radionuclide, except in those instances relating to the purely endogenous radionuclides originating from the parent soil material. Even these latter distributions may be skewed by the presence of specific processing wastes from thorium or uranium purification processes because of the introduction of nonequilibrium quantities of their daughters from the decay chains.

The soil in the west drainage channel leading from the restricted area contains surface concentrations of  $^{60}\text{Co}$  that exceed the background fallout levels away from the site. Figure 7 shows an expanded view of the top 100 meters of this drainage channel with isopleths indicating the concentration ranges. All counts made downhill from the coordinates N330, W540 showed the  $^{60}\text{Co}$  concentration to be at the fallout level measured at background locations away from the site (less than 0.1 pCi  $^{60}\text{Co}$  per gram of soil). Virtually all of the  $^{60}\text{Co}$  is contained in the first few centimeters of soil. In the south and east drainage channels  $^{60}\text{Co}$  concentrations within the drainages are at the background levels.

Cesium-137 concentrations in surface soil within the restricted area and in the main drainage channels leading from the site are similar to concentrations found in other locations of the United States that have rainfall patterns like those of eastern Kentucky. Although it is apparent that surface runoff from the site could contribute  $^{137}\text{Cs}$  to the adjacent drainages, in situ measurements and soil sample analyses indicate that  $^{137}\text{Cs}$  is not derived from the Maxey Flats site.

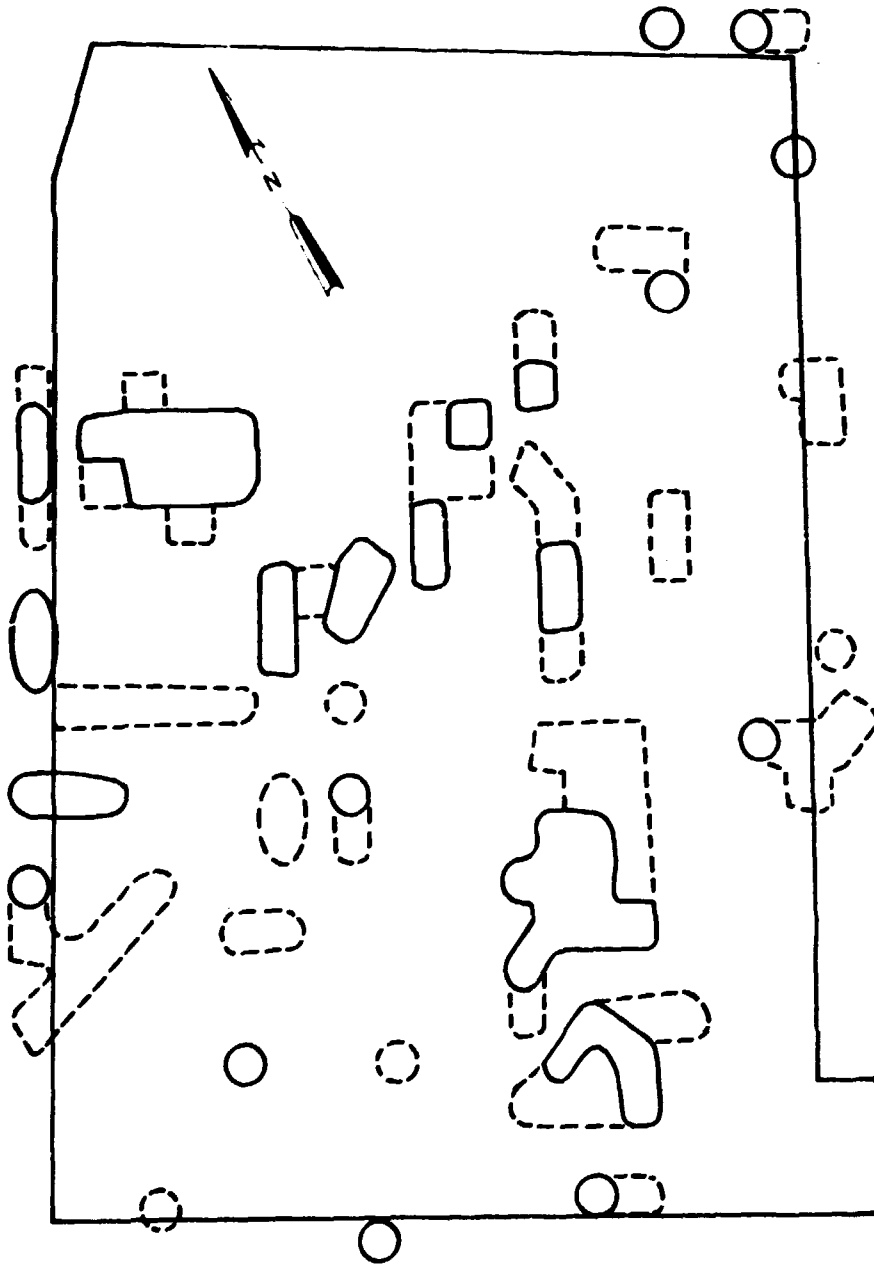


Figure 5. Cobalt-60 distribution in surface soil within and adjacent to the restricted area at Maxey Flats (May, 1980). Areas within the solid isopleths have  $^{60}\text{Co}$  concentrations  $> 2$  pCi per gram of soil. Areas within the dotted isopleths have  $^{60}\text{Co}$  concentrations 1-2 pCi per gram of soil. All other areas on site have  $^{60}\text{Co}$  concentrations  $< 1$  pCi per gram of soil.

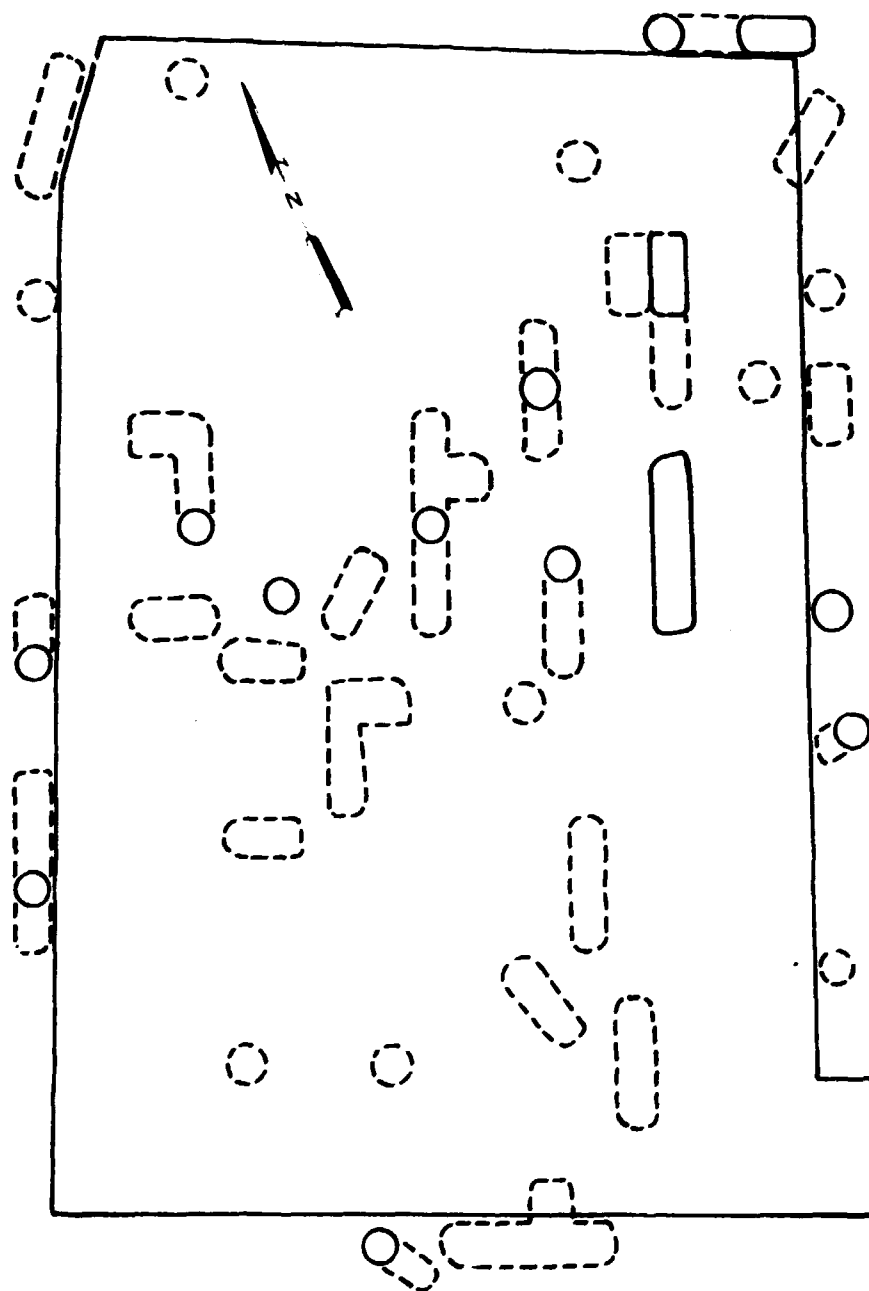


Figure 6. Cesium-137 distribution in surface soil within and adjacent to the restricted area at Maxey Flats (May, 1980). Areas within the solid isopleths have  $^{137}\text{Cs}$  concentrations  $> 2$  pCi per gram of soil. Areas within the dotted isopleths have  $^{137}\text{Cs}$  concentrations 1-2 pCi per gram of soil. All other areas on site have  $^{137}\text{Cs}$  concentrations  $< 1$  pCi per gram of soil.

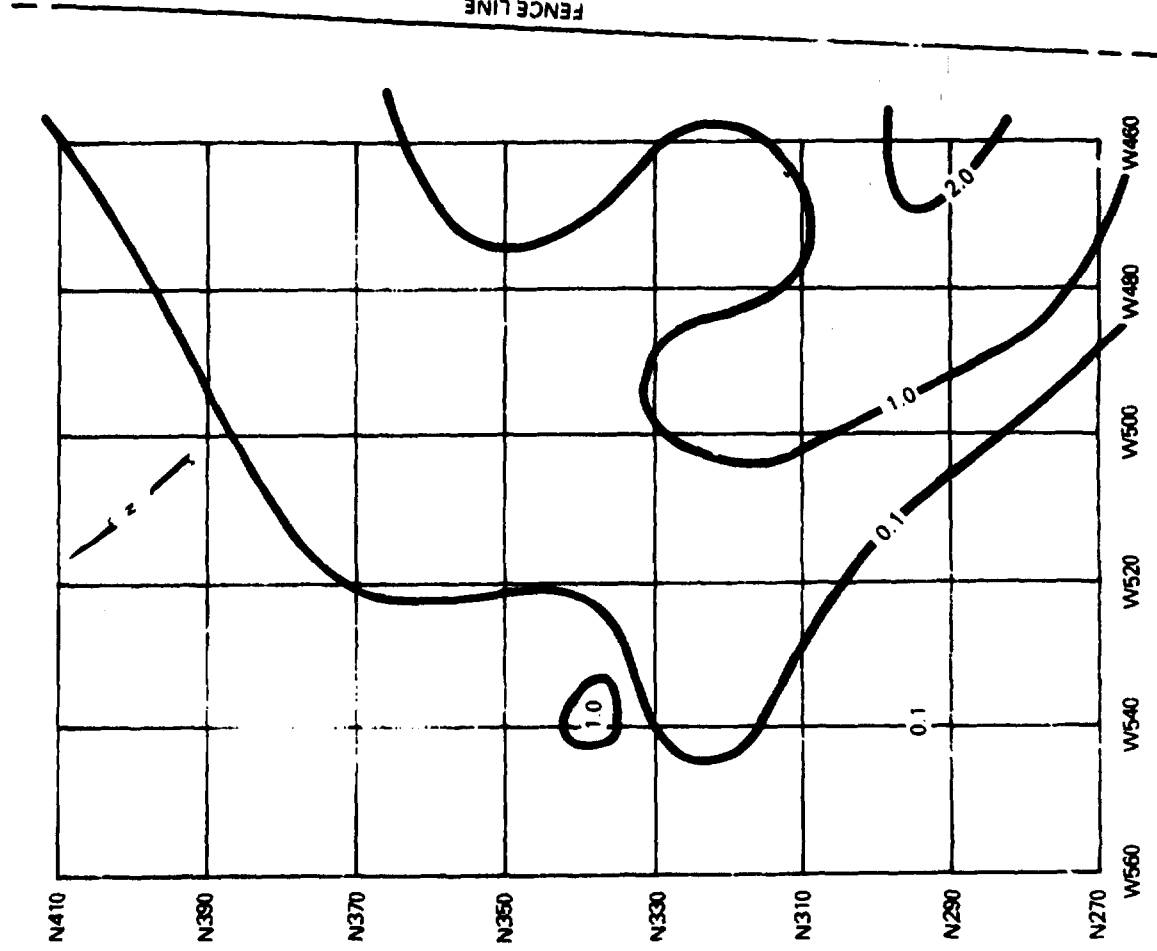


Figure 7. Cobalt-60 distribution in surface soils within the west drainage channel adjacent to the restricted area at Maxey Flats (May, 1981). The main channel follows approximately along the coordinate N330. The heavy isopleths indicate the approximate extent of the  $^{60}\text{Co}$  surface concentrations indicated in pCi per gram of soil. Except for one localized area having  $^{60}\text{Co}$  soil concentrations 1-2 pCi per gram of soil at about N335, W540, the  $^{60}\text{Co}$  concentrations have decreased to background levels about 100 meters from the site boundary. The evaporator location is about 110 meters inside, to the right of the fence line, at the approximate coordinates N360, W330.



Several possible migration pathways exist for radionuclides buried at the Maxey Flats shallow land burial site. To evaluate potential biotic pathways, forest floor litter, newly fallen leaves, newly opened leaves and surface soil were sampled in the forest surrounding the restricted area and analyzed by high resolution gamma-ray spectroscopy. Most of the radionuclides in the forest were associated with the soil, with relatively small amounts of radionuclides associated with the litter and leaves. The endogenous radionuclides  $^{40}\text{K}$  and  $^{228}\text{Th}$  dominated soil distributions, but the exogenous radionuclides  $^{60}\text{Co}$  and  $^{137}\text{Cs}$  were relatively more abundant in the litter layer. The  $^{60}\text{Co}$  content of newly fallen leaves was more variable than the other radionuclides, suggesting that the burial site has influenced the  $^{60}\text{Co}$  concentrations at a few sampling locations and newly opened leaves from one hickory tree suggested that the source of the elevated  $^{60}\text{Co}$  concentrations was the rooting substrate. The other radionuclide concentrations exhibited little variation between tree species from location to location.

## ORGANIC RESEARCH

### SURVEY STUDIES OF ORGANIC LIGANDS

Water samples from waste trench 27 and several inert atmosphere sampling wells (W1N, W2NA, W3N, and W3NA) were analyzed for their hydrophilic and hydrophobic organic contents by the procedure summarized in Figure 3. The hydrophilic organics identified to date are listed in Table 4 and the hydrophobic organics are listed in Table 5.

The chelating agent EDTA is the most abundant of the hydrophilic species, appearing in three samples at ppm levels (see Figure 7). It was identified in the waste trench 27 leachate GC trace (0.99 ppm) and two distinct samples from experimental well W2NA (1.3 ppm and 0.84 ppm, collected on May 18, 1981, and August 17, 1981, respectively). The appearance of EDTA in waste trench 27 and in the vicinity of experimental trench section 2 but not in the vicinity of experimental trench sections 1 and 3 argues that there is a very specific communication between waste trench 27 and the vicinity of experimental well W2NA. The other organic species identified in the hydrophilic organic fraction are generally much less abundant (ppb levels), and consisted mainly of carboxylic acids, potential chelators of radionuclides such as Cs and Sr. A trace of the nonradioactive tracer pentafluorobenzoic acid (added to the cap of experimental trench section 4) was identified in the leachate, suggesting communication between experimental trench section 4 and waste trench 27.

Most of the hydrophobic organics isolated from the water samples were present at ppb levels. The most interesting exception was pentafluorobenzoic acid (1.1 ppm) which appeared in waste trench 27 leachate, confirming its presence in the vicinity of the waste trench and arguing

Table 4

**Hydrophilic Organic Compounds<sup>a</sup> in Inert Atmosphere Wells  
Compared to Waste Trench Leachate**

|                                                   | Concentration (ppb) <sup>b</sup> in Water Samples |                        |       |       |     |      |
|---------------------------------------------------|---------------------------------------------------|------------------------|-------|-------|-----|------|
|                                                   | Waste<br>Trench<br>Leachate                       | Inert Atmosphere Wells |       |       |     |      |
|                                                   | WFT-27                                            | W1N                    | W2NAC | W2NAD | W3N | W3NA |
| Oxalic Acid <sup>e</sup>                          |                                                   | 1                      | 61    | 53    |     | 1    |
| Dimethyl Sulfate                                  |                                                   |                        |       | 16    |     | 1    |
| Propanedioic Acid <sup>e</sup>                    |                                                   |                        | 1     |       |     |      |
| 4-oxo-pentanoic Acid <sup>e</sup>                 | 12                                                | 1                      | 38    | 82    | 3   | 2    |
| Butanedioic Acid <sup>e</sup>                     |                                                   |                        | 20    | 14    |     |      |
| Methyl Succinic Acid <sup>e</sup>                 |                                                   |                        | 7     |       |     | 0.4  |
| Methyl Benzoate <sup>e</sup>                      |                                                   | 1                      | 78    |       |     | 4    |
| Nonanal                                           |                                                   | 3                      |       |       |     |      |
| Hexanedioic Acid <sup>e</sup>                     | 26                                                |                        | 9     | 8     | 43  | 8    |
| Phenoxyacetic Acid <sup>e</sup>                   |                                                   |                        | 147   |       |     |      |
| Octanedioic Acid <sup>e</sup>                     |                                                   |                        |       | 4     |     |      |
| Dimethyl Phthalate                                | 92                                                | 2                      | 115   | 11    | 1   | 2    |
| Citric Acid <sup>e</sup>                          | 8                                                 |                        |       |       |     | 1    |
| Nonanedioic Acid <sup>e</sup>                     |                                                   |                        | 17    | 33    | 1   |      |
| Tetradecanoic Acid <sup>e</sup>                   |                                                   | 1                      |       |       |     |      |
| Hexadecanoic Acid <sup>e</sup>                    | 18                                                | 4                      | 48    | 12    | 8   | 4    |
| Dibutyl Phthalate                                 | 73                                                | 3                      | 72    | 28    | 6   | 4    |
| Pentafluorobenzoic Acid <sup>e</sup>              | Trace                                             |                        |       |       |     |      |
| Octadecanoic Acid <sup>e</sup>                    | 17                                                | 4                      | 111   | 9     | 6   | 3    |
| Ethylenediaminetetra-<br>acetic Acid <sup>e</sup> | 990                                               |                        | 1330  | 840   |     |      |
| Diethyl Phthalate                                 | 35                                                | 103                    |       |       | 3   |      |
| Silicone Oil Oligomers                            | 67                                                |                        |       |       | 33  | 7    |

a. Methylated, BF<sub>3</sub>/Methanol

b. No entry indicates compound is below detection level

c. Sampled 5/18/81

d. Sampled 8/17/81

e. Methyl Ester

Table 5

**Hydrophobic Organic Compounds in Inert Atmosphere Wells  
Compared to Waste Trench Leachate**

|                                             | Concentration (ppb) <sup>a</sup> in Water Samples |                        |       |       |      |      |
|---------------------------------------------|---------------------------------------------------|------------------------|-------|-------|------|------|
|                                             | Waste<br>Trench<br>Leachate                       | Inert Atmosphere Wells |       |       |      |      |
|                                             | WFT-27                                            | W1N                    | W2NAD | W2NAC | W3N  | W3NA |
| Methyl Butenone                             | 9                                                 |                        | 99    |       | 21   |      |
| Chloromethylbutene                          | 10                                                | 20                     | 107   | 21    | 57   | 94   |
| Methyl Methacrylate                         | 2                                                 | 9                      |       |       |      |      |
| Pyridine                                    |                                                   | 97                     |       |       |      |      |
| Toluene                                     | 23                                                |                        | 64    |       | 92   | 72   |
| Dichloromethylbutane                        | 2                                                 | 7                      | 38    | 3     | 21   | 36   |
| Dichloriodomethane                          | 4                                                 |                        | 6     |       |      |      |
| Benzotriazole                               | 3                                                 |                        |       |       |      |      |
| Benzofuran                                  | 10                                                |                        | 16    |       | 42   | 37   |
| 1,3-dihydro-2H-<br>Indol-2-one              | 6                                                 |                        |       |       |      |      |
| Undecane                                    | 3                                                 |                        |       |       | 27   | 15   |
| Dodecane                                    | 4                                                 |                        |       |       | 19   | 19   |
| Benzothiazole                               | 13                                                |                        |       |       |      |      |
| Tridecane                                   | 3                                                 |                        |       |       | 17   | 18   |
| Phthalic Anhydride                          | 17                                                |                        |       |       |      |      |
| 2,5-dimethyl Benzene                        | 17                                                |                        |       |       |      |      |
| Butanoic Acid                               |                                                   |                        |       |       |      |      |
| 2(3H)-benzothiazolone                       | 835                                               |                        |       |       |      |      |
| N,N-bis(phenylmethyl)-<br>amine             |                                                   | 3                      |       |       | 4    |      |
| Hexadecanoic acid                           | 23                                                |                        |       |       |      |      |
| Pentafluorobenzoic Acid                     | 1067                                              |                        |       |       |      |      |
| Alkanes (C <sub>20</sub> -C <sub>31</sub> ) | 197                                               |                        |       |       |      | 158  |
| Alkylphenoxy Oligomers                      | 102                                               | 197                    | 963   | Trace | 1144 | 735  |

a. No entry indicates compound is below detection level

b. Sampled 5/18/81

c. Sampled 8/17/81

again for communication between experimental trench section 4 and the waste trench. In general, water samples from the experimental trench area contained only a scattering of hydrophobic organics. Some of the compounds, however (e.g. pyridine), are not typically found in uncontaminated soil samples. In contrast, the leachate from waste trench 27 contained quite a diversity of hydrophobic organics, possibly associated with waste dumped into the trench.

#### DETAILED CHEMICAL SPECIATION STUDIES

The second major research effort supports a detailed chemical speciation study aimed at determining whether the organic compounds identified in the survey study are complexed to radionuclides. After steric exclusion fractionation of the waste trench 27 leachate, several peaks appeared in the UV chromatogram, corresponding to column-included organic-containing species. Plutonium eluted in one region of the UV chromatogram, just prior to  $^{90}\text{Sr}$  and  $^{137}\text{Cs}$ . EDTA co-eluted with plutonium, suggesting that EDTA is complexed with plutonium in the waste trench leachate. A variety of organic acids and other hydrophilic compounds co-eluted with the  $^{90}\text{Sr}$  and  $^{137}\text{Cs}$ . It thus appears that chelation of radionuclides by polar organic species such as organic acids may provide a mechanism for the migration of radionuclides at the commercial shallow land burial site.

#### CONCLUSIONS

EDTA is the major organic component in waste trench 27 leachate and an inert atmosphere well (0.84-1.33 ppm). Other hydrophilic organic compounds, including carboxylic acids, were identified at ppb levels. Hydrophobic organic compounds were also identified in waste trench leachate at ppb levels. Water samples were fractionated by steric exclusion chromatography prior to organic and radiochemical analysis to determine which of the organic species and radionuclides are complexed. Plutonium co-eluted with EDTA, whereas  $^{137}\text{Cs}$  and  $^{90}\text{Sr}$  co-eluted with a variety of organic acids and other hydrophilic compounds. Observations strongly suggest that the chemical species of the radionuclides can mediate their environmental dispersion. The correlations offer strong evidence to support the conclusion that plutonium probably migrates as an EDTA complex, and that polar organic compounds may influence the migration of  $^{90}\text{Sr}$  and  $^{137}\text{Cs}$ .

Nonradioactive tracers (sodium bromide and pentafluorobenzoic acid) added to sections of the experimental trench were detected in waste trench leachate, suggesting migration from the experimental study area to the waste-filled area. The tritium and inert tracer concentrations in water from the experimental trench area confirm that subsurface flow has occurred from the experimental trench and toward the waste burial trenches during the time period of the measurements. Major

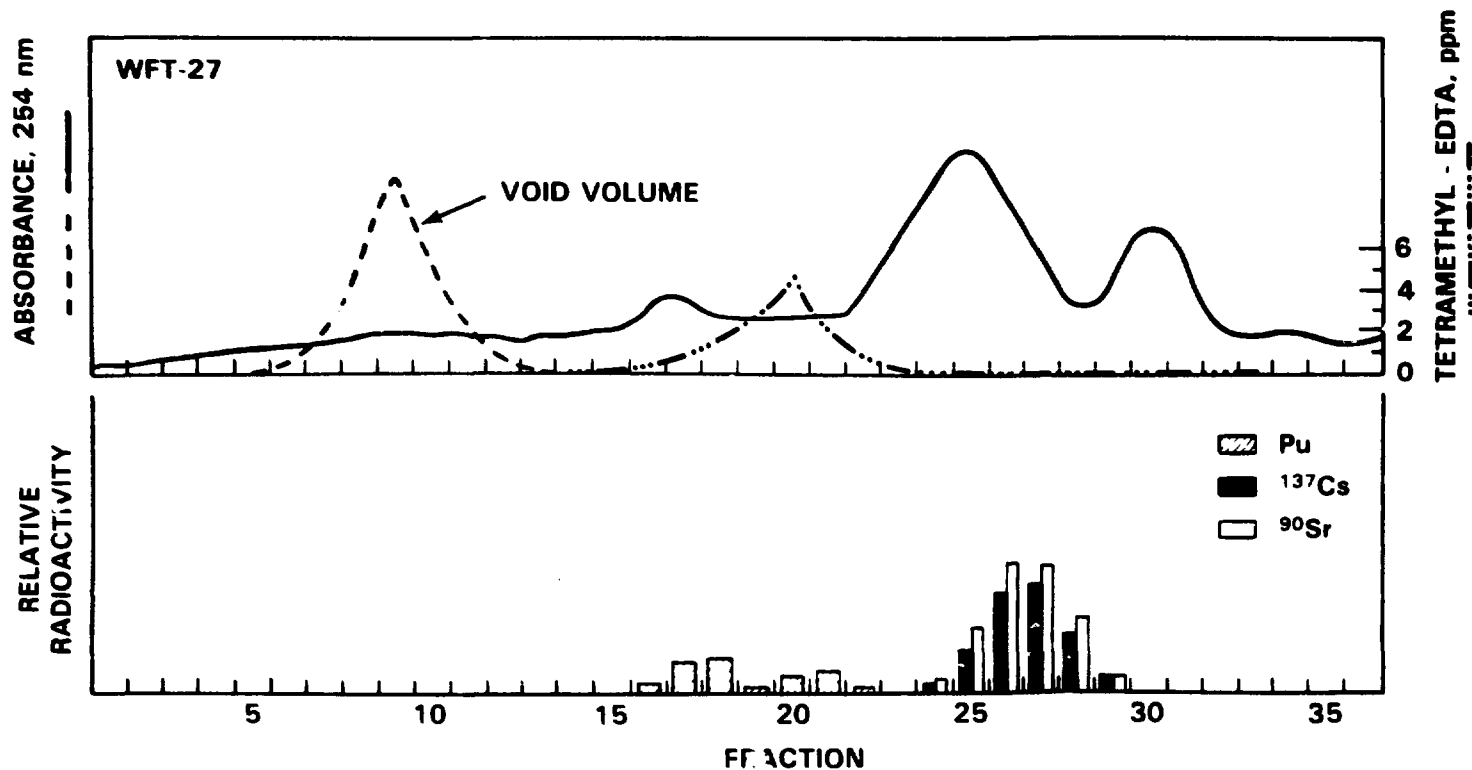


Figure 8. Steric exclusion chromatography (Sephadex G-15) fractionation of waste trench 27 leachate (Maxey Flats). The void volume of the Sephadex column was determined using dextran. Tetramethyl-EDTA was quantified by GC analysis.

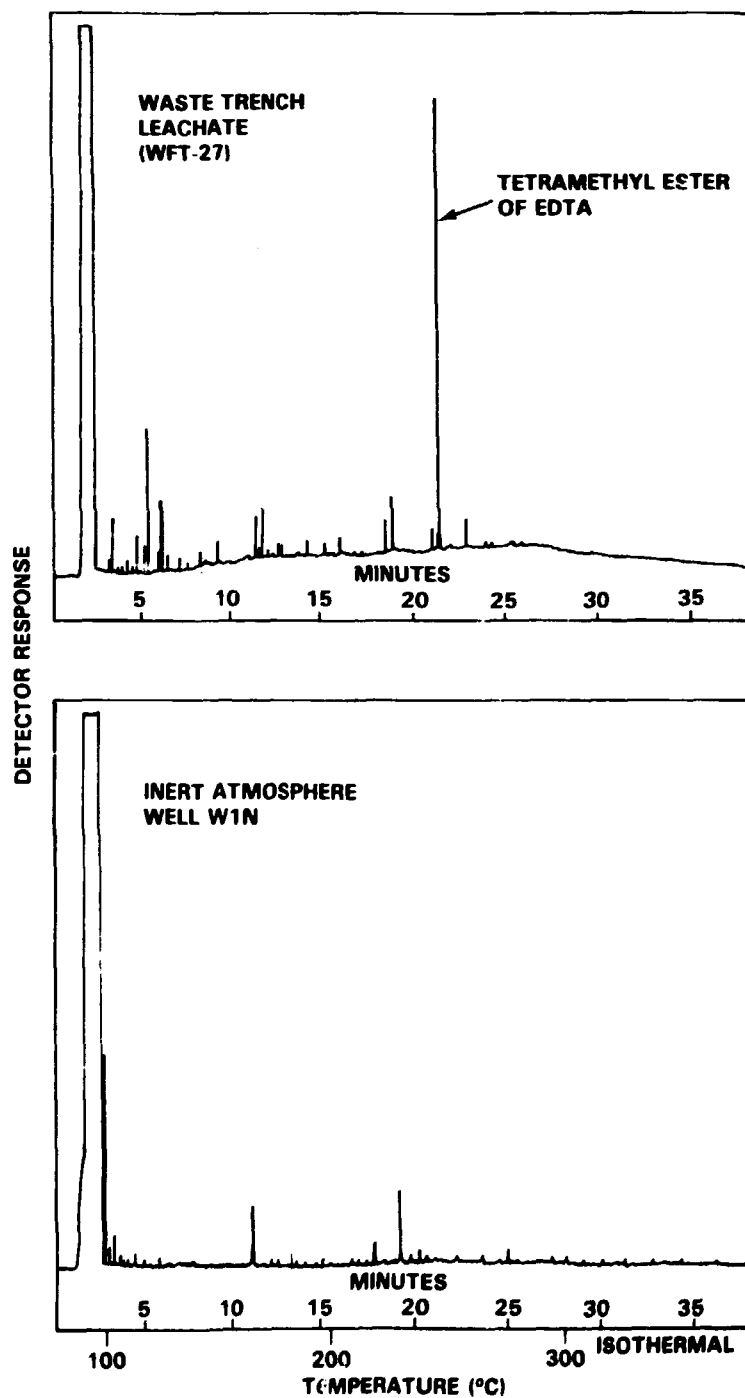


Figure 9. GC analysis of hydrophilic organic fractions (methylated) from Maxey Flats waters. EDTA is the major organic component in the waste trench leachate (37% of hydrophilic organics), but was not detected in inert atmosphere well W1N.

increases in water level had occurred in the experimental trench without movement through trench caps, as confirmed by nonradioactive tracer studies.

In situ counting measurements and soil sampling performed in the main drainage channels leading from the site confirm that contamination is slight, and only  $^{60}\text{Co}$  is derived from the site. Slightly elevated  $^{60}\text{Co}$  levels were measured at the tops of the east and south drainage channels next to the fenced area, but concentrations were comparable to global fallout levels in the hollows below. Cobalt-60 concentrations measured in the west drainage channel are confirmed by soil sample analyses to be surface-deposited. The  $^{60}\text{Co}$  concentrations on the west side of the site approach global fallout levels about 100 meters from the site. Cesium-137 concentrations measured at all locations by in situ counting and by soil sample analysis were comparable to global fallout levels.

Contamination at Maxey Flats has been largely contained within the restricted area. Site remedial and repair work have resulted in a general redistribution of surface material on some portions of the restricted area, with the corresponding decrease or outright removal of previously existing locations of greater than average radioactive content. Consequently, the trend has been toward lower radionuclide surface distributions and diffuse patterns of distribution.

Based on these studies, future research efforts will be necessary to describe the Maxey Flats site and provide modeling data for it and other commercial shallow land burial sites. These include determination of the strength of the plutonium-EDTA complex; determination of the specific chemical species of radionuclide complexes; quantification of hydrophilic organics; determination of the effect of Eh and pH on radionuclide migration; and completion of the areal distribution studies at Maxey Flats. Research should describe the extent of preferential groundwater flow patterns and the rates and mechanisms of groundwater transport; determine the effect of soil solution changes on radionuclide migration, determine and apply countermeasures to minimize water infiltration; and determine the effects of infiltration barriers on offsite radionuclide migration, concentrations of radionuclide complexes in waste trench leachate, and chemical environment of the waste trenches and monitoring stations.

#### ACKNOWLEDGEMENTS

This research is supported by the U. S. Nuclear Regulatory Commission. Pacific Northwest Laboratory is operated by Battelle Memorial Institute under contract DE-AC06-RLO 1830 with the U. S. Department of Energy.

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## ENVIRONMENTAL MONITORING HANDBOOK

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## ABSTRACT

The progress made during the second year of work on the Handbook is described together with directions similar work should take in the future. The Handbook has been extensively revised since the first version was completed in September, 1981. The objectives and scope of the Handbook were altered to some extent, and the final document contains less support and background information, but is directed more toward monitoring specifics than the original draft.

The Handbook addresses monitoring during the three phases of a disposal site - preoperational, operational, and post-operational. Included are procedures for 1) selecting monitoring locations and frequencies, 2) selecting environmental media to be sampled and physical site characteristics to be monitored, and 3) applying statistical criteria to determine the significance of results. Suggested programs for arid and humid locations are presented.

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INTRODUCTION

Environmental monitoring is required of all low-level waste disposal sites. The Nuclear Regulatory Commission's proposed licensing rule, 10 CFR Part 61, requires monitoring during all three phases of the life cycle of a commercial site, and DOE Order 5484.1 requires environmental monitoring of Department of Energy nuclear facilities, including waste disposal sites. While the level of effort needed may be an open question, and is site-specific, there is no question that environmental monitoring is a vital part of a successful site operation. Some of the problems the closed sites have encountered resulted from inadequate monitoring, in both the preoperational and operational phases. In some cases, migration pathways were missed in planning the monitoring program, and monitoring locations were originally poorly selected to intercept waste migration. Proper application of monitoring principles and proper planning and attention to details in the preoperational phase can avoid these problems. Although environmental monitoring has been conducted for many years, the application of monitoring principles to low-level

waste sites has not been well documented. It is in this area that the Handbook can meet an important need in the National Low-Level Waste Management Program (NLLWMP). The technology available for monitoring is improving rapidly in response to concerns over lower levels of contamination. As the technology advances, it is natural that it will be applied. The NLLWMP will need to keep its projects current if it is to provide the technical assistance expected of it.

#### PROGRESS DURING FY 82

The first draft of the Handbook was completed in September, 1981. Two revisions have been prepared in the past year, the first after a review by NLLWMP personnel and the second after a formal peer review by a committee of six individuals outside of the program, each of whom was chosen for expertise in different aspects of environmental monitoring or waste disposal. We believe both reviews were invaluable and resulted in an improved document. The major changes made in the document this year were a reduction in overall size to make it easier to use, a substantial decrease in the amount of background technical information provided, and an emphasis earlier in the document on practical environmental monitoring information. To a considerable extent this was done by placing background material in appendices. In this way readers familiar with such material, or who do not need it for their specific site or specific use, are not diverted from the strictly environmental monitoring information, which is concentrated in the body of the report.

The Handbook now should be used to determine if it will meet a need in waste disposal, and several methods of doing this are discussed in the final section of this paper.

#### DESCRIPTION OF HANDBOOK

##### PURPOSE

The purpose of the Handbook is to provide operators of sites that use shallow-land burial for the disposal of low-level waste with the information necessary to design environmental monitoring programs for their sites. The information is intended to be generic in nature, state-of-the art in approach, and to include the application of statistical principles to program design.

##### SCOPE

Monitoring is defined relatively broadly in the Handbook, and includes onsite measurements to monitor burial-trench performance, as well as the usual offsite monitoring practices. Although the site itself may

be considered a facility, and not part of its environment, the principles and types of measurements needed are so similar to those done offsite, that they should be considered as part of one program, although the results may be handled separately for reporting purposes.

## CONTENTS

The Handbook begins with a discussion of the objectives and principles of environmental monitoring programs and their design. The information on which a program is based is then described--pathway analysis, critical pathways, waste composition and properties (the source term), radiation dosimetry, standards, and the site and area characteristics used in deciding on sampling locations and frequency.

Procedures that may be used to determine where, when, and how often monitoring of specific environmental media should be conducted are then described. This information is used, together with the statistical aspects developed in the document, to describe generic programs for the preoperational, operational, and postoperational phases of a disposal site. This is followed by a discussion of sample collection and general measurement techniques, quality assurance, and reporting. Additional information on some topics referred to in the body of the report are given in appendices. We believe that all the essential aspects of waste-site monitoring are addressed in the document, so that a site operator, an environmental monitoring program manager, and those who assess and evaluate programs and their results will find the Handbook useful. An abridged table of contents is given in Table 1.

## PRINCIPLES OF PROGRAM DESIGN

A monitoring program for a disposal site consists of these basic components 1) measuring concentrations of radioactive and chemical substances in the environment, 2) measuring direct penetrating radiation from the buried waste, 3) measuring certain physical site and area characteristic (e.g., wind speed and direction, ground-water flow rate and direction, erosion, trench subsidence), 4) collecting information on the area (e.g., population distribution, land and water use), 5) interpreting the results in terms of potential health and environmental impact, comparisons with standards and performance objectives, and comparison with the expected site and trench performance.

These monitoring activities can be divided into three groups depending on their purpose, 1) those designed to measure the impact of the site on the public, 2) those designed to monitor performance of the site, 3) those measurements and data collection activities designed to support the first two activities. A useful starting point in program design is a pathway analysis - a common approach to environmental studies of radionuclide movement from a nuclear facility, but equally valid for non-radiological pollutants. A model diagram of the major pathways for a

**Table 1. Generic Handbook on Environmental Monitoring For  
Low-Level Radioactive Waste Disposal Sites -  
Table of Contents (Abridged)**

**1. INTRODUCTION**

Objectives, Scope, and Contents of Handbook

**2. DESIGN OF ENVIRONMENTAL MONITORING PROGRAMS**

Objectives and Principles of Monitoring

Pathway Analysis, Dosimetry, and Standards

Site and Waste Characteristics

**3. MONITORING SYSTEMS**

Sampling Frequencies and Locations

Air, Water, Trench Monitoring

Soil, Plants, Animals, Direct Radiation, Physical Parameters

**4. MONITORING PROGRAM SUMMARY**

Preoperational, Operational, Postoperational

**5. STATISTICAL CONSIDERATIONS OF MONITORING**

**6. SAMPLING AND MEASUREMENT TECHNIQUES**

Air, Water, Soil, Plants and Animals, Foodstuffs, Radioactive

and Non-radioactive Pollutants

**7. QUALITY ASSURANCE**

**8. DATA INTERPRETATION AND REPORTING**

Table 1. (contd)

APPENDICES

- A. PATHWAY ANALYSIS - Detailed Example
- B. HYDROLOGIC MEASUREMENTS  
Hydraulic Conductivity, Porosity, Retardation Factors, Ground-Water Flow Direction, Solute Transport, Numerical Solutions of Ground-Water Flow Equation, Transmissivity, Storage Coefficient, Other
- C. WELL CONSTRUCTION - Overview
- D. ATMOSPHERIC DISPERSION MODELS - Overview
- E. DOSIMETRY AND DOSE CALCULATIONS - Detailed Examples
- F. STATISTICAL METHODOLOGIES

shallow-land burial site is shown in Figure 1. The materials to be sampled and studied and the chains that are the most critical ones to monitor for evaluating human exposure and for evaluating site performance can be seen by examining the diagram. If the transfer coefficients between compartments can be estimated, the relative importance of each pathway can be determined. An example of the quantitative evaluation of the waste  $\longrightarrow$  ground water  $\longrightarrow$  vegetation  $\longrightarrow$  human pathway is given in the Handbook. In general, measurements made close to man give the most reliable exposure information - if the concentrations are sufficiently high. Measurements made close to the source are more likely to be positive, but less reliable in estimating dose because the site-specific transfer coefficients are not well known.

Of particular importance in a pathway analysis is the source term. The quantity of radionuclides and other potential pollutants placed in the trench is difficult to measure or determine, and the amount available for migration (the source term) changes with time. Because the same contaminant can exist in different chemical and physical forms, it can move at different rates and possibly by different pathways from the same trench. Although not strictly part of an environmental monitoring program, it is important that the program manager have or obtain information on the identity, quantity, and physical/chemical form of the buried waste. This information is needed for three principle reasons: 1) to identify the radiological and chemical pollutants for which the monitoring program should search, 2) to identify leachate indicators, which may not be toxic themselves, but are early indicators of migration, and 3) to be aware of the presence of complexing agents or other chemicals that can alter the expected migration characteristics of toxic components of the waste.

The number of radionuclides of concern are limited by their half-lives and the waste types acceptable for shallow-land burial. The number of possible chemically-toxic materials is probably greater. A listing of important radioactive and chemical constituents of low-level waste, based on historical information, is provided. Methods of onsite assay and verification need to be developed and used. A brief discussion of the site and area characteristics and parameters needed to design a program and interpret its results complete the section on design principles.

## MONITORING SYSTEMS

### Subsurface Water

A procedure is presented for deciding on sampling locations and frequencies for monitoring migration from burial trenches by the subsurface water pathways (ground water and soil moisture), which are considered the most probable means of waste dispersal at most sites. The procedure is capable of "providing early warning of release of radionuclides from the disposal site before they leave the site boundary", a monitoring require-

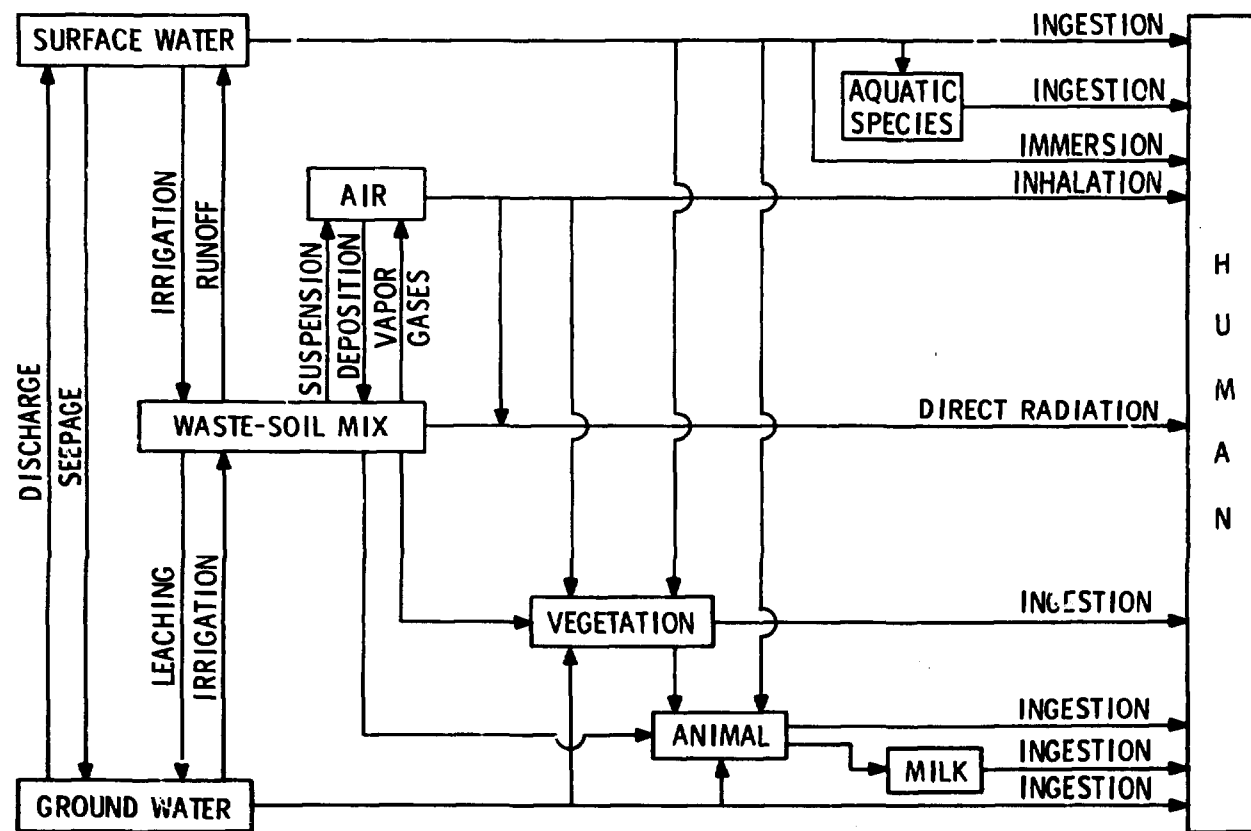


Figure 1. Simplified Pathway Diagram for a Low-Level Waste Burial Facility.



ment in the revised NRC licensing rule 10 CFR Part 61 (May 1982). In the saturated zone, the direction and velocity of ground-water flow is measured or calculated. The results are used to establish primary and secondary monitoring lines. Wells are then placed along these lines at distances from the trench grouping to be monitored, and sampled at frequencies determined by the hydrological analysis described in the Handbook. Ground-water modeling, preferably by numerical (digital) methods, is recommended for determining the general flow direction and speed. The trench monitoring regime begins with sampling the trench sump water (if any). Vadose zone monitoring is also discussed, and is considered useful for arid regions. At locations where the depth of the vadose zone is less than about 50 feet, the difficulty of monitoring this zone may not justify the expense to obtain results of limited value. For completeness, and because such monitoring is performed at some sites, the Handbook presents methods for measuring and calculating flow rate in the vadose zone, and for determining travel time of water through a trench, which is considered as a special case of vadose zone monitoring. The latter process also leads to a method for estimating the rate at which a pollutant leaves a trench, and thus provides a source term.

#### Other Systems

A rationale for air monitoring based on direction and speed is also given. This procedure requires measuring joint frequency distributions of wind speed and wind direction as functions of atmospheric stability. Alternatively, the standard deviations of the plume concentrations can be measured, instead of relying on published estimates. The number and locations of air monitoring stations required to detect releases with a pre-determined probability can be calculated from this meteorological information, the size of the site, and the sensitivity of the pollutant measurements.

The rationale for other types of sample collection and analysis, and for physical measurements, are determined principally by availability and judgement. However, a regular procedure for measuring and monitoring changes in hydraulic head and gradient, meteorological parameters, ground-water velocity, surface erosion, trench subsidence, surface stream flow, and onsite ponding of surface water must be conducted.

#### RECOMMENDED PROGRAMS

Monitoring programs for the preoperational and operational phases of a disposal site in an arid and humid location are given in some detail. The postoperational phase is less detailed, and can be described in very general terms only at this stage.

#### Preoperational

The purpose of this program is to characterize the site before any operations have begun. This program consists principally of collecting

data to evaluate the physical, radiological and non-radioactive pollutant environment of the site and the surrounding area. Some of this information should be available as a result of the site-selection and characterization process, and the two programs should be integrated. Suggested sampling and analytical programs are given in the Handbook. An abridged listing of the materials to be collected and the analyses to be performed are given in Table 2. The program should be conducted for at least one year, although two years is preferable.

#### Operational Program

The operational program is designed to determine environmental impact, measure any waste migration, and evaluate site performance compared to standards, regulations, and performance objectives. The materials sampled and the analyses performed are similar to the preoperational program, except that ground-water monitoring is focused on the monitoring lines discussed earlier and onsite and offsite monitoring are separate and distinct. In this phase, the monitoring group must be alert to changes in the programs indicated by changes in site characteristics, composition of buried waste, and past results.

#### Postoperational Program

The details of this phase will depend on the results of the operational program. It is expected that some measurements will be discontinued. The important pathways, radionuclides, and leachate indicators should be well known, and the programs would concentrate on these. Thus, the air sampling program can probably be discontinued, but subsurface water monitoring should continue, as well as observations of erosion and trench cap stability. It is likely that in situ monitoring techniques will be available that will eliminate the need for some of the sample collection and analytical procedures now used.

### STATISTICAL CONSIDERATIONS

The statistics covered in the Handbook are principally for the purpose of deciding when a measured value exceeds the background or normal value and thus indicates a possible pollutant release. At sufficiently high levels, of course, such decisions are obvious. At levels close to normal, statistical criteria are needed, and their use begins in the preoperational phase. It is necessary to collect a sufficient number of samples and perform sufficient analyses for the important parameters to establish a baseline level against which operational and postoperational results can be compared. This number is about 30, and will give measured means and standard deviations sufficiently close to the true values that the normal distribution (after suitable data transformation, if necessary) may be used in place of the *t* distribution for subsequent statistical tests. Temporal and spatial variations in the data are also discussed, and if they are found, call for additional sampling. During the operational phase, the monitoring data is compared with the preoper-

Table 2. Preoperational Program (Abridged).

## SAMPLE TYPES

|                             |                                   |
|-----------------------------|-----------------------------------|
| Air - particulate           | Soil - surface, onsite, offsite   |
| - water vapor               | Vegetation - onsite, offsite      |
| - radioiodine, gases        | Bottom Sediment - onsite, offsite |
| Precipitation               | Small Mammals                     |
| Direct Radiation            | Game Birds                        |
| Water - surface, subsurface | Aquatic Biota                     |
| - onsite, offsite           | Farm Crops                        |
| Soil - subsurface, onsite   | Milk                              |

## MEASUREMENTS

| <u>Sample Type</u> | <u>Measurements</u>                                                                                                             |
|--------------------|---------------------------------------------------------------------------------------------------------------------------------|
| Air                | Total alpha, beta; gamma-ray spec.,<br>$^3\text{H}$ , $^{14}\text{C}$ , $^{129}\text{I}$ , $^{85}\text{Kr}$ , $^{222}\text{Rn}$ |
| Precipitation      | Gamma-ray spec.                                                                                                                 |
| Water              | Gamma-ray spec., $^3\text{H}$ , $^{90}\text{Sr}$ , U,<br>$^{14}\text{C}$ , $^{99}\text{Tc}$ , $^{226}\text{Ra}$                 |
| Soil               | Gamma-ray spec., $^3\text{H}$ , $^{90}\text{Sr}$ , U                                                                            |
| Vegetation         | Gamma-ray spec., $^3\text{H}$                                                                                                   |
| Bottom Sediment    | Gamma-ray spec.                                                                                                                 |
| Animals            | Gamma-ray spec.                                                                                                                 |
| Crops              | Gamma-ray spec.                                                                                                                 |
| Milk               | Gamma-ray spec., $^3\text{H}$ , $^{129}\text{I}$ , $^{90}\text{Sr}$                                                             |

Note: Some samples should also be analyzed for plutonium and some other alpha emitters to establish a baseline. The amounts of alpha emitters in future sites may increase.

ational means and standard deviations, principally with the use of control charts. Offsite and onsite results are similarly compared with the use of difference control charts.

Procedures for obtaining an operational background level, setting a level above this which requires that some action be taken, and for constructing and using the control charts are presented. An economic model is also presented for minimizing the number of samples to be collected in the operational phase. The model compares the cost of delayed detection with the cost of more frequent sampling, and gives an optimal sampling frequency and sample number.

#### SUMMARY AND FUTURE WORK

This paper presents a qualitative summary of the considerable body of information provided in the Handbook. Because of time and space limitations, the document itself must be consulted for details, quantitative aspects, and rationale for the environmental monitoring concepts given. Several methods for using the Handbook to improve its quality, measure its utility, expand its scope, and, it is hoped, to upgrade existing monitoring programs, are suggested below.

The Handbook should first be tested to determine if it actually meets its objectives and the needs of disposal site operators. This can be done by designing site-specific programs based on the generic guidance provided in the Handbook. Two methods are proposed: 1) select site characteristics and parameters typical of several areas of the country - arid, semi-arid, humid - high permeability, humid - low permeability, and design site-specific programs based on these characteristics and 2) apply the generic guidance to existing sites and their programs. The second method cannot test the preoperational phase, but will give a more realistic test of the operational phase guidance than the first method, since actual data and measurements are available. This method is also a direct technology transfer exercise.

Another valuable technology transfer exercise would be to use the Handbook for any potential new sites that are identified. One of the major objectives of the NLLWMP is to provide the technology needed to open a new waste disposal site by 1985 and this would be an appropriate time to apply the Handbook.

#### ACKNOWLEDGEMENT

Environmental monitoring requires knowledge and experience in several scientific disciplines. The body of knowledge that is required does not reside in one person, and several contributed to the preparation of the Handbook. The version discussed in this paper was written by:

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4th ANNUAL PARTICIPANTS' MEETING  
DOE LOW-LEVEL WASTE MGMT. PROGRAM  
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MONITORING FOR LOW-LEVEL RADIO-  
ACTIVE WASTE DISPOSAL SITES

**GENERIC HANDBOOK ON  
ENVIRONMENTAL MONITORING FOR  
LOW-LEVEL  
RADIOACTIVE WASTE DISPOSAL  
SITES**

prepared by

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**August, 1982**

**Work Package ANL 038 02 82**

**U.S. DOE**

**National Low-level Waste Management Program**

JS - 8/31/82

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### OBJECTIVES OF ENVIRONMENTAL MONITORING PROGRAMS

- DETECT AND MEASURE ANY WASTE MIGRATION
- ASSESS IMPACT OF SITE OPERATIONS ON PUBLIC AND ENVIRONMENT
- SATISFY REGULATIONS AND REQUIREMENTS
- OBTAIN DATA FOR LONG-TERM PREDICTION OF SITE BEHAVIOR
- OBTAIN DATA ON SITE AND POLLUTANT BEHAVIOR FOR MODIFYING OPERATIONS AND FUTURE DESIGN



## COMPONENTS OF ENVIRONMENTAL MONITORING PROGRAMS

- MEASURE CONCENTRATIONS OF RADIOACTIVE AND CHEMICAL SUBSTANCES IN THE ENVIRONMENT
- MEASURE DIRECT PENETRATING RADIATION
- MEASURE SITE AND AREA CHARACTERISTICS
- COLLECT APPLICABLE AREA DATA AND INFORMATION
- INTERPRET RESULTS

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- MEASUREMENTS AND DATA COLLECTION TO SUPPORT ABOVE ACTIVITIES

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TRENCH  
VADOSE ZONE  
SATURATED ZONE

### SURFACE WATER

RUNOFF  
BASE FLOW

### AIR

### PHYSICAL, CHEMICAL SITE PARAMETERS

## MONITORING PROGRAMS

PREOPERATIONAL  
OPERATIONAL  
POSTOPERATIONAL

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### OPERATIONAL BACKGROUND LEVEL

$$OBL = M_B + S_B$$

$$AL = OBL + 1.5S_B$$

MODIFY FROM EXPERIENCE

### CONTROL CHARTS

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### ECONOMIC SAMPLING MODEL

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## FUTURE WORK

### DESIGN OF SITE-SPECIFIC PROGRAMS

HYPOTHETICAL SITES  
OPERATING SITES

### POTENTIAL FUTURE SITES

### HANDBOOK REVISION

SLB TECHNOLOGY-HUMID<sup>1</sup>

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Shallow land burial is the predominant mode of disposal of past and present solid low level radioactive waste in the U.S. By and large such disposal has been reasonably successful, i.e. both economic and safe. A higher success rate and greater public confidence and acceptance can be achieved by attention to potential site performance problems by either a well-designed prevention or a timely correction of problems. To aid this approach to low-level waste management, the National Low-Level Waste Program sponsored detailed reports identifying the principal technical problems in shallow land burial (1, 2). Key issues relating to the nature of the disposal site, to the nature of the waste, to the site operational procedures and to the evaluation of site performance are included (Table 1). For SLB sites in humid regions the leaching of radionuclides and their subsequent transport in ground water form the dominant technical considerations. The project discussed here is directed toward prevention of hydrologic mobilization and transport of radionuclides. Aspects of SLB ranging from site selection and characterization through design and operation to stabilization and closure are integrated into this work.

This project is a part of the DOE Low-Level Waste Management Program's Work Breakdown Structure (WBS) Element 3.5.4, "Shallow Land Burial Technology," and is supportive of Milestone C, "Providing the documentation required to support the locating of a Shallow Land Burial Site in the United States." The central effort has been establishment of an experimental burial site, the Experimental Test Facility (ETF), in which improved site characterization and advanced burial procedures are demonstrated (Table 2). Among the issues identified by the Steering Committee (1), most will be addressed by the work carried out at the ETF (Table 1).

## SITE CHARACTERIZATION METHODOLOGY

Both DOE and NRC view the site selection process as the most important aspect of shallow land burial. Utilization of a poor site will require reliance on engineered systems to meet the desired performance objectives, whereas a site that is carefully selected could rely on natural geologic, hydrologic, and meteorologic conditions to minimize contaminant transport and require only minimum engineering features to supplement existing conditions. Both DOE and NRC have proposed site selection criteria for shallow land burial

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<sup>1</sup> Research sponsored by U.S. Department of Energy under Contract W-7405-eng-26 with Union Carbide Corp.

Table 1. Shallow Land Burial Issues Identified in Reference 1

| Title                                                                                                | Type of Problem Resolved |                       | Addressed<br>in<br>ETF |
|------------------------------------------------------------------------------------------------------|--------------------------|-----------------------|------------------------|
|                                                                                                      | Remedial<br>Action       | Future<br>Improvement |                        |
| <u>Site Issues</u>                                                                                   |                          |                       |                        |
| Development of Techniques to Characterize and Improve Sites for Optimizing Burial Ground Performance | X                        | X                     | Yes                    |
| Development and Evaluation of Erosion and Intrusion Control Barriers                                 | X                        | X                     | Yes                    |
| Evaluation of the Technical Feasibility of Intermediate Depth Burial                                 |                          | X                     | No                     |
| <u>Waste Issues</u>                                                                                  |                          |                       |                        |
| Determination of Mechanisms Controlling Water/Radionuclide Transport in Unsaturated Zones            | X                        | X                     | Yes                    |
| Determination of the Effects of Organic and Inorganic Complexation on Transport of Radionuclides     | X                        | X                     | Yes                    |
| Determination of Sorptive Capacities for Radionuclides in Soil/Solution Systems                      |                          | X                     | Yes                    |
| Determination of the Effects of Soil Microorganisms on Transport and Transformation of Radionuclides |                          | X                     | No                     |
| Determination of the Potential Hazards from Long-Lived Radionuclides                                 |                          | X                     | No                     |

Table 1. Shallow Land Burial Issues Identified in Reference 1  
(continued)

| Title                                                                                                  | Type of Problem Resolved |                       | Addressed<br>in<br>ETF |
|--------------------------------------------------------------------------------------------------------|--------------------------|-----------------------|------------------------|
|                                                                                                        | Remedial<br>Action       | Future<br>Improvement |                        |
| <u>Operational Issues</u>                                                                              |                          |                       |                        |
| Development of Methods to Prevent or Minimize Structural Failure of Trenches                           | X                        | X                     | Yes                    |
| Development of Instrumentation to Verify Waste Receipts                                                |                          | X                     | No                     |
| Development of Operational Practices to Minimize Reliance on Post-Operational Control and Surveillance |                          | X                     | Yes                    |
| <u>Performance Evaluation Issues</u>                                                                   |                          |                       |                        |
| Development of Methodology and Equipment for Monitoring Burial Sites                                   | X                        | X                     | Yes                    |
| Integration and Verification of Models for Evaluation of Waste Burial                                  | X                        | X                     | Yes                    |

**Table 2. Objectives of Engineered Test Facility (ETF)**

- 
1. Evaluation of Site Characterization Techniques
  2. Trench Treatment Evaluations
  3. Integrating Site Characterization Data with  
Model Development
  4. Construction and Validation of Model Describing  
Site Performance
-



sites. Although the DOE criteria are developed for defense wastes and the NRC criteria for commercial (private) wastes, careful comparison indicates that both sets of criteria are highly oriented to considerations regarding the geology and hydrology of the site. In fact, selection of a proper site is considered to be the most important factor in assuring that the disposal operation will meet the performance objectives; engineered improvements to the site are considered an adjunct to proper site selection.

A site selection process for the ETF was not performed as described in the DOE or NRC site selection scenarios. As a result the ETF is not a test of site selection processes, but rather a facility in which site characterization for humid regions is being tested, and engineered features applied in order to allow this site to meet the performance objectives. The site utilized for the ETF was selected on the following bases:

1. The ETF must be located on Federal or State owned land.
2. The site should be close to or a part of an existing burial ground.
3. The site should be near ORNL.
4. The site should be capable of handling low-level radioactive waste.
5. The site must not be influenced by nearby operations.
6. The site should be representative of conditions anticipated in burial operations in humid climates.

At the onset of the planning for the ETF it was apparent that the site was marginal (in terms of the selection criteria) and that engineering features would be required to minimize migration. However, this type of site is necessary if the experimental objectives are to be accomplished within a five year period of experimental observation.

The site selection for the ETF is located within a portion of ORNL Solid Waste Storage Area (SWSA) 6, which is an area of the ORNL reservation that has been used for low-level waste disposal operations since 1969. Although previous studies of ORNL burial operations in this area have provided considerable meteorologic, geologic, and hydrologic background on the ETF site, detailed information on the specific area of interest did not exist. Thus, first priority was a detailed site characterization.

A report on site characterization results (3) is in the final stages of review and is scheduled for publication at the end of FY 1982. The contents of that report will not be repeated here, but only briefly summarized. Included in the site characterization report are descriptions of the bedrock, soil properties, meteorology, and surface and ground water. The site is underlain by a silty intraclastic limestone interbedded with mudstone/shale. The soil is illitic in nature and retains the structure of the parent material. Precipitation levels were lower than average for the study period to date, and as a result the water level in the area receded to very low levels. Tracer studies showed the ground water moves principally along the direction of structural strike which corresponds to the direction of a prominent joint set. A significant amount of ground water also moves in other joint sets and fractures of the bedrock that are at an orientation different than that of the structural strike. The significant observation is, however, that ground water flow is dominantly fracture controlled. The surface water was gauged for water budget calculations. Site characterization methodology is indicated in Table 3.

Perhaps the most significant feature to be recognized in this site characterization has been the relict structure of the partially weathered, folded rock that traverses the site (3). Tracer tests and drawdown tests in wells show the profound effect of this relict structure on near surface ground water flow. Consequently monitoring and modeling will require that the site-specific flow characteristics of the ETF be incorporated into the experimental design with a high level of detail.

#### EXPERIMENTAL DESIGN FOR TRENCH TREATMENT STUDIES

In addition to site characterization studies, the ETF is designed to evaluate the effects of various waste/trench treatments as a means of minimizing contaminant transport and subsidence for trenches where there is a high probability that surface or ground water will come into contact with the buried waste. Treatments to be studied include grouting the waste in the trench and the use of liners. Development of waste treatments was reported last year (4).

A Latin Square type of statistical design represents the simplest approach for this type of experiment and provides three replicates for each treatment and three control trenches having no treatment. Each treatment exists in only one row and in one column of a 3 x 3 matrix. Other types of statistical design (e.g. a factorial design) would require additional trenches and greater amounts of wastes.

Table 3. Data Types and Methods References

| Data             | Method References                                                                                                                   |
|------------------|-------------------------------------------------------------------------------------------------------------------------------------|
| 1. Geology       |                                                                                                                                     |
| A. Stratigraphy  | Lahee, F. H., 1961, Field Geology Chap. 2, 3, 4; pp. 21-33, 37-52, 62-130.                                                          |
| B. Structure     | Lahee, F. H., 1961, Field Geology, Chap. 7 and 15; pp. 185-195, 270-279, 446-453.                                                   |
| 2. Geophysics    |                                                                                                                                     |
| A. Resistivity   | Dobrin, M. B., 1960, Introduction to Geophysical Prospecting Chap. 17; pp. 339-373.                                                 |
| B. Seismic       | Dobrin, M. B., 1960, Introduction to Geophysical Prospecting, Chap. 5 and 6; 69-104, 105-147.                                       |
| 3. Pedology      |                                                                                                                                     |
| A. Soil Type     | USDA, 1962, Soil Survey Manual Handbook No. 18.                                                                                     |
| B. Mineralogy    | U.S. SCS, 1975, Soil Taxonomy: A Basic System of Soil Classification for Making and Interpreting Soil Survey, Handbook 436.         |
| C. Chemistry     | U.S. SCS, 1972, Soil Survey Laboratory Methods and Procedures for Collecting Soil Samples. Soil Survey Investigations Report No. 1. |
| 4. Precipitation | Anon. 1978, Stevens Water Resources Data Book Chap. 1; pp. 12-19.                                                                   |
| 5. Surface Water | Anon. 1978, Stevens Water Resources Data Book, Chap. 4; pp. 37-51                                                                   |
| A. Flow          |                                                                                                                                     |
| B. Water Quality |                                                                                                                                     |

Table 3. Data Types and Methods References  
(continued)

| Data               | Method References                                               |
|--------------------|-----------------------------------------------------------------|
| 6. Ground Water    |                                                                 |
| A. Water Levels    | Johnson UOP, 1975, Ground Water and Wells, Chap. 5; pp. 81-98.  |
| B. Flow Directions | Johnson UOP, 1975, Ground Water and Wells, Chap. 6; pp. 99-144. |
| C. Water Quality   | Johnson UOP, 1975, Ground Water and Wells, Chap. 4; pp. 65-80.  |

## TRENCH DESIGN

The site available for the ETF is a 0.3 hectare tract in the northwest portion of SWSA 6. The experimental trenches are located on a small hillock at an elevation of 242 to 244 m which slopes downward from north to south. In order to minimize interference with the wells drilled during the characterization phase, the nine experimental trenches were located within the space enclosed by the wells. Trench size selected was 3 m x 3 m x 3 m deep in order to fit the available space and not require a long time interval for sufficient waste to be accumulated to fill the trenches.

Trenches were excavated with a backhoe which is used to excavate trenches used for routine waste disposal operations. Photographs were taken of the sidewalls of each trench to document stratigraphic variability.

## WASTE FORM

Originally it was anticipated that a synthetic waste would be used in the experiments to minimize variability of waste among the trenches; however, the preparation of about 160 m<sup>3</sup> of "synthetic" waste presented significant blending and handling problems. Baled, compactible waste was selected for use in the ETF because it is easier to handle and is more uniform in composition than the non-compactible waste. Since the baled waste currently being produced at ORNL averages around 3 m<sup>3</sup>/mo. (after compaction) much more than one year's supply of waste was required to fill the nine experimental trenches. Approximately 200 bales were needed for the trenches.

Average ORNL low-level waste production during FY 1981 was about 160 m<sup>3</sup>/mo., of which 26 m<sup>3</sup>/mo. was classified as compactible. A Consolidated Baling Company compactor, installed in 1978, produces bales that are nominally 0.50 x 0.76 m x 1 m (0.4 m<sup>3</sup> or 15 ft<sup>3</sup>) in size, and weigh about 300 kg (600 to 650 lbs). Waste is compacted inside a plastic lined cardboard box which is then metal strapped and finally wrapped in plastic film. The compactor is capable of achieving a 9:1 compaction ratio (volume basis) using ORNL waste.

## MONITORING

The 12 wells installed for the pre-construction tracer tests will be used as a part of the monitoring system for the ETF. These wells are located around the perimeter of the ETF trenches, with the central one located within the array of nine trenches. In addition 24 wells were placed at a spacing of 1.5 m from the trench wall. A total of 36 wells are currently available for monitoring tracer movement from the trenches.

Wells for the pre-construction tracer test are 15.2 cm ID, PVC plastic pipe (schedule 40), having the lower 1.5 m screened and gravel packed outside the casing. The upper portion of the well casings are sealed from the formation with bentonite. The 24 trench monitoring wells are 6 m deep, 7.6 cm ID (schedule 40 PVC pipe), and slotted and gravel packed in the bottom 3 m. All wells are capped to prevent surface water and animal intrusion.

Water samples will be withdrawn from each of the monitoring wells by recommended methods (5). Samples will be filtered and acidified prior to analysis using standard methods (6). Table 4 illustrates the sampling frequencies established for the treatment evaluation portion of the ETF. These frequencies represent the minimum for each of the measurement/samples, and may be modified as experimental data and trends accumulate.

#### DATA ANALYSIS, EVALUATION, AND REPORTING

To date the trenches have been filled and closed. Present efforts are devoted to monitoring and modeling. A significant amount of data (precipitation, stream flow, water table levels, chemical analyses, and subsidence measurements) will be collected during the characterization and monitoring phases of the ETF. In addition to numeric data, strip charts, photographs, and written descriptions of cores and soil samples will be produced and will require storage.

A data management system, utilizing computer storage, will be developed for the numeric information and will allow the data to be classified and retrieved as desired. Each principal investigator will be responsible for insuring that the data is reasonable prior to entering it into the data management system, and will also be responsible for checking the data after input to eliminate transcription errors.

Non-numeric information will be stored in a centralized filing system to provide access to all investigators. A log system will be used to identify materials entering into the files.

For modeling, the site will be considered as an initial array of finite elements (spatial volumes), the distribution of whose hydraulic properties will be dictated by the measured site characteristics. Initially, two finite element models (7, 8), will be applied to the drainage basin of the site. Currently, these models are oriented to the description of ground water flow using inputs of meteorological data and hydraulic properties of the porous medium and outputting ground water flow rates and directions and water table elevations. The chemical adsorption subroutine of FEMWASTE for calculating

Table 4. Sampling Frequencies for Treatment Portion of ETF

| Measurement/Sample<br>(Number) | Weekly | Bi-Weekly | Monthly | Quarterly | Continuous | Remarks |
|--------------------------------|--------|-----------|---------|-----------|------------|---------|
| <u>Water Levels</u>            |        |           |         |           |            |         |
| Perimeter Wells (12)           |        |           |         |           | X          |         |
| Monitoring Wells (24)          |        | X         |         |           |            |         |
| Piezometer Wells (3)           |        |           |         |           | X          |         |
| Trench Wells                   |        |           |         |           |            |         |
| Controls (3)                   |        |           | X       |           |            |         |
| Grouts (3)                     |        |           | X       |           |            |         |
| Liners (3)                     |        |           | X       |           |            |         |
| <u>Samples</u>                 |        |           |         |           |            |         |
| Perimeter Wells (12)           |        | X         |         |           |            |         |
| Monitoring Wells (24)          |        | X         |         |           |            |         |
| Piezometer Wells (3)           |        | X         |         |           |            |         |
| Trench Wells                   |        |           |         |           |            |         |
| Controls (3)                   |        |           |         | X         |            |         |
| Grouts (3)                     |        |           |         | X         |            |         |
| Liners (3)                     |        |           |         | X         |            |         |
| <u>Surface Runoff</u>          |        | X         |         |           |            |         |
| <u>Precipitation</u>           |        |           |         |           | X          |         |

contaminant transport will need to be improved to account for non-linear adsorption isotherms, ion exchange reactions, and precipitation/dissolution reactions. An integrated compartment model will be adapted to the site since, potentially, it has more versatility in handling irregularly-shaped spatial volumes than the regularly-shaped finite element models. Validation of each model will be an ongoing refinement as more geohydrological data from the site are collected. For example, a number of finite element constructions could generate an adequate description of the site's water regime when only a limited range of observational conditions have been encountered. But when a period of prolonged precipitation or drought is encountered, certain finite element constructions will predict geohydrological responses better than others. In addition pump tests at various monitoring wells will provide water table responses which will be predicted better by some constructions than others. Through such repeated validation tests, an optimum and unique site model can be constructed. More importantly the entire process and experience of applying a model interactively with experimentally determined site characteristics and validating that model with field tests, will be documented and available for future site operators to use.

Given the evolution of a unique site model, which describes the period of experimental observation, extrapolation of site performance into the long-term will be performed by driving the site model through hundreds of years of simulated climatological data.

The model described above will play a crucial role in the evaluation of the liner and grout treatments. It will be used to predict when and where ground water tracers would have moved had they not been retained within treated trenches. The appearance or lack of appearance of these tracers at these predicted places and times will be interpreted as treatment failure or success, respectively. Liner integrity, after three years of performance, can be tested by water pumped in through the access well in these trenches: retention of water within the trench will demonstrate the integrity of the liners. Tracers can also be added repeatedly to these trenches to demonstrate their lack of migration. Additional samples of the liner material will be buried at the site and recovered yearly for testing liner properties including tear strength, tensile strength and seam strength (ASTM D-751). Based on the change in these properties, an estimation of an effective liner lifetime will be made.



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## SHALLOW LAND BURIAL TECHNOLOGY DEVELOPMENT-ARID

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## ABSTRACT

The experimental results obtained during FY-1982 on biointrusion barrier testing, migration barrier testing, and ground and surface water management system testing are described. The results of the small lysimeter study on biointrusion barriers are presented and the larger scale experiments in progress are described. The results of the experiments to determine the migration potential for water under unsaturated conditions are described. Preliminary results on the wick system experiment are presented. A comparison of model calculations and experimental results on the water movement experiments is also presented.

## INTRODUCTION

The work reported in this paper provides the arid site contribution to Headquarters Controlled Milestone LL82.8 - "Submit to HQ Reports on Shallow Land Burial Technology Field Tests: ORNL and Los Alamos National Laboratory." This paper will present the results obtained to date on the Shallow Land Burial Technology Field Tests that are currently underway in the Los Alamos Experimental Engineered Test Facility (EETF). The EETF was described in detail at the Third Annual Information Meeting for the DOE Low-Level Waste Management Program (1). An updated map of the facility showing all of the emplaced experiments for this and other tasks in the National Low-Level Waste Management Program (NLLWMP) is included as Figure 1. The work to be reported here is divided into four subtasks in FY-1982.

- 1) Biointrusion Barrier Testing
- 2) Migration Barrier Testing
- 3) Ground and Surface Water Management Systems Testing
- 4) Model Verification.

In FY83 a fifth subtask will be added

- 5) Integrated Systems Testing.

The Shallow Land Burial Technology field tests described in this paper have been performed for the NLLWMP to provide data and information base on SLB Technology resulting from large scale field experiments performed in the EETF. The general philosophy behind the field tests is that SLB sites must be capable of being characterized, modeled, analyzed, and monitored. To know what is necessary to characterize a site, and to be able to model, analyze, and monitor a site, the transport pathways for contamination to leave the site must be clearly defined. The major contaminant pathways from SLB sites result from biological processes such as the transport by plants and animals and from the interaction of water with the buried wastes with subsequent mobilization and transport. The results of these field tests will both quantify the transport pathways and provide documented methods for preventing contamination from leaving the site by these pathways.

A SLB facility must perform as an entire system, and the interactions between the components of the system must be considered. The results presented here and in other papers in these Proceedings describing other work at Los Alamos will emphasize the interactive nature of the components that make up an effective SLB system.

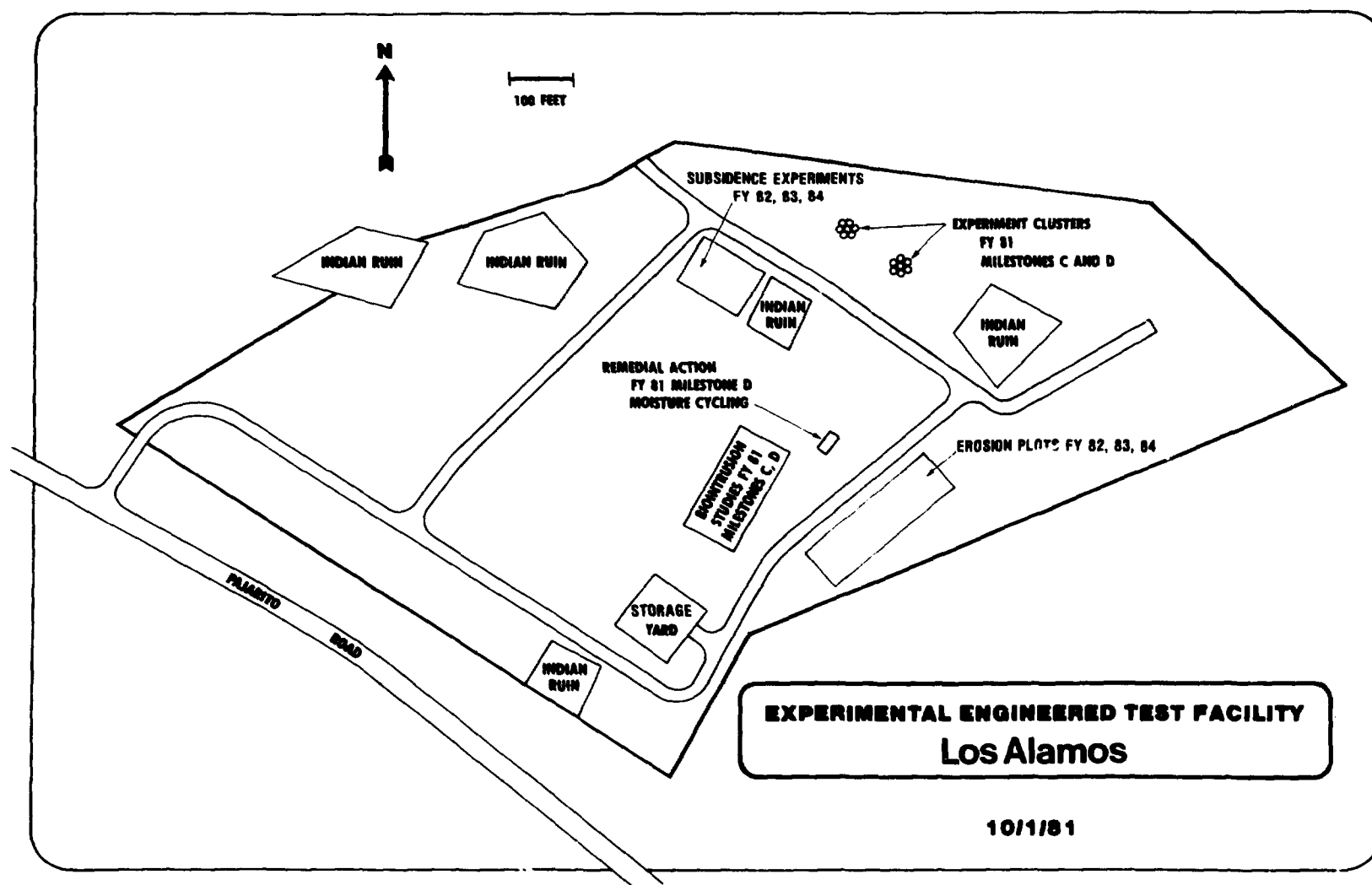


FIGURE 1.

## BIOINTRUSION BARRIER TESTING

The details of the small-scale biointrusion barrier testing experiments were presented at the Third Annual Participant's meeting (2).

Only the experimental results will be summarized here. The experiment design and physical characteristics of the biobarrier materials used in the plant root intrusion study are given in Tables 1 and 2, respectively.

The cesium concentration in plants growing on a particular barrier system was used to determine whether plant roots had intruded through that barrier system. A concentration of 10 ppm or greater of cesium in plant tissue was considered to conclusively indicate that root intrusion through the barrier system had occurred. Background concentrations of cesium in vegetation generally averaged less than 1 ppm, although the large natural variability usually associated with such measurements (CVs of 100%) dictated the need to select the 10 ppm threshold. The barrier system was considered to be penetrated from the first time a cesium level of 10 ppm was measured, even if concentrations at subsequent collection times were below 10 ppm. Cesium (an analog of potassium) is transported from above-ground parts back to the root system as a plant matures (3). Thus a plant's roots may have penetrated the biobarrier and levels of cesium greater than 10 ppm measured, yet at later sampling periods, levels of cesium below 10 ppm would be measured.

Analysis of the tracer data indicated that cover profile performance was a function of plant species, soil and barrier thickness, barrier type, and time following seeding. Crushed tuff, the sandy backfill material used in covering low-level waste sites at Los Alamos, offers little resistance to plant roots, irrespective of plant species (Table 3) and soil-barrier thickness combinations (Table 4). In as little as 101 days (14 wk) roots had intruded through 69 of the 72 lysimeters that contained crushed tuff as a barrier material. The pattern of high root intrusion through crushed tuff was consistent throughout the remainder of the study.

It should be noted that the decrease in the rate of barley root intrusions during the latter sampling periods (Table 3) was expected since barley, which is an annual, reaches senescence before alfalfa and sweet clover. A decrease in the rate of root intrusions by alfalfa and clover, as measured by the cesium tracer, would also be expected as the plants mature.

The clay, cobble, and cobble-gravel barrier systems were all much more effective than crushed tuff in limiting root intrusion, even at minimum soil and barrier thickness combinations (Tables 5-7).

TABLE 1. FACTORIAL DESIGN OF THE STUDY PLANT ROOT INTRUSION THROUGH BIOBARRIERS

| Factor                     | Number of Levels | Remarks                                                    |
|----------------------------|------------------|------------------------------------------------------------|
| Plant species              | 3                | Barley, alfalfa, and sweet clover                          |
| Top coil thickness         | 2                | 30 cm, 60 cm                                               |
| Biobarrier type            | 4                | Crushed tuff, bentonite clay, cobble, and cobble-gravel    |
| Biobarrier thickness       | 3                | Clay (15 cm, 30 cm, 45 cm)<br>Others (30 cm, 60 cm, 90 cm) |
| Replications within cells  | 4                |                                                            |
| Total number of lysimeters | 288              |                                                            |

TABLE 2. PHYSICAL CHARACTERISTICS OF BIOBARRIER MATERIALS USED IN THE PLANT ROOT INTRUSION STUDY

| Material       | Particle Diameter | Remarks                                           |
|----------------|-------------------|---------------------------------------------------|
| Crushed tuff   | 50 – 100 $\mu$ m  | Included tuff particles as large as 5 cm diameter |
| Bentonite clay | Colloidal         | Clay mixed with water (1:10 by weight)            |
| Cobble         | 7.5 – 12 cm       | Washed                                            |
| Gravel         | 1 – 2 cm          | Washed                                            |

**TABLE 3. CUMULATIVE NUMBER OF ROOT PENETRATIONS THROUGH BIOBARRIER AS A FUNCTION OF VEGETATION TYPE, BIOBARRIER TYPE, AND ELAPSED TIME SINCE SEEDING. SAMPLE SIZE FOR EACH COLUMN IS 4.**

|                      | Elapsed Time<br>(days) |     |     |     |     | Never<br>Penetrated |
|----------------------|------------------------|-----|-----|-----|-----|---------------------|
|                      | 72                     | 101 | 119 | 143 | 172 |                     |
| <b>Crushed Tuff</b>  |                        |     |     |     |     |                     |
| Alfalfa              | 14                     | 23  | 23  | 23  | 23  | 1                   |
| Barley               | 24                     | 24  | 24  | 24  | 24  | 0                   |
| Clover               | 17                     | 22  | 23  | 23  | 23  | 1                   |
| <b>Clay</b>          |                        |     |     |     |     |                     |
| Alfalfa              | 4                      | 8   | 10  | 11  | 15  | 9                   |
| Barley               | 13                     | 14  | 14  | 14  | 14  | 10                  |
| Clover               | 7                      | 10  | 11  | 11  | 11  | 13                  |
| <b>Cobble</b>        |                        |     |     |     |     |                     |
| Alfalfa              | 1                      | 4   | 5   | 8   | 8   | 16                  |
| Barley               | 12                     | 18  | 18  | 18  | 18  | 6                   |
| Clover               | 0                      | 2   | 4   | 4   | 4   | 20                  |
| <b>Cobble-Gravel</b> |                        |     |     |     |     |                     |
| Alfalfa              | 0                      | 0   | 3   | 3   | 4   | 20                  |
| Barley               | 15                     | 18  | 20  | 20  | 20  | 4                   |
| Clover               | 0                      | 1   | 2   | 4   | 5   | 19                  |

**TABLE 4. CUMULATIVE NUMBER OF ROOT PENETRATIONS THROUGH CRUSHED TUFF BIOBARRIERS AS A FUNCTION OF SOIL THICKNESS, BIOBARRIER THICKNESS, AND ELAPSED TIME SINCE SEEDING. SAMPLE SIZE FOR EACH COLUMN IS 12, WITH PERCENTAGE OF PENETRATIONS IN PARENTHESES**

| Elapsed<br>Time<br>(days) | Soil-Barrier Thickness Combinations<br>(cm) |          |          |          |          |         |
|---------------------------|---------------------------------------------|----------|----------|----------|----------|---------|
|                           | 30-30                                       | 60-30    | 30-60    | 60-60    | 30-90    | 60-90   |
| 53                        | 12 (100)                                    | 5 (42)   | 6 (50)   | a        | a        | a       |
| 72                        | 12 (100)                                    | 12 (100) | 12 (100) | 8 (67)   | 7 (58)   | 4 (35)  |
| 77                        | 12 (100)                                    | 12 (100) | 12 (100) | a        | a        | a       |
| 101                       | 12 (100)                                    | 12 (100) | 12 (100) | 12 (100) | 12 (100) | 9 (75)  |
| 119                       | 12 (100)                                    | 12 (100) | 12 (100) | 12 (100) | 12 (100) | 10 (83) |
| 148                       | 12 (100)                                    | 12 (100) | 12 (100) | 12 (100) | 12 (100) | 10 (83) |
| 172                       | 12 (100)                                    | 12 (100) | 12 (100) | 12 (100) | 12 (100) | 10 (83) |
| Never<br>Penetrated       | 0 (0)                                       | 0 (0)    | 0 (0)    | 0 (0)    | 0 (0)    | 2 (17)  |

<sup>a</sup>Samples not taken.

TABLE 5. CUMULATIVE NUMBER OF ROOT PENETRATIONS THROUGH BENTONITE CLAY BIOBARRIERS AS A FUNCTION OF SOIL THICKNESS, BIOBARRIER THICKNESS, AND ELAPSED TIME SINCE SEEDING. SAMPLE SIZE FOR EACH COLUMN IS 12, WITH PERCENTAGE OF PENETRATIONS IN PARENTHESES

| Elapsed Time<br>(days) | Soil-Barrier Thickness Combinations<br>(cm) |        |         |        |        |         |
|------------------------|---------------------------------------------|--------|---------|--------|--------|---------|
|                        | 30-15                                       | 60-15  | 30-30   | 60-30  | 30-45  | 60-45   |
| 53                     | 5 (42)                                      | 2 (17) | 2 (17)  | a      | a      | a       |
| 72                     | 11 (92)                                     | 5 (42) | 7 (58)  | 0 (0)  | 1 (8)  | 0 (0)   |
| 77                     | 12 (100)                                    | 6 (50) | 9 (75)  | a      | a      | a       |
| 101                    | 12 (100)                                    | 7 (50) | 9 (25)  | 2 (17) | 2 (17) | 0 (0)   |
| 119                    | 12 (100)                                    | 7 (58) | 9 (75)  | 2 (17) | 4 (33) | 1 (8)   |
| 148                    | 12 (100)                                    | 8 (67) | 9 (75)  | 2 (17) | 4 (33) | 1 (8)   |
| 172                    | 12 (100)                                    | 8 (67) | 11 (92) | 3 (25) | 5 (42) | 1 (8)   |
| Never<br>Penetrated    | 0 (0)                                       | 4 (33) | 1 (8)   | 9 (75) | 7 (58) | 11 (92) |

\*Samples not taken.

TABLE 6. CUMULATIVE NUMBER OF ROOT PENETRATIONS THROUGH COBBLE BIOBARRIERS AS A FUNCTION OF SOIL THICKNESS, BIOBARRIER THICKNESS, AND ELAPSED TIME SINCE SEEDING. SAMPLE SIZE FOR EACH COLUMN IS 12, WITH PERCENTAGE OF PENETRATIONS IN PARENTHESES

| Elapsed Time<br>(days) | Soil-Barrier Thickness Combinations<br>(cm) |        |        |        |        |         |
|------------------------|---------------------------------------------|--------|--------|--------|--------|---------|
|                        | 30-30                                       | 60-30  | 30-60  | 60-60  | 30-90  | 60-90   |
| 53                     | 4 (33)                                      | 0 (0)  | 0 (0)  | a      | a      | a       |
| 72                     | 5 (42)                                      | 3 (25) | 3 (25) | 1 (8)  | 1 (8)  | 0 (0)   |
| 77                     | 7 (58)                                      | 3 (25) | 3 (25) | a      | a      | a       |
| 101                    | 9 (75)                                      | 4 (33) | 3 (25) | 3 (25) | 2 (17) |         |
| 119                    | 9 (75)                                      | 4 (33) | 3 (25) | 3 (25) | 3 (25) | 2 (17)  |
| 148                    | 9 (75)                                      | 5 (42) | 5 (42) | 3 (25) | 3 (25) | 2 (17)  |
| 172                    | 9 (75)                                      | 6 (50) | 7 (58) | 3 (25) | 3 (25) | 2 (17)  |
| Never<br>Penetrated    | 3 (25)                                      | 6 (50) | 5 (42) | 9 (75) | 9 (75) | 10 (83) |

\*Samples not taken.



**TABLE 7. CUMULATIVE NUMBER OF ROOT PENETRATIONS THROUGH COBBLE-GRAVEL BIOBARRIERS AS A FUNCTION OF SOIL THICKNESS, BIOBARRIER THICKNESS, AND ELAPSED TIME SINCE SEEDING. SAMPLE SIZE FOR EACH COLUMN IS 12, WITH PERCENTAGE OF PENETRATIONS IN PARENTHESES**

| Elapsed<br>Time<br>(days) | Soil-Barrier Thickness Combinations<br>(cm) |        |        |        |        |        |
|---------------------------|---------------------------------------------|--------|--------|--------|--------|--------|
|                           | 30-30                                       | 60-30  | 30-60  | 60-60  | 30-90  | 60-90  |
| 53                        | 4 (33)                                      | 0 (0)  | 1 (8)  | a      | a      | a      |
| 72                        | 4 (33)                                      | 2 (17) | 4 (33) | 4 (33) | 1 (8)  | 0 (0)  |
| 77                        | 5 (42)                                      | 3 (25) | 4 (33) | a      | a      | a      |
| 101                       | 5 (42)                                      | 3 (25) | 4 (33) | 4 (33) | 2 (17) | 1 (8)  |
| 119                       | 7 (58)                                      | 4 (33) | 4 (33) | 4 (33) | 3 (25) | 3 (25) |
| 148                       | 8 (67)                                      | 5 (42) | 4 (33) | 4 (33) | 3 (25) | 3 (25) |
| 172                       | 9 (75)                                      | 6 (50) | 4 (33) | 4 (33) | 3 (25) | 3 (25) |
| Never<br>Penetrated       | 3 (25)                                      | 6 (50) | 8 (67) | 8 (67) | 9 (75) | 9 (75) |

<sup>a</sup>Samples not taken.

**TABLE 8. COMPARISON OF LABORATORY AND FIELD MEASUREMENTS OF THE UNSATURATED HYDRAULIC CONDUCTIVITIES OF CRUSHED TUFF**

| Laboratory                   |
|------------------------------|
| $K = 0.0054 \theta_v^{7.62}$ |
| Field                        |
| $K = 0.0037 \theta_v^{7.55}$ |

K = Hydraulic Conductivity in m/s.

$\theta_v$  = Fractional Water Content by Volume.

For example, 172 days after seeding, 97% of the crushed tuff barrier systems had been intruded by plant roots while corresponding values for clay, cobble, and cobble-gravel, respectively, were 44%, 58%, and 60%. Thus, the latter 3 barrier systems resulted in 2 times more protection against root intrusion than the sandy backfill during 1 growing season. Increasing soil and barrier thickness by factors of 2 and 3 greatly improved performance of the clay, cobble, and cobble-gravel root barrier systems (Tables 5-7). Maximum soil-barrier thickness combinations for the latter three barrier materials generally reduced root intrusions to less than 25%.

Based on a log-linear contingency table analysis, the type of vegetation, biobarrier material, and soil biobarrier thickness were all statistically significant factors ( $P < 0.05$ ) affecting the probability of penetration. In addition, the combination of vegetation and biobarrier material interacted significantly ( $P < 0.05$ ). Barley was much more effective in penetrating cobble and cobble-gravel systems than were alfalfa or sweet clover (Table 3).

Although clay, cobble and cobble-gravel barrier systems work equally well in this short-term experiment, some problems were encountered with the use of bentonite clay and cobble as barrier materials. Bentonite clay, which was saturated with water before use, was subject to shrinking caused by depletion of water from the clay. Visual observation on clear lucite lysimeters suggests that plant roots were the cause of this shrinkage.

Examination of profiles containing cobble barrier systems suggests that the large pore spaces between the rocks gradually fill with soil overburden. As such, cobble barrier systems may not be effective over long periods of time. The 2-cm-diameter gravel, which was placed over cobble in the cobble-gravel system, greatly retarded the rate of soil migration into the spaces between the cobble.

Each of the four 1.9-m-diameter culverts was placed on end and filled with soil and one of the four barrier materials to test the effectiveness of the barrier material in inhibiting animal burrowing with depth. One pocket gopher (*Thomomys bottae*) was placed into each culvert and allowed to establish a burrow system within the profile. Gophers were maintained through a four month period by supplemental feeding. Gophers were removed from the culverts before injecting the burrow system with an expanding polyurethane foam. After the foam had hardened, a three-dimensional map of the burrow system was obtained.

Results demonstrated that, under the conditions of this experiment, cobble, cobble-gravel, and bentonite clay were equally effective in preventing animal intrusion with depth. Crushed tuff barrier material, however, was readily used for tunneling and offered little resistance in burrowing activity.

For reasons discussed previously, bentonite clay would probably not be effective as an animal intrusion barrier because of the drying and shrinking of the clay material. Burrowing activities by these animals probably increases the rate of soil infiltration into the pore space in cobble barrier systems.

Although evidence, based on surface cast material, suggested that tunneling activity occurred in the gravel overlaying the cobble in the cobble-gravel barrier system, tunnels could not be maintained in this loosely aggregated material. The cobble, 7.5-12 cm in diameter, could not physically be moved by the gophers.

Based on these short-term, small-scale experiments, it appears that clay, cobble, and cobble-gravel systems greatly reduce both plant root intrusion and animal intrusion with depth. Based on visual observations in clear lucite lysimeters and lysimeters that were sacrificed for determining root mass profiles, cobble-gravel barrier systems appear to be the best candidates for further testing on larger scales. Attractive features of using a cobble-gravel barrier system include a long environmental life and cost effectiveness, assuming the availability of such materials or a similar analog. Based on waste burial operations at Los Alamos, estimates suggest that cobble-gravel barrier systems would cost about the same as the current method of using crushed tuff and soil as trench covering material.

Four critical questions related to the use of cobble-gravel barrier systems are:

- 1) performance of cobble-gravel intrusion barrier systems over extended time frames,
- 2) performance at field scale under natural precipitation regimes with native vegetation,
- 3) performance under various degrees of subsidence, and
- 4) effect of intrusion barrier on water balance (percolation into the cover profile).

Current studies will address all four of these issues. For example, the intermediate scale plots at Area G (Fig. 2), the Laboratory's operational low-level waste site and the caisson experiment at the EETF address the effects of the rock barrier systems on moisture balance. Experiments now underway at the EETF will evaluate 4 different levels of subsidence on intrusion barrier effectiveness.

The field scale intrusion barrier experiment at the Laboratory's operational low-level waste site will be monitored continuously to

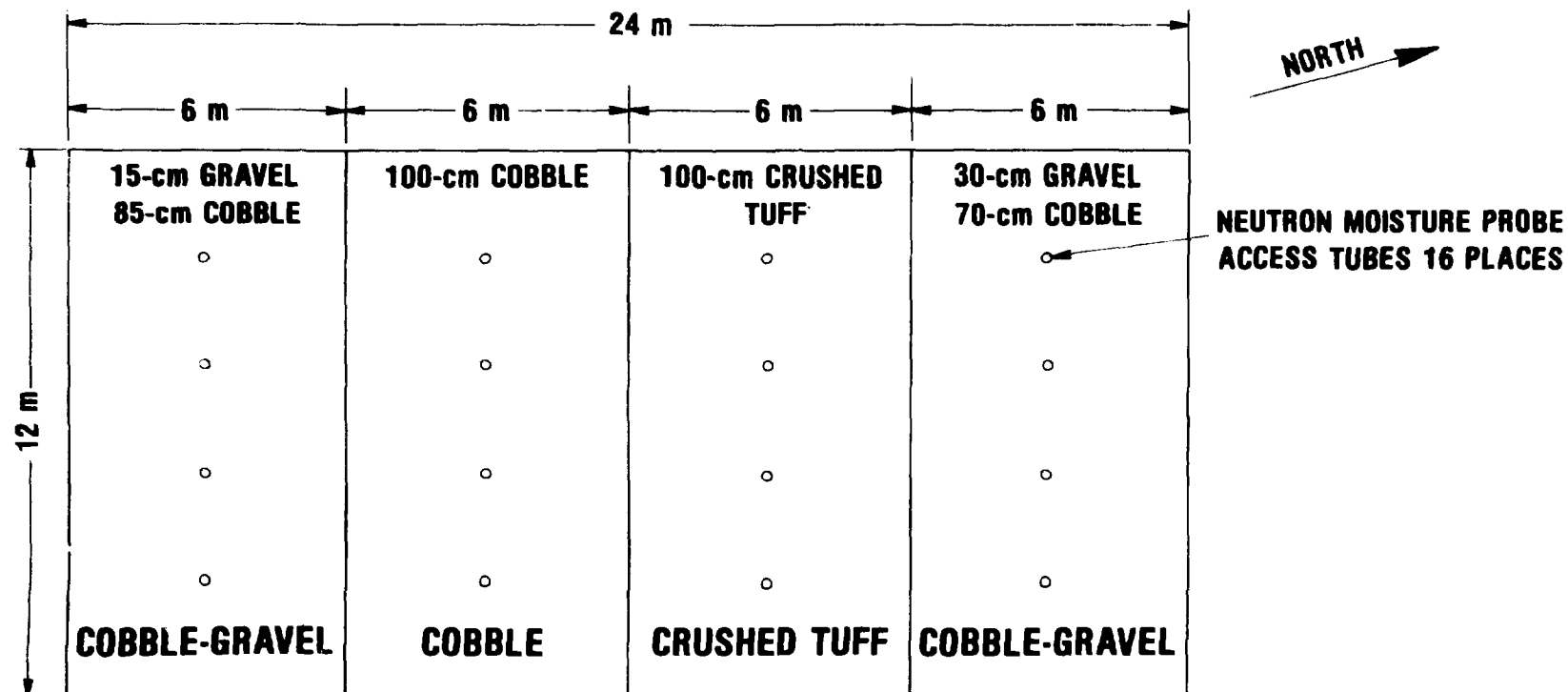


FIGURE 2.

**INTERMEDIATE SCALE BIO-INTRUSION BARRIER EXPERIMENT**  
**AREA G LOS ALAMOS NATIONAL LABORATORY**  
 15-cm TOPSOIL      100-cm BIO-INTRUSION BARRIER

determine effectiveness of the barrier systems in preventing plant root and burrowing animal intrusion. In addition, soil moisture measurements beneath the cover profile will be used to evaluate the effect of the barrier systems on water infiltration. Additional experiments at the operational low-level waste site will be conducted and employ state of the art cover designs as new data are acquired at the EETF. For example, based on CREAMS simulations we are now able to recommend topsoil depths that reduce the probability of water being available for infiltration through the intrusion barrier and into the waste. The next experiment on a newly closed trench at Area G will incorporate the improved cover design to prevent biointrusion and water infiltration by optimizing topsoil and barrier configurations.

The intermediate-scale intrusion barrier studies in the Experiment Clusters at the EETF (see Fig. 4 in Migration Barrier Testing section) are being monitored through time to determine barrier effectiveness and soil-moisture status beneath the cobble-gravel and crushed tuff barrier systems. The scaling factor distinguishes the caisson experiment from the large-scale Area G experiment.

The critical information needs for designing cover systems that minimize water and biota associated problems at low-level waste sites relate to plant and animal effects on the water balance. The overwhelming importance of vegetation in controlling the water balance indicates the need to develop vegetative schemes that fully exploit the benefits of vegetation in minimizing infiltration, runoff, and erosion. Revegetation schemes at low-level waste sites do not presently incorporate the best technology, nor do they recognize the strong interactions between plants and the water balance. Studies are needed to expand on the fragmentary knowledge of soil-plant-water relationships for native species that will eventually occupy the low-level waste site. This information will be essential for recommending what and when to plant in order to rapidly achieve a stable vegetation community that minimizes water related problems.

It has become increasingly obvious to us as we observe old low-level waste sites that the impact of animal burrowing on excavation of the waste and on the water balance has been greatly underestimated for some sites. At Los Alamos, old sites that are now being recovered as part of a remedial action program, are greatly disturbed by animals such as the pocket gopher. To provide a quantitative basis for our thesis, definitive studies need to be conducted to quantify the extent of burrowing animals' impact. An ideal experimental set up for such studies is the USDA erosion plots where the effects of burrowing activity on chemical transport and water balance could be measured very precisely.

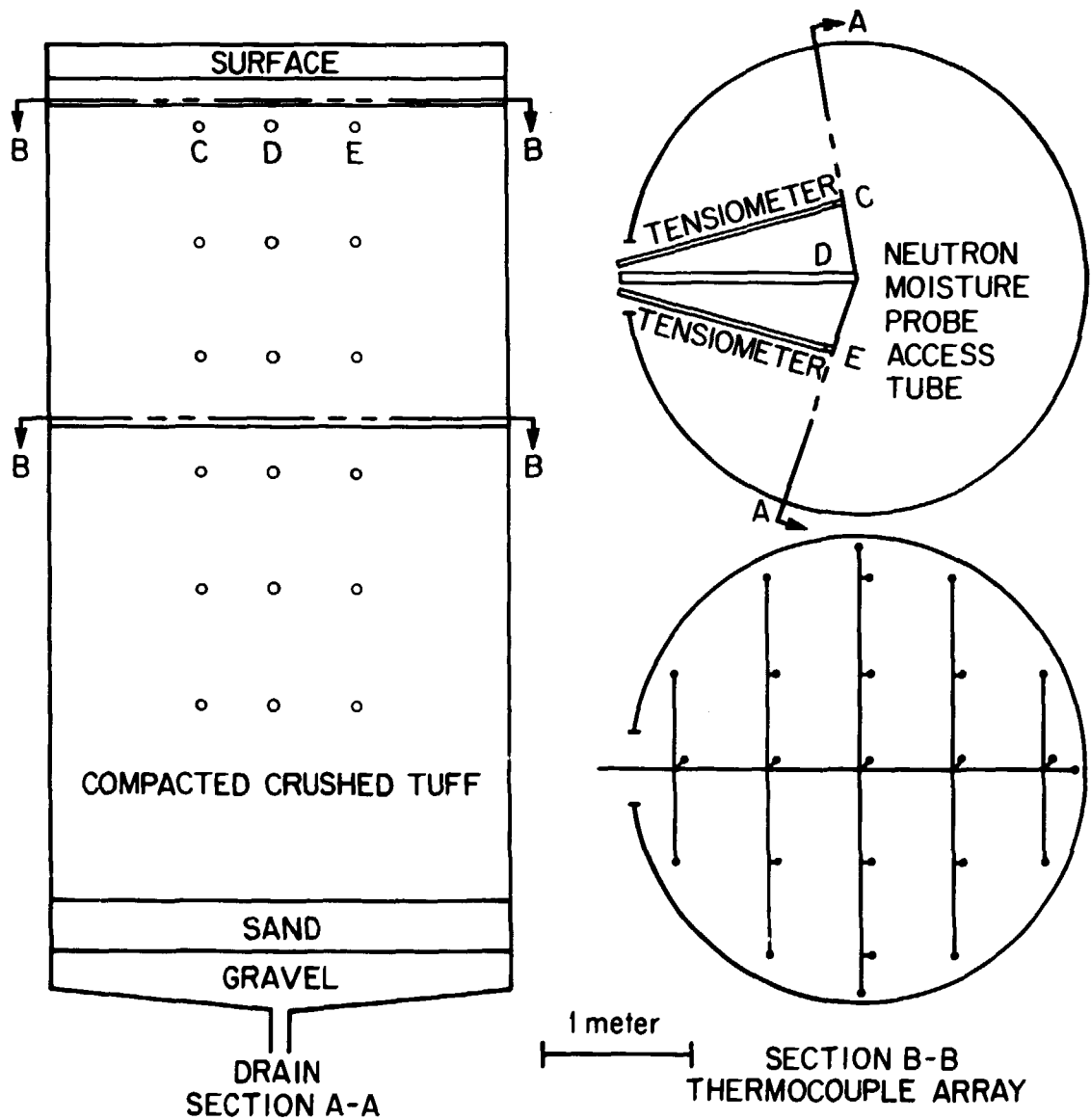


FIGURE 3. EXPERIMENTAL CONFIGURATION FOR MEASURING WATER AND CONTAMINANT MOVEMENT

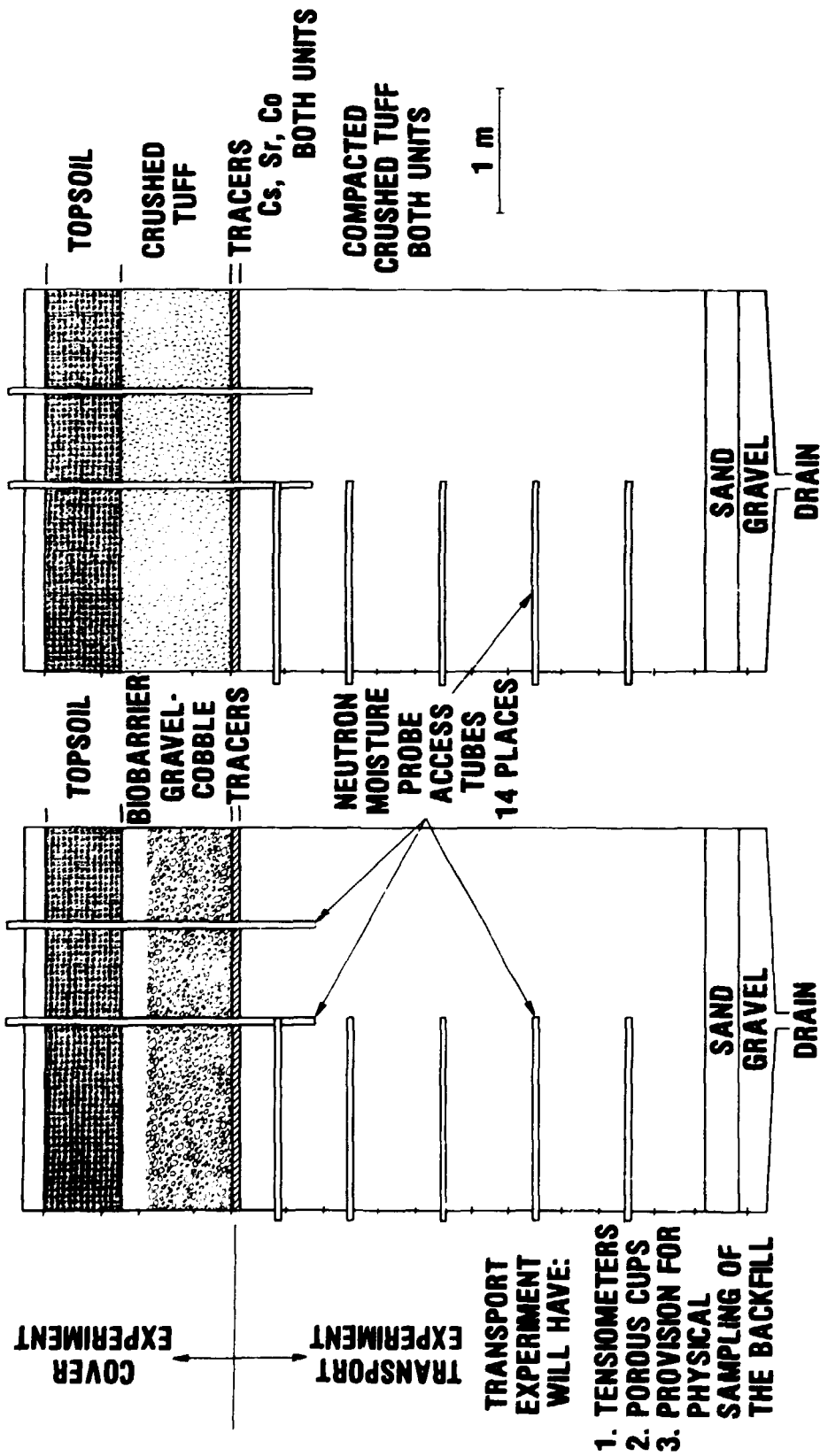


FIGURE 4.

# BIOINTRUSION BARRIER/TRANSPORT EXPERIMENT

Employing state of the art concepts in ecology for designing low-level waste sites has a high probability of resolving some of the major issues relative to long term stability. Our current thinking is that a suitable vegetation cover on low-level waste sites in arid systems might be an evergreen species such as piñon pine, juniper, or ponderosa pine, at Los Alamos, for example. This tree cover is a stable vegetation community, is relatively shallow rooted, uses moisture (transpires) throughout the year, greatly reduces erosion when a filter layer develops, and is an undesirable habitat for most burrowing animals such as the pocket gopher.

#### MIGRATION BARRIER TESTING

By a number of pathways water can infiltrate into a SLB facility and interact with the buried wastes. This interaction can result in the mobilization and subsequent transport of contaminants out of the burial facility and into the environment surrounding the facility. There does not exist adequate data from carefully instrumented large scale field experiments on the movement of water and contaminants under unsaturated conditions to enable a site operator to define and engineer suitable barriers to prevent the migration of radionuclides out of a SLB facility. Also, the mechanisms of leaching and transport under unsaturated conditions are not well known. Before the need for a migration barrier can be determined and the specification of a migration barrier can be made, the data and information mentioned above must be obtained.

The purpose of this subtask is to:

- 1) Measure in large scale field experiments the movement of water and contaminants under unsaturated conditions.
- 2) Determine from large scale field experiments the mechanisms of leaching and transport of contaminants under unsaturated conditions.
- 3) Evaluate methods for predicting water movement under unsaturated conditions based on laboratory experiments where possible to make unnecessary the large scale field experiments performed under this task for every soil type that may be considered for a SLB facility
- 4) Based on the results obtained in items 1 and 2 specify and field test barrier systems to prevent the movement of radionuclides out of a SLB facility into the surrounding environment.



5) Coordinate the results obtained in this subtask with the modeling effort underway in other subtasks.

During Fiscal Year 1982 extensive progress has been made on items 1 and 3 above. The first step is to measure the movement of water only under unsaturated conditions on a scale typical of that encountered in SLB. This experiment was performed in the Experiment Clusters in the EETF. Photographs and drawings of these units have been previously published (1).

The experimental configuration and instrument emplacement is shown in Fig.3. The horizontal neutron moisture probe access tubes extend 1.5 m into the soil column. These tubes are aluminum with a 50.8 mm outside diameter. The thermocouples are copper-constantan and have all been calibrated at the ice point and at the boiling point of water at Los Alamos elevation, approximately 93°C. The two large thermocouple harnesses are emplaced to determine the temperature variation across the soil column at two elevations. The other thermocouples are to determine the temperature as a function of depth in the column. The thermocouple harnesses were the only instruments emplaced during the filling operation. All other instruments are emplaced after the units were backfilled and completed.

For these experiments, two of the 3-m-diameter by 6-m-deep caissons were filled with crushed Bandelier Tuff. This required the preparation, delivery, loading, and compaction of approximately 80 m<sup>3</sup> of materials. Because of this large volume, the personnel and equipment of the Zia Company at Los Alamos were used in the filling procedure. The tuff used can be described geologically as a volcanic ash flow composed mostly of silicic glass and having a grain size distribution close to that of a silty sand.

Front end loaders were used to mine the tuff, which was then crushed by running the front end loader over it repeatedly. The crushed material was then loaded into trucks and hauled to the Zia company batch cement plant where it was screened. All material that passed through a 12.7 mm screen was stockpiled for mixing with water. Known amounts of water were added to the crushed and screened material in the batch plant. The moist tuff was then loaded into cement trucks for delivery to the experiment.

In the filling process with the finely crushed Bandelier Tuff, great care was taken to work with tuff that was homogeneously mixed with water, trying to approach, but staying below, the optimum water content. This optimum water content is the amount of mixing water for a given tuff and compaction process which will give a maximum of dry tuff mass per volume. The tuff was dynamically compacted with "jumping jacks." The actual dry density obtained ranged from 91.3 to 101.4% of what is considered optimum dry density for crushed tuff, indicating that the amount of moisture used, mostly between 10 and 13%

by mass, is close to the optimum moisture plateau. Water content, dry density, and wet density were measured at several heights during the filling operation by a Portaprobe Model MC-2 Percent-Test Nuclear Meter. Results of these measurements will be discussed later. The crushed tuff was loaded into each caisson in an amount to give a compacted layer of from 15 to 20 cm, and then compacted as described above.

Compaction was necessary to minimize, if not eliminate, future subsidence which would be detrimental to the knowledge of the location of the instruments used to make the measurements necessary for the instantaneous profile method. Also, subsidence might cause deformation in the horizontal neutron moisture probe tubes. Compaction will decrease the saturated hydraulic conductivity, but more important, increase the unsaturated hydraulic conductivity known to exist at the moisture contents normally prevailing in the Bandelier Tuff. Hydraulic conductivity decreases sharply with increasing soil moisture tension. This is most evident in materials with coarse pores and/or low packing and less so in materials with fine pores and/or high packing. Somewhere the two curves will intersect; the worst case is if highly packed material is used at the low moisture conditions that are normal in the Bandelier Tuff because higher unsaturated conductivities will take place at finer pore size distribution, provided the degree of unsaturation is high enough.

To obtain the homogeneity and high density required for the experiments, the wet tuff had to be compacted with minimum interference from emplaced instruments. The only exceptions were the two thermocouple harnesses which were carefully emplaced and covered with crushed tuff, with careful compaction in the next layer to prevent damage.

Immediately after filling was completed, emplacement of the horizontal neutron moisture probe access tubes was started. The horizontal tubes are about 1.5 m long with an aluminum cap welded on the end. Thus, a hole in the tuff was necessary in which to place the tube. The hole was made by driving a stainless steel tube of the same diameter with a sharpened end into the compacted material using a 4.5 kg sledge hammer.

The hole was driven in about 100 mm increments and the tube removed and emptied. The tubes were maintained horizontal to within one degree. In the portions of the experiment where the tuff was the wettest, emplacement was practically impossible by this method, whereas, in the drier material, this method worked quite well.

The tensiometers were emplaced by augering holes and then inserting the tensiometers. The other thermocouples were emplaced by drilling a small hole and then inserting the thermocouple.

When instrument emplacement was completed, the experiment was started. A layer of water about 6 cm thick was ponded on the surface of the experiment. Ponding was begun on May 11, 1982. This was continued until June 7, 1982. During this time the column became saturated, the tensiometers filled from the inside, and the saturated hydraulic conductivity of the crushed Bandelier Tuff was measured to be  $1.44 \times 10^{-6}$  m/sec. On June 7, 1982, the water flux onto the experiment was stopped and the surface was tightly covered with plastic to prevent any further infiltration of water or evaporation of water from the experiment. The experiment was allowed to drain until August 26, 1982, at which time another experiment was started.

During drainage, simultaneous measurements were made of water content using the neutron moisture probe and soil water matric potential using the emplaced tensiometers. Based on these experimental measurements, the unsaturated hydraulic conductivity of the crushed Bandelier Tuff was measured as a function of water content using the instantaneous profile method (4). Also, the moisture characteristic curve for the material was made from the data collected. Both of these measurements were made in moisture content regimes where they are difficult to make under laboratory conditions.

The experimental results are summarized in Table 8. The field measurements were made at higher water contents than were the laboratory measurements and the agreement on the water content dependence of the unsaturated hydraulic conductivity is excellent. Numerically, for the same water content, the laboratory and field measurements agree to within a factor of about 1.5, which is excellent agreement.

Preliminary application of the Campbell method for predicting unsaturated hydraulic conductivities based on the matric potential curve were disappointing. Further work is in progress with other predictive methods and the Campbell method will be tried again when all the data are available from the experiment.

Future work will be directed to the other purposes of this subtask which were described earlier. The water movement experiments are well underway and the emphasis will be on experiments to determine the mechanism of leaching and transport under unsaturated conditions. When these data are available, migration barrier needs can be specified, and migration barriers tested in large scale field experiments.

Two types of experiments will be performed as part of the NLLWMP on the movement of contaminants. In one experiment, illustrated in Figure 4, the solubilization and movement of elements representative of low-level waste, Cs, Sr, and Co, emplaced as salts will be measured. Enough water will be added to the experiment to move water through the layer of tracers while at the same time maintaining

unsaturated conditions. The movement of water past this interface will be monitored with the tensiometers and neutron probe, and aqueous samples will be drawn using the water samplers emplaced as shown in the Figure. This experiment is completely emplaced.

In the second experiment, typical breakthrough curve experiments will be performed in Unit B with the same experimental configuration shown in Figure 3. This is a joint experiment with Dr. Herman Bouwer of the U.S. Water Conservation Laboratory in Phoenix, Arizona.

#### GROUND AND SURFACE WATER MANAGEMENT SYSTEM TESTING

By a number of pathways, water can infiltrate into a SLB facility and interact with the buried wastes. This interaction can result in the mobilization and subsequent transport of contaminants out of the burial facility and into the environment surrounding the facility. This water either comes in from the surface through the trench cover or comes in horizontally through the sides of the trenches. Since disposal in formations where the water table is near the level or above the bottom of the trenches will probably not be allowed in the future, the source of ground water considered here is that which can come in through the side of the trenches.

The purpose of this subtask is to field test in large scale experiments of natural materials combined in such a way as to use the natural hydrologic characteristics of the materials to direct the movement of water in a known and predictable manner. Plants can also be considered to be part of a system to remove water from the cover of a SLB facility if there is infiltration.

Small scale modeling (5) has demonstrated that it is possible to maintain dry structures in porous media by using capillary barriers. These barriers are created by differences in particle size distribution no matter what the separation interface consists of. This phenomenon is due to the predominance of suction over gravity forces. The percolating liquid will only penetrate the coarser material after the overlying finer one is totally saturated. Consequently, the structure is maintained dry, even under a permanent regime of non-saturating infiltration. As long as the matric potential at the coarse-fine interface remains negative, theoretically, water infiltrating into the finer layer will not cross the interface but will flow laterally within the finer layer until percolation reoccurs upon reaching the edge of the coarser layer. The limiting particle size differences beyond which this phenomenon ceases to exist have been determined; Corey and Horton (6) referred to it as the "wick effect." They found that, under unsaturated conditions, a gravel lens caused lateral flow in a finer-textured overlying material. The lateral distance over which the water can be transported is limited and is also a function of the slope of the interface.

At a particular matric potential, all pores which are sufficiently small to exert a capillary suction equal to or greater than the matric potential will be filled with water. Consequently, a fine-grained medium will contain more water than a coarse grained one at equal matric potentials. During drainage, an initially saturated soil will drain its largest pores first, while in wetting a dry soil, the smallest pores will fill first. As the matric potential is allowed to increase (getting less negative), the water content increases as progressively larger pores are filled with water. This will be accompanied by an increase in hydraulic conductivity that would tend to increase horizontal diversion of water already promoted by the presence of the sloping interface. The coarser medium, on the other hand, with its predominantly larger pores, will remain dry with a low hydraulic conductivity until the matric potential reaches zero at the interface. The system will fail when the matric potential at the interface becomes non-negative.

There has only been one field experiment to examine the performance of wick systems (7). This experiment showed, in general, that the concept was valid on a field scale.

By relying on the differences in water content at a given matric potential for natural materials with a different particle size, the movement of water under unsaturated conditions can be controlled. Properly selected plants can also be part of a system to manage ground and surface water. For more humid sites, a specialized geotextile may serve as a wick system.

The experimental configuration used in the first wick system experiments emplaced at Los Alamos is shown in Figure 5. For both experiments the coarse grained material is gravel with about 2.5 cm diameter. In one experiment, the fine material is the same crushed Bandelier Tuff used in the Migration Barrier experiments described in the previous section. In the other, it is the same topsoil used in the upper portions of the experiments in caissons C and D also described in the previous section. Tensiometers were placed just above the interface as shown in the Figure. Also, in this experiment, a geotextile was placed between the fine and coarse grained materials. Its sole purpose is to prevent the interpenetration of the fine and coarse grained materials to keep a sharp interface between the material properties. The neutron moisture probe access tube is positioned to allow neutron probe measurements to detect failure of the wick systems.

Both wick systems were constructed and allowed to settle to their final configuration. Initially, natural rainfall only was applied to the experiments. Then, a schedule of gradually increasing regular amounts of precipitation was artificially applied to the experiments. Daily measurements were made of the soil matric potential measured by the tensiometers and the water content using the neutron moisture probe.

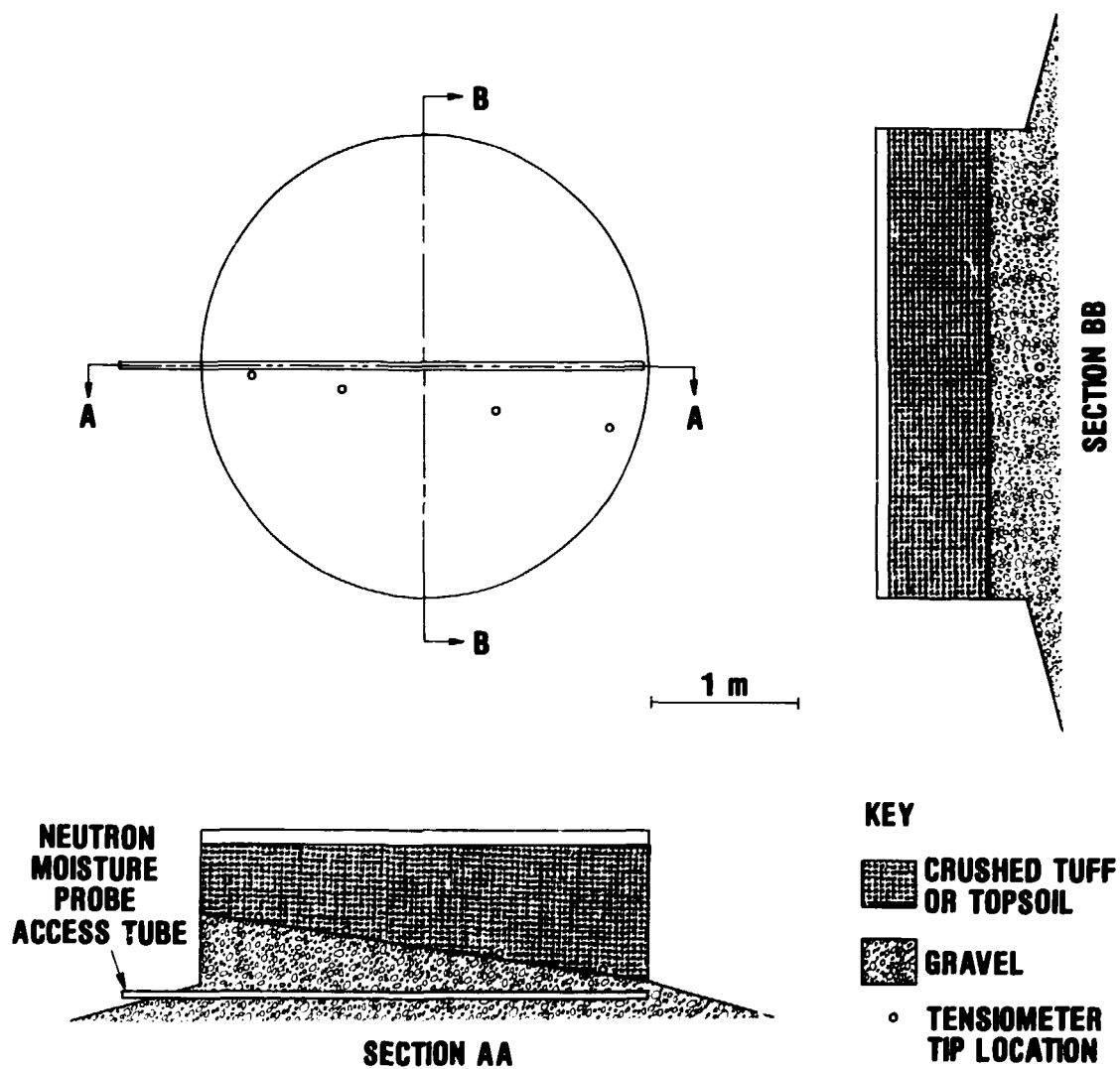


FIGURE 5.

**WICK EXPERIMENT**

## SUMMARY OF RESULTS TO DATE

Problems were encountered on the wick system using the topsoil. This particular material tended to crack after moisture was applied, creating channels for the water to move directly to the interface, which resulted in the failure of the system to operate as expected. The tuff-gravel system, however, performed extremely well. Under natural precipitation conditions the system did not fail. Table 9 summarizes the tensiometer and neutron probe readings taken regularly after the application of 5 cm of water by ponding on the surface. Water was applied by calculating the required number of gallons for a 5 cm rainfall, and then pumping this much water onto the system.

The data in Table 9 show how the tuff-gravel system performed after the application of a 5 cm rainfall. Starting at the high end of the slope the water slowly moved down toward the low end of the wick. This is evidenced by the decreasing tensiometer readings with time going from the high to the low end of the slope. In addition, the neutron probe measurements did not indicate failure of the system at any point.

Natural material wick systems are capable of moving water in a controlled and predictable manner if care is used in choosing the materials used. Since the experimental results are just starting to become available from these experiments, it is difficult to make any recommendations at this time.

The experiments described above will be continued by adding increasing amounts of water until the system fails and in this way the failure mode will be determined. Experiments will be performed in which the slopes are different from those in the current experiments. Also, in the integrated systems testing, which will be started in Fiscal Year 1983, wick systems will be tested in combination with other system components such as biobarriers and erosion control configurations. As more experimental results become available, an experiment will be designed and emplaced to look at the fine details of the performance and failure modes of wick systems. Wicks consisting mainly of geotextiles will also be tested.

The generic program technical needs which have not been satisfactorily addressed are:

- 1) effects of subsidence on wick system performance,
- 2) performance of wick systems as part of a complete trench cover,
- 3) the problem of how to remove the water from the trench environment once the wick diverts it away from the top of the trench,

TABLE 9. SUMMARY OF WICK EXPERIMENT

(Matric Potential in KPa)

|                    | Before<br>Flooding | Time After Flooding<br>with 5 cm Water<br>(in hours) |      |      |      |      |
|--------------------|--------------------|------------------------------------------------------|------|------|------|------|
|                    |                    | 5                                                    | 6    | 7    | 49   | 72   |
| <b>Tensiometer</b> |                    |                                                      |      |      |      |      |
| 1                  | 29.0               | 27.5                                                 | 23.0 | 19.0 | 15.0 | 13.0 |
| 2                  | 26.0               | 27.2                                                 | 27.5 | 24.0 | 16.0 | 12.5 |
| 3                  | 25.5               | 27.0                                                 | 27.0 | 24.5 | 18.0 | 14.5 |
| 4                  | 26.5               | 27.5                                                 | 27.0 | 25.0 | 19.5 | 15.0 |
|                    |                    | <b>Volumetric Water Content - %</b>                  |      |      |      |      |
| <b>Station</b>     |                    |                                                      |      |      |      |      |
| 1                  | 9.7                | 9.8                                                  |      | 10.2 |      | 10.4 |
| 2                  | 10.0               | 10.0                                                 |      | 9.9  |      | 10.3 |
| 3                  | 8.8                | 9.2                                                  |      | 9.1  |      | 9.4  |
| 4                  | 6.7                | 7.1                                                  |      | 6.8  |      | 7.2  |
| 5                  | 6.5                | 6.6                                                  |      | 6.7  |      | 6.6  |
| 6                  | 5.4                | 5.5                                                  |      | 4.8  |      | 5.3  |



i.e., how to prevent the diverted water from running back into the burial pit along the pit wall host material interface,

4) applicability of wicks as a function of increased amounts of total rainfall annually,

5) the role of geotextile wicks in SLB, and

6) adequate modeling of wick systems.

All of these needs will be addressed in work to be done in Fiscal Year 1983 and 1984 at Los Alamos.

#### MODEL VERIFICATION

The modeling work reported for this subtask was done by Dr. Bryan J. Travis of Group ESS-5 of the Los Alamos National Laboratory. These calculations were done with the WAFE code which is briefly described in Table 10. The calculations reported here were on the drainage experiment described in the Migration Barrier section of this paper. Material properties, initial moisture content, matric potential curves, and the calculated moisture dependence of the hydraulic conductivity were all used as input data. The results of the calculations and a comparison with the experimental results are shown in Figures 6, 7, and 8. Figure 6 compares the calculated and measured drainage at the bottom of the experiment. Figure 7 shows the water content profile after eight days of drainage, while Figure 8 is a similar plot after 32 days of drainage.

**TABLE 10****WAFE**

- MASS AND HEAT TRANSPORT IN POROUS MEDIA**
- MULTI-PHASE, MULTICOMPONENT FLOW**
- TIME PLUS TWO DIMENSIONS**
- NOT RESTRICTED TO DARCY FLOW**
- VARIABLE MATERIAL PROPERTIES**
- TIME DEPENDENT SOURCES, SINKS, BOUNDARY CONDITIONS**
- BUOYANCY**
- NUMERICALLY IMPLICIT**

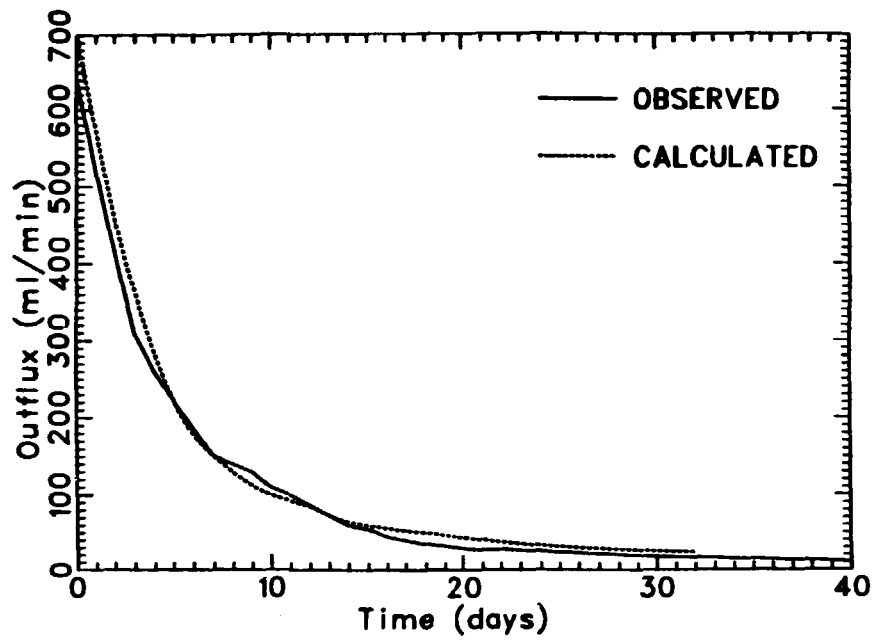


FIGURE 6.

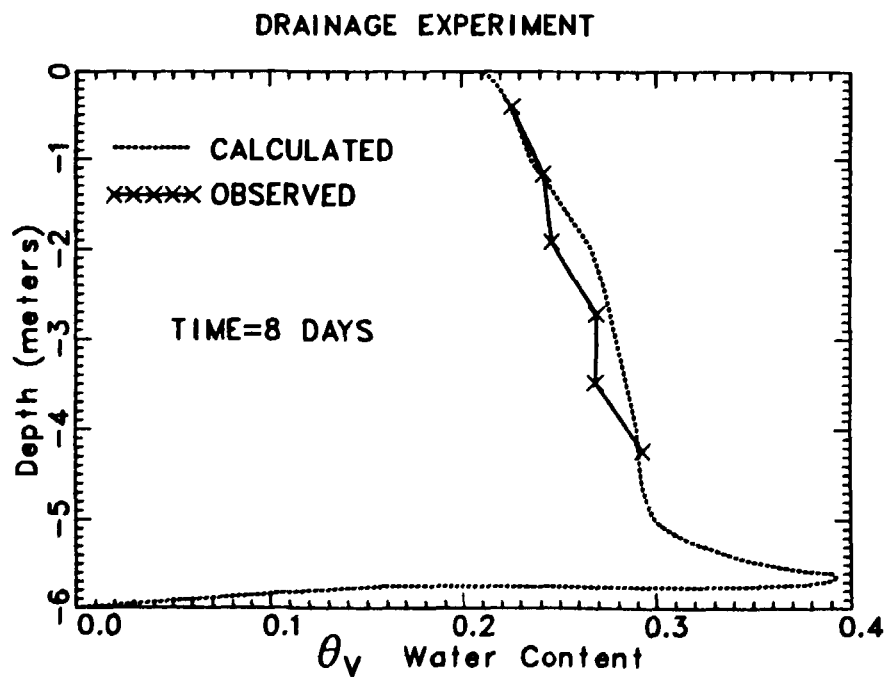


FIGURE 7.

**DRAINAGE EXPERIMENT**

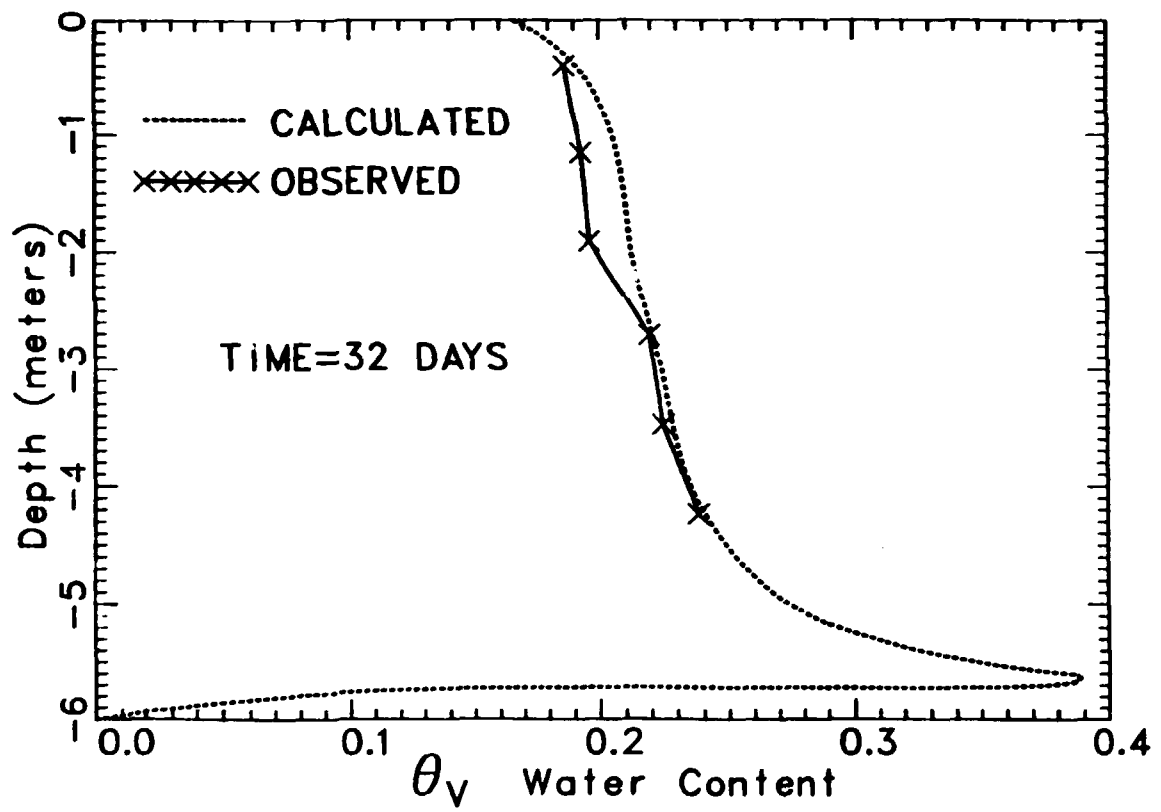


FIGURE 8.

DRAINAGE EXPERIMENT

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## TRENCH DESIGN AND CONSTRUCTION TECHNIQUES FOR LOW-LEVEL RADIOACTIVE WASTE DISPOSAL

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### ABSTRACT

Problems encountered with the disposal of low - level radioactive waste by shallow land burial have brought about the realization that a reevaluation of disposal trench design and construction techniques is needed. Subsidence, the "bathtub effect", and the infiltration of groundwater into the disposal trench are among the problems that have cast doubts on the ability of some facilities to control the movement of their buried radionuclides. They have also made the goal of total passive maintenance unattainable in the near future.

The Nuclear Regulatory Commission and the U. S. Army Corps of Engineers, Waterways Experiment Station (WES) have entered into an agreement whereby the WES is preparing a document which will detail the critical parameters involved with the design and construction of low - level radioactive waste disposal trenches. These parameters will include those which should be controlled through the issuance of specifications or license restrictions in order to ensure that the facility will function as expected. This paper outlines the scope of the work, the work accomplished to date, the work remaining, and the basic philosophy behind the approaches taken in the preparation of this document.

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### INTRODUCTION

The disposal of solid low - level radioactive waste by shallow land burial has been practiced in the United States since the early 1940's. Early burial operations were conducted using burial techniques common to non-radioactive solid waste disposal. However, problems such as subsidence, burrowing animals, the "bathtub effect", erosion, and penetration of the trench cover by deep rooted plants have occurred at some disposal facilities. These problems have cast doubts on the ability of some facilities to control the migration of their buried radionuclides. This, coupled with a growing concern over the long-term environmental consequences has resulted in a reevaluation of the design and construction criteria for low - level radioactive waste disposal trenches.

Within the performance objectives of the proposed Draft 10 CFR Part 61, two time frames are mentioned relating to the Nuclear Regulatory Commission's goal of assuring protection of the public's health and safety: the short-term operational phase and the long-term period after operations cease. The short-term concerns itself with the protection of the workers and the general population during the operation of the facility. The long-term period involves the length of time after burial operations have ceased that the low - level radioactive waste (LLW) still possesses the potential to harm the health and safety of the general population. It is also stated that the long-term performance of the LLW disposal facility should receive greater emphasis than short-term conveniences and considerations since protecting the public's long-term health and safety is more important. The goals necessary to achieve this (as stated in Draft 10 CFR Part 61) are: (1) protection of the general public from potential releases to the environment; (2) stability of the disposed waste and the site to eliminate the need for ongoing maintenance of the site following closure; and (3) protection of individuals from inadvertent intrusion into the waste. Inadvertent intrusion is described as coming in contact with the waste through activities related to the normal usage of the land such as farming or home construction. Therefore, a LLW disposal trench must not only guard against potentially harmful releases of radioactivity into the environment but its design must require no active maintenance after closure and also protect the inadvertent intruder.

The extent and complexity of a LLW trench design depends on a variety of site-specific characteristics and requirements. There can be no single "best" trench design which would apply everywhere in the United States. The design must be matched to the conditions imposed upon it. For example, a trench design which is adequate for a humid region might prove oversized for an arid region. Therefore, the site's geotechnical, geohydrologic, and environmental factors must be considered on a site specific basis to efficiently design and construct a trench system adequate to control the migration of radionuclides.

It is with this guidance and these parameters in mind that the U. S. Army Engineer Waterways Experiment Station is preparing this document whose ultimate end product is to produce recommended trench designs given specific geologic and environmental conditions.

#### SCOPE OF WORK

The Waterways Experiment Station has been asked to provide technical assistance to the U. S. Nuclear Regulatory Commission (NRC) by preparing a document which details those critical parameters involved with the design and construction of low - level radioactive waste disposal trenches. This assistance to the NRC was divided into four major tasks.

##### Task 1 - Identify Practices Currently In Use

Task 2 - Assess Feasibility of Using Existing Practices

Task 3 - Assess Engineered Improvements

Task 4 - Develop Review Procedures and Industry Guidance

The statement of work for the Task 1 report involved an assessment of the state-of-the-art practices on not only the disposal of low - level radioactive waste but also sanitary and hazardous waste. These practices were then to be evaluated with respect to waste containment, surveillance, maintenance requirements, ease of construction, and overall performance.

The work involved in preparing the Task 2 report involved assessing the feasibility of using the practices identified in Task 1 at some of the existing disposal sites. These existing disposal sites were to include, but not be limited to, one of the two commercial sites located in the western United States, the Barnwell site, and one of three inactive commercial sites located in the eastern United States. This report was to highlight the advantages and disadvantages of each disposal technique reviewed and contain recommendations of preferred trench designs and construction techniques for use in distinct geotechnical/environmental settings.

The Task 3 document presents various engineered modifications designed to improve the construction and performance of a LLW disposal trench. This report assesses performance, cost, and safety among other aspects.

Using the information developed in Tasks 1-3, the WES will prepare a document detailing the critical parameters involved in the LLW trench design, construction techniques, and engineered modifications which should be controlled through the issuance of specifications or license restrictions. Also, this document should specify the recommended trench design, construction techniques, and engineered modifications given a specific geologic/environmental setting. This constitutes our Task 4 report.

#### WORK ACCOMPLISHED TO DATE

Presently, the Task 1 and Task 2 reports have been submitted to the Nuclear Regulatory Commission and have undergone review. The task three report has also been completed pending review by the NRC.

The Task 1 report involved an extensive literature review employing various computer reference data bases available through various government agencies and private organizations. The objective of this was to aid in formulating the present state-of-the-art practices in sanitary, hazardous, and low - level radioactive waste disposal. However, only those aspects of sanitary and hazardous solid waste disposal that were applicable to low - level radioactive waste disposal were discussed.



In general, the Task 1 report outlined the problems generally involved with LLW, reviewed the technical requirements for land disposal facilities as set forth in the proposed Draft 10 CFR Part 61 which affect trench design and construction, and presented the methods, materials, and equipment commonly employed in the solid waste disposal fields. Problem areas discussed included definition and classification of low - level radioactive waste, groundwater management, leachate and gas production, and site layout problems affecting the disposal trench. Each section of the proposed Draft 10 CFR Part 61 concerning site suitability requirements which affect trench design and construction and those sections involved with the disposal site design requirements were addressed. Following this, various methods of slope stability analysis were presented along with the many materials (natural and man-made) used to line or cover the disposal trench. Also included were the equipment, excavation practices, and installation procedures for the various construction materials used in the LLW disposal trench.

The Task 2 report reviewed and assessed the trench design and construction techniques used at the active and inactive sites visited. The advantages and disadvantages of the various techniques were presented as well as the feasibility of using some of the disposal practices outlined in Task 1. The intent of the Task 2 report was not to critique or judge the sites. Rather, it was to review what was being done and assess the benefits derived from a modification of that disposal practice or from a different disposal technique.

Descriptions of each site included a general site description, a geological description, the design method employed, and the operational methods used. Each of the sites selected represents a distinct geologic/environmental setting that requires a site specific analysis. However, the geological/environmental aspects of each of the sites can be likened to other sites in other areas of the country. Therefore, using this information, a general setting can be formulated for which the site would be a specific example. For the general geological/environmental settings addressed, certain preferred design and construction techniques were discussed.

The first two reports presented the design and construction techniques commonly used in the LLW disposal field and those areas where improvements could be made. The Task 3 report reviews those materials and material-usage concepts which could benefit the trench's performance. Also, some basic cost information is included to afford the reader some feel of how these methods and materials compare economically.

#### WORK REMAINING

After questions and comments generated from NRC's review of the Task 3 report have been received and accommodated, the Task 4 report will be the only remaining work undergoing preparation. The general approach behind the preparation of the Task 4 report will be to break

down the design and construction of the disposal trench into the different unit processes such as burial area preparation, trench layout, excavation-design and methods, drainage-design and construction, liners-design and installation, waste emplacement, backfill-materials and methods, and cover systems-design and construction. Keyed into these unit processes will be those site parameters which dictate what types of methods and materials will be required to ensure the trench will function as required. Information generated from operational experiences will also be included where applicable.

Information for these reports was obtained from interviews and briefings received from hazardous, sanitary, and low - level radioactive waste disposal facility operators, state and federal agencies involved in the construction and operation of these facilities' construction and operation, an assessment of the information obtained from the general reference material, research performed at several national laboratories and in the private sector, and suppliers of the various construction materials that have been used or show potential for use at a LLW disposal facility.

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SHALLOW LAND BURIAL TRENCH ISOLATION  
-DESIGN, CONSTRUCTION AND MONITORING

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ABSTRACT

The scope of this work covers an evaluation of trench cap designs, trench construction and trench loading by artificially speeding up the creation of void space in the trenches and monitoring structural and hydraulic effects under these conditions.

Eight trenches were construction and instrumented, four in a semi-arid region and four in a more humid mountainous region. The two main concepts being tested are the "soil arch" and the "soil beam".

Additionally, it is expected that methods will be developed for optimization of monitoring of sites as well as improved predictions of the relative long term stability of the tested designs.

Structural and hydrologic monitoring has only recently begun but preliminary results indicate that the "soil beam" is a viable concept even under more humid conditions.

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INTRODUCTION

It is universally agreed that the primary method of release of radio-nuclides to the biosphere from buried nuclear waste is through water movement. The most apparent scenario for shallow land burial sites for low-level nuclear waste is a trench cap collapse as waste decays and/or compacts followed by infiltration of surface water leading to contamination of surface and ground water.

During the past nine months the University of Arizona, under NRC contract has constructed eight (8) shallow land burial trenches of four different designs (1 set). One set of trenches is sited in the vicinity of Tucson at the University of Arizona Casa Grande Highway Farm (CGHF) and one set at the Mount Lemmon (ML) site. The former site is semi-arid (11 inches of rain/yr) while the latter site is mountainous (25 inches of rain/yr). The concept is to test the trenches at the same time under two different environments.

The scope of work covers an evaluation of trench cap designs. Waste decay and therefore trench loading were simulated in two ways. First by the placement of straw and second by artificially speeding up the creation of void space in the trench by construction of a collapsible platform. The monitoring of structural and hydraulic effects were undertaken under both of these conditions. Due to a number of reasons beyond the control of the research group the set of trenches at the ML site was not completed until May, 1982.

The principal research objectives are to develop trench designs that minimize surface subsidences and at the same time minimize the infiltration of surface water. To this end a program to monitor and measure water movement through the experimental trenches and to develop the procedures needed to implement a tracer monitoring program at a waste burial site were developed. In order to address these problems, various trench designs and backfilling procedures have been investigated, utilizing both controlled compaction and the use of geotextiles as an earth reinforcing material. In addition, different tracers and monitoring techniques are being utilized. From this experiment information regarding the amount and types of tracers that are most useful will be obtained as well as information about the number and placement of the sampling points needed to monitor the burial ground adequately under different environmental conditions.

## EXPERIMENT AND TRENCH DESIGNS

### EXPERIMENT DESIGNS

Two engineering concepts are being tested which may provide a solution to the problems of trench cap subsidence or collapse resulting from consolidation of waste materials and the formation of subsurface voids after burial. These concepts are the "Soil Arch" (Fig. 1) and the "Soil Beam" (Fig. 2).

The concept of the soil arch depends on the relative motion between the structure (trench cap), the adjacent soil, and the decaying waste. By considering the cap as a free body and analyzing its equilibrium following creation of a void beneath it, the following factors can be determined:

1. Whether or not the soil's shear strength properties are sufficient to maintain equilibrium in the cap backfill and therefore effectively cause the void created by waste degradation to be bridged by a soil arch.
2. The depth of cover required for all load (overburden plus applied) to be carried by the soil arch.
3. The maximum percentage of surface load that can be carried for any lesser depth of cover.

## SOIL ARCH CONCEPT

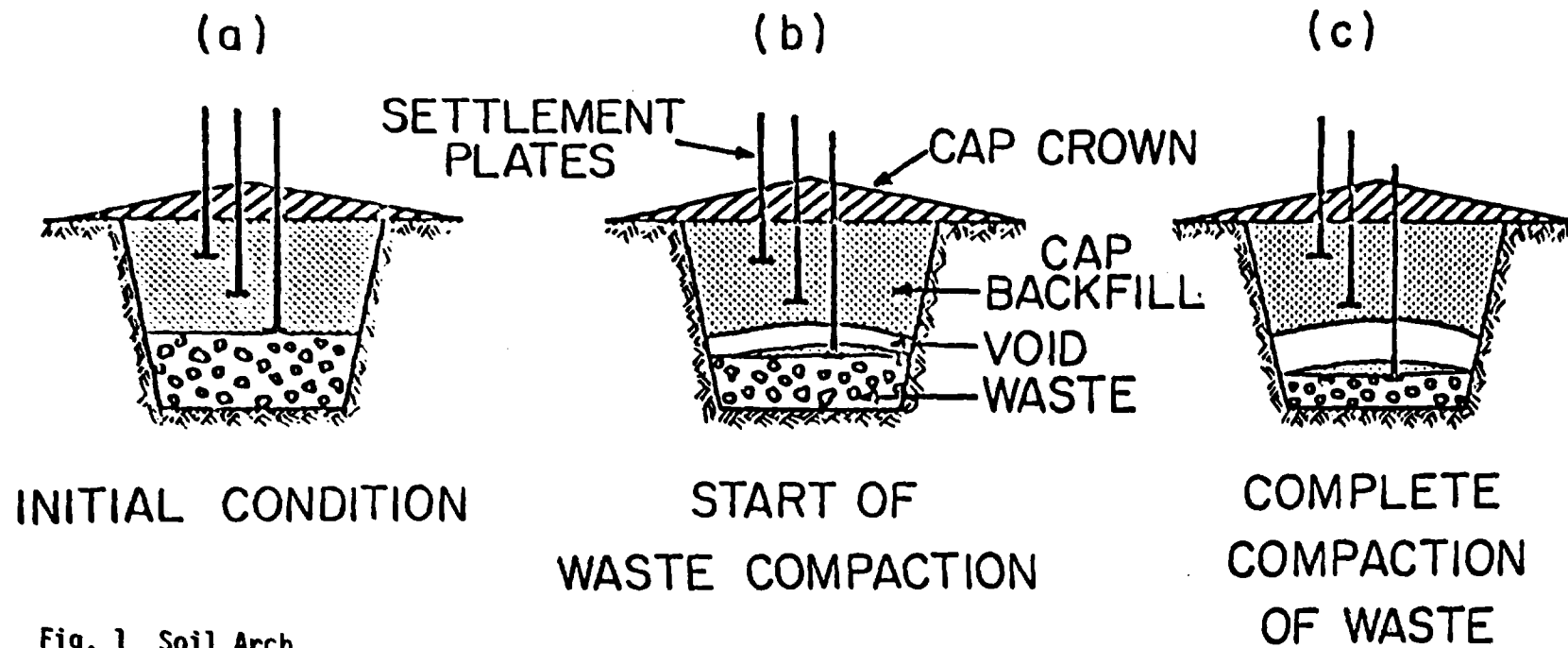
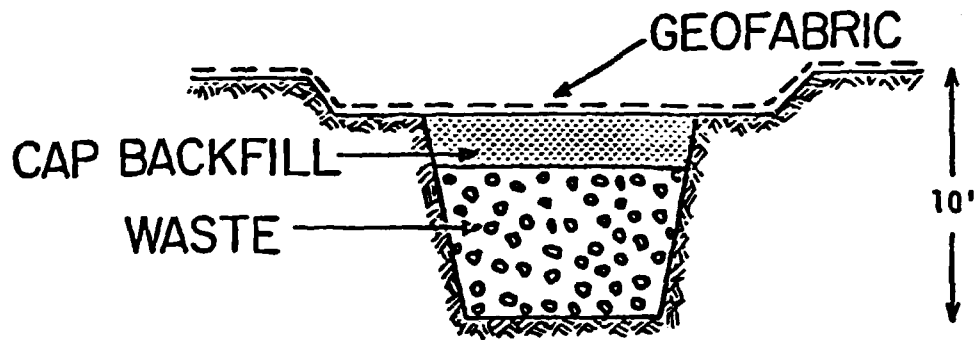


Fig. 1 Soil Arch

## GEOFABRIC SUPPORTED TRENCH CAP

### (a) CONSTRUCTION PHASE



### (b) COMPLETED TRENCH



### (c) TRENCH AFTER COLLAPSE OF WASTE

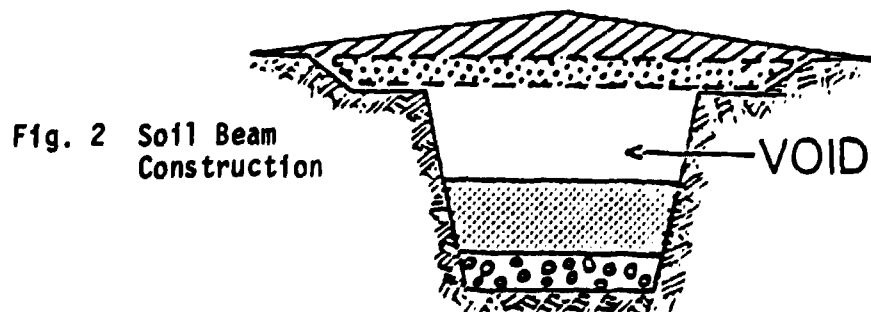


Fig. 2 Soil Beam Construction

The concept of the soil beam is the utilization of a geofabric such as Mirafi 500N to envelop a portion of the trench backfill material to create a semi-rigid support member or beam bridging a subsurface void caused by the decay and/or consolidation of the waste.

#### TRENCH DESIGNS

The trenches were designed to a standard size of approximately 10 feet deep and 10 feet by 20 feet in plan; therefore the total volume of each trench is about 2000 cubic feet. Figures 3, 4 and 5 are schematics of all of the trench designs.

#### SITE SELECTION AND TRENCH CONSTRUCTION

##### CGHF SITE

The CGHF site was selected because it is a dry area (approximately 11 inches of annual rain) with sufficient research space and totally under University of Arizona control. Table 1 contains a summary of the laboratory soil test data for this site.

The four trenches were constructed with a surface runoff collection/drainage system as shown in Fig. 6. The trenches were built essentially according to the design as portrayed in Figs. 3, 4 and 5.

##### ML SITE

The ML site was selected because of its more humid environmental characteristics and relatively close proximity to Tucson (about 35 miles). The climatic conditions at the ML site are more nearly equivalent to those conditions in the bulk of the contiguous 48 states. Freezing conditions and snow are prevalent during the winter with a total annual precipitation of approximately 25 inches. In this area there is weathering due to a freeze-thaw cycle plus the added effect of abundant natural soil moisture. Table 1 contains a summary of the ML site laboratory soil tests.

The four trenches on Mt. Lemmon were sited as described in Figs. 7 and 8. Due to the heavy forest and varied terrain it was impossible to site the trenches in the orderly way that was done at the CGHF site. There has been slight modification in the trench depths and monitoring systems due to underlying rock masses. It is felt that these changes will not effect the experiment. Figures 9, 10, 11 and 12 described the "as built" trench configurations at the ML site.



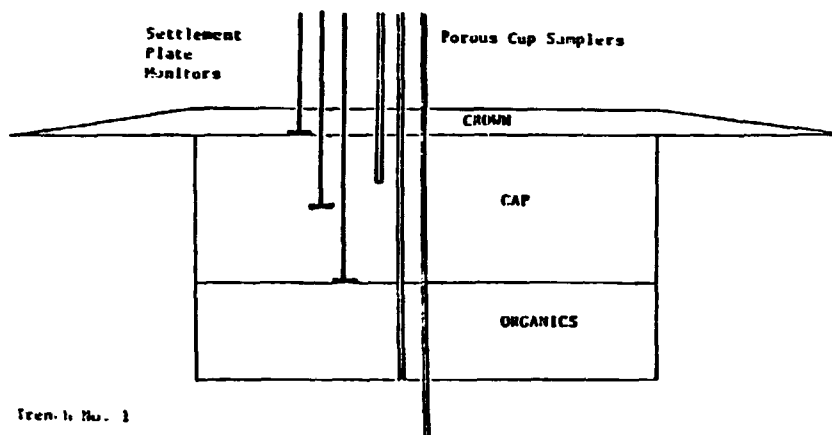


Fig. 3 Trench No. 1

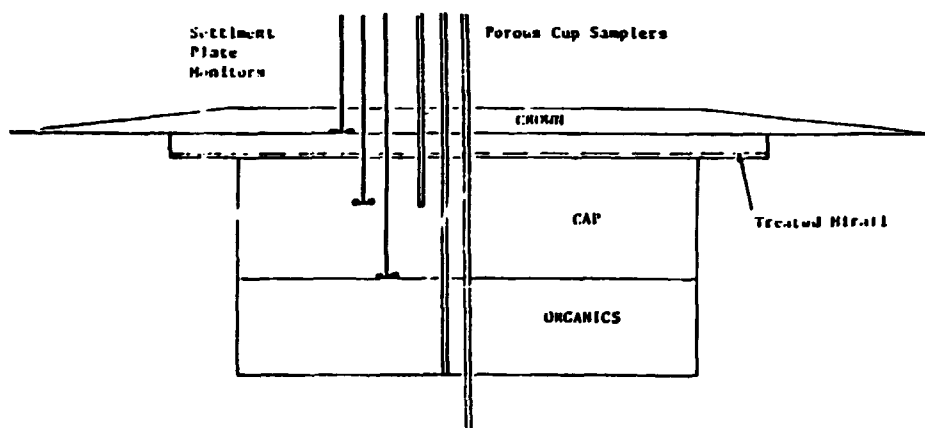


Fig. 4 Trench No. 2

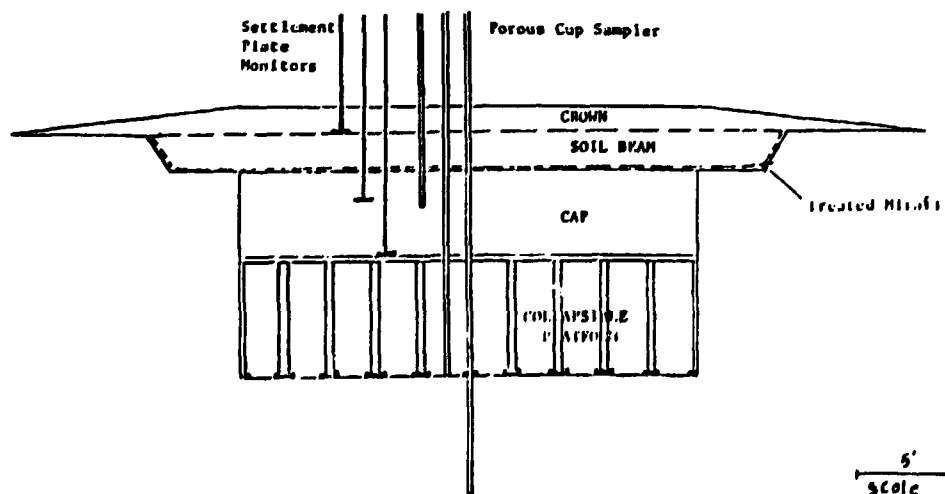
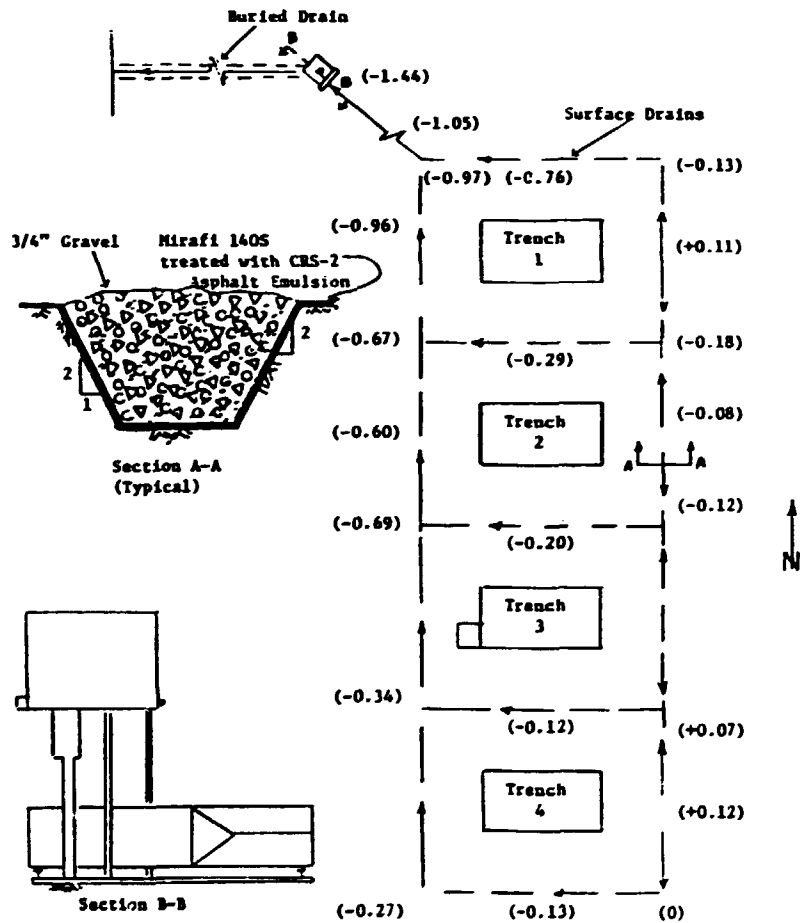


Fig. 5 Trench No. 3 and Trench No. 4 (Trench 4 has organics in lieu of collapsible platform)

5'  
scale

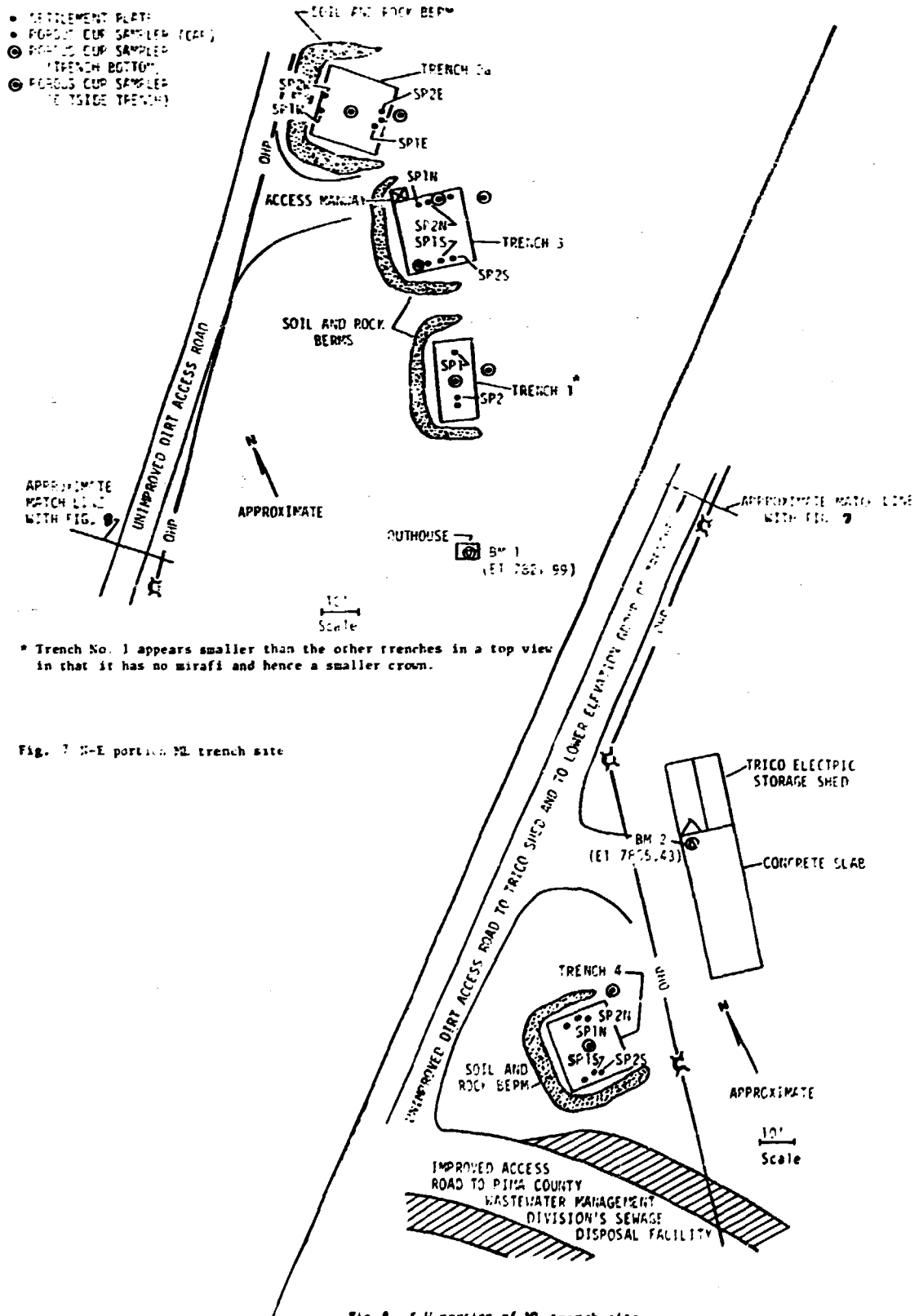
Table 1 Summary of Laboratory Test Data

| <u>Soil Property</u>                          | <u>CGHF<br/>Site</u> | <u>ML<br/>Site</u> |
|-----------------------------------------------|----------------------|--------------------|
| Unified Soil Classification Designation       | SM                   | SM                 |
| Specified Gravity of Solids                   | 2.55                 | 2.55               |
| Uniformity Coefficient                        | 125                  | 5                  |
| Shrinkage Limit (%)                           | 20                   | -                  |
| Liquid Limit (%)                              | 20                   | 27                 |
| Plastic Limit (%)                             | 23                   | 27                 |
| Plasticity Index                              | 3                    | Non Plastic        |
| Standard Proctor Maximum Dry Density (pcf)    | 102                  | 101                |
| Standard Proctor Optimum Moisture Content (%) | 14.0                 | 15.8               |
| Modified Proctor Maximum Dry Density (pcf)    | 113                  | 108                |
| Modified Proctor Optimum Moisture Content (%) | 13.5                 | 13.4               |
| Unconfined Compressive Strength (psf)         | 354                  | 1683               |
| Peak Effective Friction Angle (degrees)       | 33                   | 35                 |
| Peak Effective Cohesion Intercept (psf)       | 1200                 | 240                |



NOTE: Figures in parentheses represent difference in elevation (ft) at bottom of drainage ditch between point indicated and southwest corner of system (zero reference) arrows indicate direction of flow

Fig. 6 Trench Location Plan and Surface Runoff Collection/Drainage System, CCHP Site



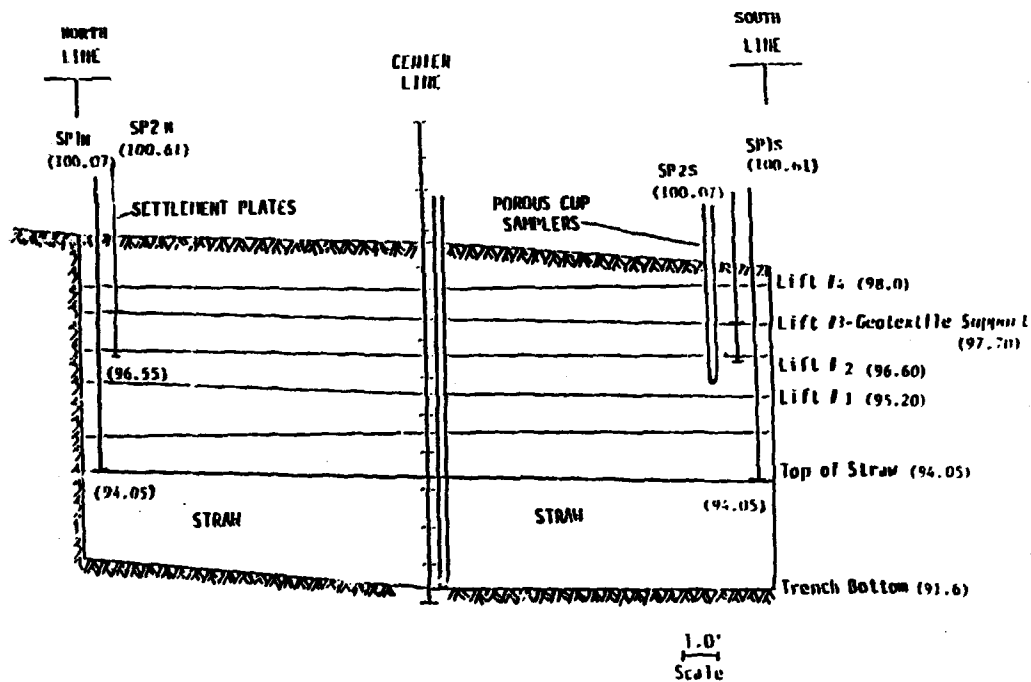


Fig. 9 Trench No. 1 Longitudinal Cross Section As-Built (Numbers in parentheses indicate average elevation of lifts at time of completion of individual lifts or elevations of top and bottom of settlement rod at time of installation)

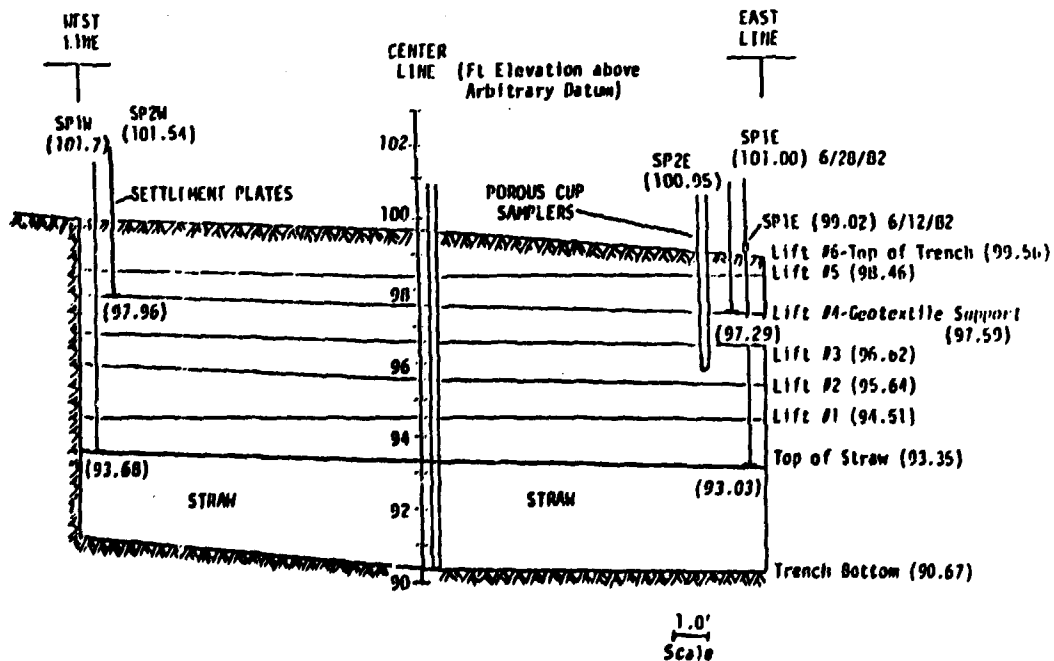


Fig. 10 Trench No. 2a Longitudinal Cross Section As-Built (Numbers in parentheses indicate average elevation of lifts at time of completion of individual lifts or elevations of top and bottom of settlement rod at time of installation)

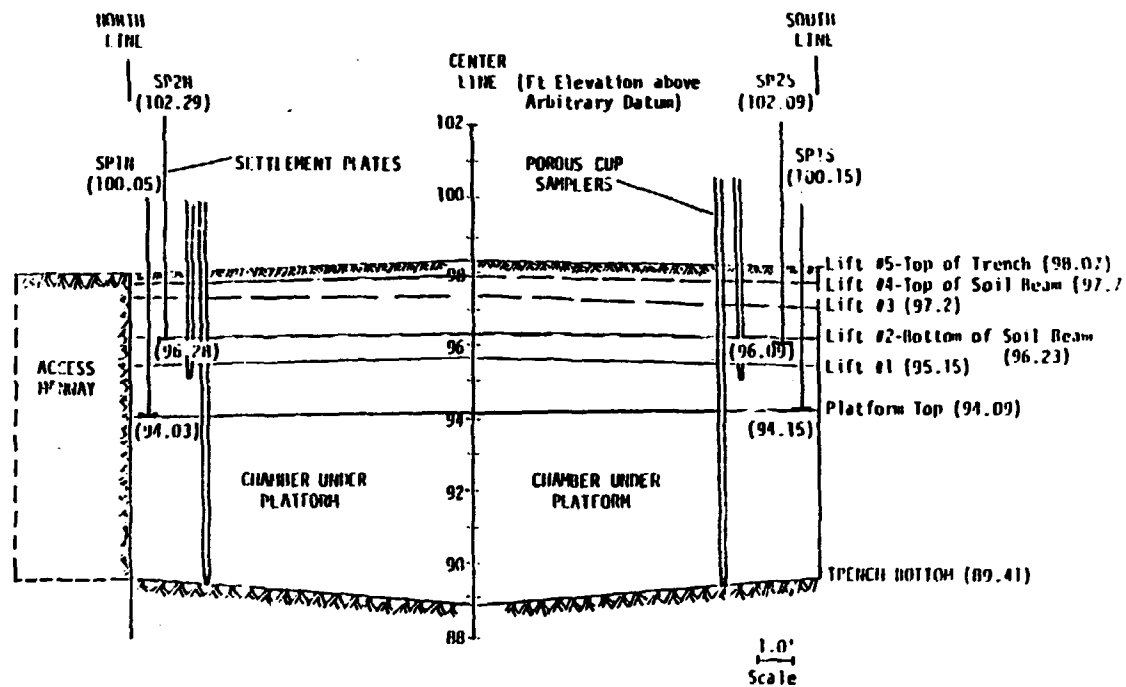


Fig. 11 Trench No. 3 Longitudinal Cross Section As-Built (Numbers in parentheses indicate average elevation of lifts at time of completion of individual lifts on elevations of top and bottom of settlement rod at time of installation)

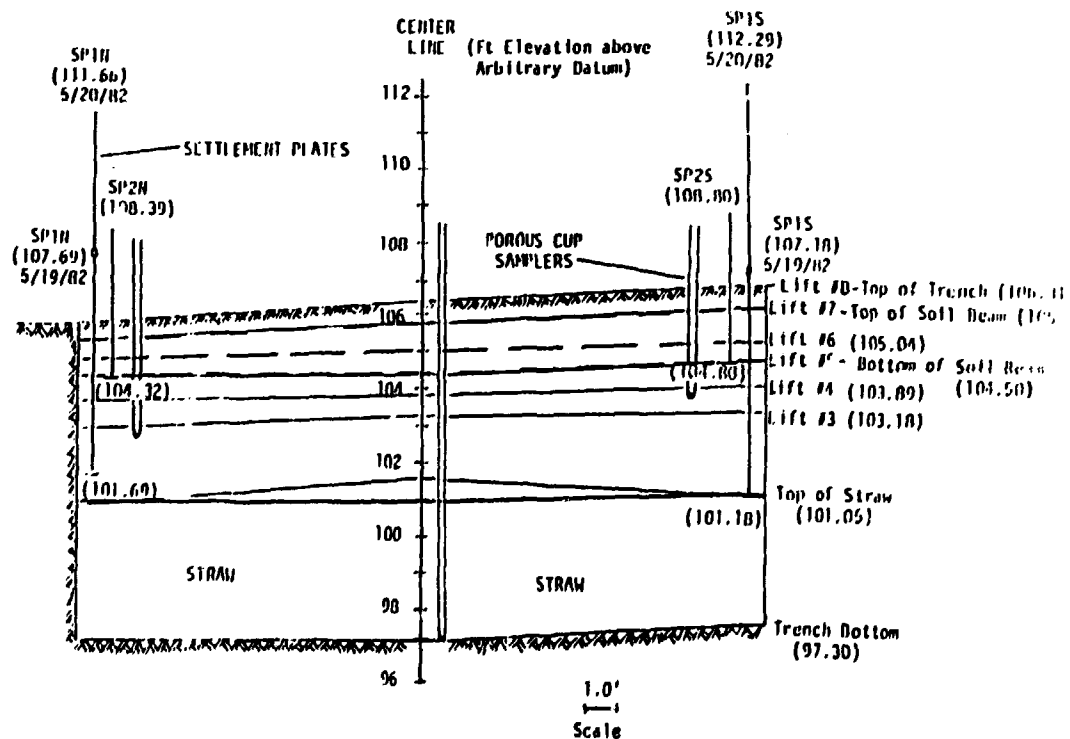


Fig. 12 Trench No. 4 Longitudinal Cross Section As-Built (Numbers in parentheses indicate average elevation of lifts at time of completion of individual lifts on elevations of top and bottom of settlement rod at time of installation)

## MONITORING PLAN

### HYDROLOGIC PLAN

This plan requires the use of sufficient tracers and monitoring points to be able to identify where the leakage has occurred and establish the pathway of the water. A better understanding of this can be gathered by referring to Fig. 13. The water samples taken either in or out of the trenches will identify from which trench and by what pathway the water has taken.

To insure sufficient environmental effects at the CGHF site a sprinkler system will be used to increase the precipitation from 10 inches to 100 inches of rain. This has not been done as yet. The tracers applied and their locations are described in Tables 2 and 3.

### STRUCTURAL PLAN

Settlement plates were installed in three locations in each of the trenches. One plate on top of the simulated waste; one plate midway in the cap; and one plate at the top of the cap below the crown.

## DATA ANALYSIS AND PRELIMINARY RESULTS

### STRUCTURAL ANALYSIS AND RESULTS

At this time the settlement plate data suggest that the Trench No. 4 cap design is superior to the other two (Trench No. 3 and Trench No. 4 have basically the same cap design). This is based mainly on the CGHF data. As can be seen from Fig. 14, the near surface settlement (SP3) of Trench No. 4 is approximately 0.3 feet; the rate of settlement is decreasing. On the other hand, a near-surface settlement of almost four times that amount (1.1 feet for SP3) has occurred on Trench No. 2 (Fig. 15); the rate of settlement has not decreased substantially. Figure 16 shows the settlement-time relationship for Trench No. 1. The near surface settlement for this trench (SP3) was lower than that of any of the other trenches up to 150 days after trench completion. Then, almost coincident with Tucson's "monsoon" season, Trench No. 1 surface settlements started to increase quite rapidly so that now they are almost equal in magnitude to those of Trench No. 2. It is felt that this is mainly due to the effect of moisture on the strength of the unreinforced soil that comprises the Trench No. 1 cap.

Figures 17 and 18 show the settlement time relationships for Trench 1 and 2a respectively on Mt. Lemmon. As can be seen from the figures, the magnitudes of settlement are not as great as the CGHF magnitudes and the rates for the most part are levelling off.

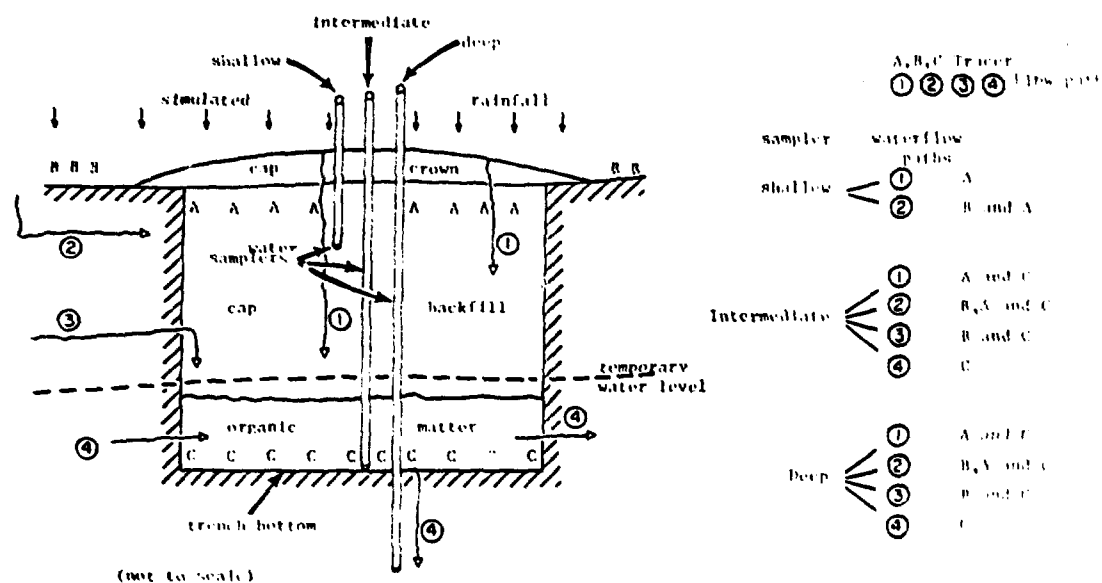


FIG. 11 Schematic Representation of Water Entry Pathways Into Trench and Tracer Pipe. The water flow paths shown represent the conditions where: 1) when the cap backfill is more permeable than the native soil (this permeability being caused by vertical cracking; and 2) the undisturbed soil outside the trench is anisotropic with its vertical permeability being less than its horizontal permeability results in horizontal flow.



Table 2 Porous Cup Depths and Tracer Information - Trenches at CGHF Site

| <u>Trench Number</u> | <u>Porous Cup Sampler Depths</u> | <u>Tracer Location</u> | <u>Amounts and Type of Tracer Used and Depth of Spraying</u>                                                   |
|----------------------|----------------------------------|------------------------|----------------------------------------------------------------------------------------------------------------|
| 1                    | 3 ft. b.l.s.*                    | Cap Backfill:          | 756 g of potassium Iodide (KI) at land surface                                                                 |
|                      |                                  |                        | 756 g of potassium Iodide (KI) at 1 ft. b.l.s.                                                                 |
|                      | 10 ft. b.l.s.                    |                        | 756 g of potassium Iodide (KI) at 2 ft. b.l.s.                                                                 |
|                      | 15.5 ft. b.l.s.                  | Trench Bottom:         | 400 g of heptafluorobutyric acid (CF <sub>3</sub> CF <sub>2</sub> COOH) at 10 ft. b.l.s.                       |
|                      | 3 ft. b.l.s.                     | Cap Backfill:          | 500 g of fluorescein-uranine (C <sub>20</sub> H <sub>10</sub> Na <sub>2</sub> O <sub>5</sub> ) at land surface |
| 2                    | 10 ft. b.l.s.                    |                        | 1000 g of sodium bromide (NaBr) at 1 ft. b.l.s.                                                                |
|                      | 13 ft. b.l.s.                    |                        | 300 g of sodium bromide (NaBr) at 2 ft. b.l.s.                                                                 |
|                      |                                  | Trench Bottom:         | 400 g of pentafluoropropionic acid (C <sub>2</sub> F <sub>5</sub> COOH) at 10 ft. b.l.s.                       |
|                      | 3 ft. b.l.s.                     | Cap Backfill:          | 130 g of chlorodifluoroacetic acid (CF <sub>2</sub> ClCOOH) at land surface                                    |
|                      |                                  |                        | 130 g of chlorodifluoroacetic acid (CF <sub>2</sub> ClCOOH) at 1.5 ft. b.l.s.                                  |
| 3                    | 10 ft. b.l.s.                    |                        | 130 g of chlorodifluoroacetic acid (CF <sub>2</sub> ClCOOH) at 2 ft. b.l.s.                                    |
|                      | 13.8 ft. b.l.s.                  | Trench Bottom:         | 400 g of trifluoroacetic acid (CF <sub>3</sub> COOH) at 10 ft. b.l.s.                                          |
|                      | 3 ft. b.l.s.                     | Cap Backfill:          | 250 g of EDTA at land surface                                                                                  |
|                      |                                  |                        | 360 g of M-trifluoromethylbenzoic acid (CF <sub>3</sub> C <sub>6</sub> H <sub>4</sub> COOH) at 2 ft. b.l.s.    |
|                      |                                  |                        | 250 g of EDTA at 2 ft. b.l.s.                                                                                  |
| 4                    | 10 ft. b.l.s.                    | Trench Bottom:         | 300 g of pentafluorobenzoic acid (C <sub>6</sub> F <sub>5</sub> COOH) at 10 ft. b.l.s.                         |
|                      | 14.5 ft. b.l.s.                  |                        |                                                                                                                |

\* b.l.s. = below land surface

Table 3 Porous Cup Depths and Tracer Information - Trenches at ML Site

| <u>Trench Number</u> | <u>Porous Cup Sampler Depths</u> | <u>Tracer Location</u> | <u>Amounts and Type of Tracer Used and Depth of Spraying</u>                                                                                                                                                       |
|----------------------|----------------------------------|------------------------|--------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------|
| 1                    | 3 ft. b.l.s.*                    | Cap Backfill:          | 756 g of potassium iodide (KI) at 2 ft. b.l.s.                                                                                                                                                                     |
|                      | 8 ft. b.l.s.                     | Trench Bottom:         | 400 g of heptafluorobutyric acid ( $\text{CF}_3\text{CF}_2\text{COOH}$ ) at 8 ft. b.l.s.                                                                                                                           |
| 2a                   | 3 ft. b.l.s.                     | Cap Backfill:          | 700 g of sodium bromide (NaBr) at 1.5 b.l.s.<br>500 g of fluorescein-uranine ( $\text{C}_{20}\text{H}_{10}\text{Na}_2\text{O}_5$ ) at 1.5 ft. b.l.s.                                                               |
|                      | 9 ft. b.l.s.                     | Trench Bottom:         | 163 g of pentafluoropropionic acid ( $\text{C}_2\text{F}_5\text{CO}_2\text{H}$ ) at 9.6 ft. b.l.s.                                                                                                                 |
| 3                    | 3 ft. b.l.s.                     | Cap Backfill:          | 160 g of chlorodifluoroacetic acid ( $\text{F}_2\text{C Cl CO}_2\text{H}$ ) at 0.4 ft. b.l.s.                                                                                                                      |
|                      | 9 ft. b.l.s.                     | Trench Bottom:         | 160 g of chlorodifluoroacetic acid ( $\text{F}_2\text{C Cl CO}_2\text{H}$ ) at 1.8 ft. b.l.s.<br>Will apply 400 g of trifluoroacetic acid ( $\text{CF}_3\text{COOH}$ ) at 9 ft. b.l.s. prior to collapsing trench. |
| 4                    | 3 ft. b.l.s.                     | Cap Backfill:          | 200 g of m-trifluoromethylbenzoic acid ( $\text{CF}_3\text{C}_6\text{H}_4\text{COOH}$ ) at 1 ft. b.l.s.                                                                                                            |
|                      | 9.6 ft. b.l.s.                   | Trench Bottom:         | 100 g of m-trifluoromethylbenzoic acid ( $\text{CF}_3\text{C}_6\text{H}_4\text{COOH}$ ) at 2 ft. b.l.s.<br>300 g pentafluorobenzoic acid ( $\text{C}_6\text{F}_5\text{CO}_2\text{H}$ ) at 9.6 ft. b.l.s.           |

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\* b.l.s. = below land surface

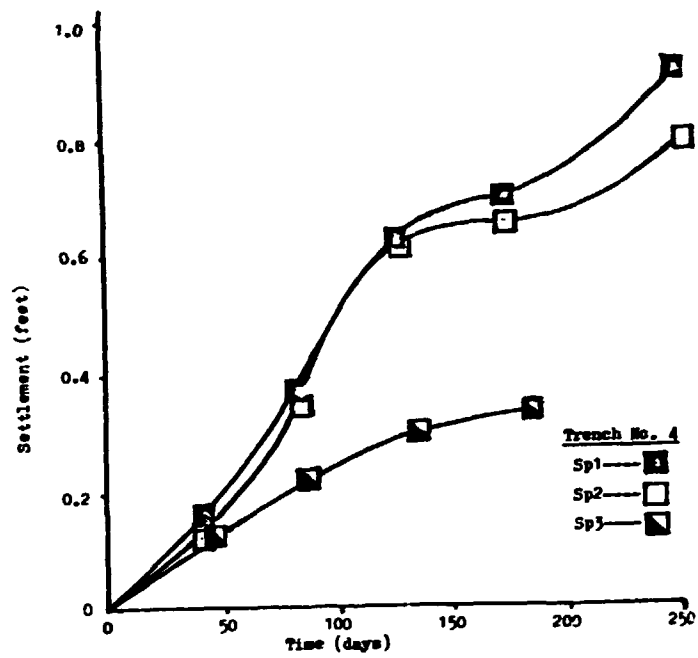


Fig. 14 Settlement Vs. Time (CGHP)

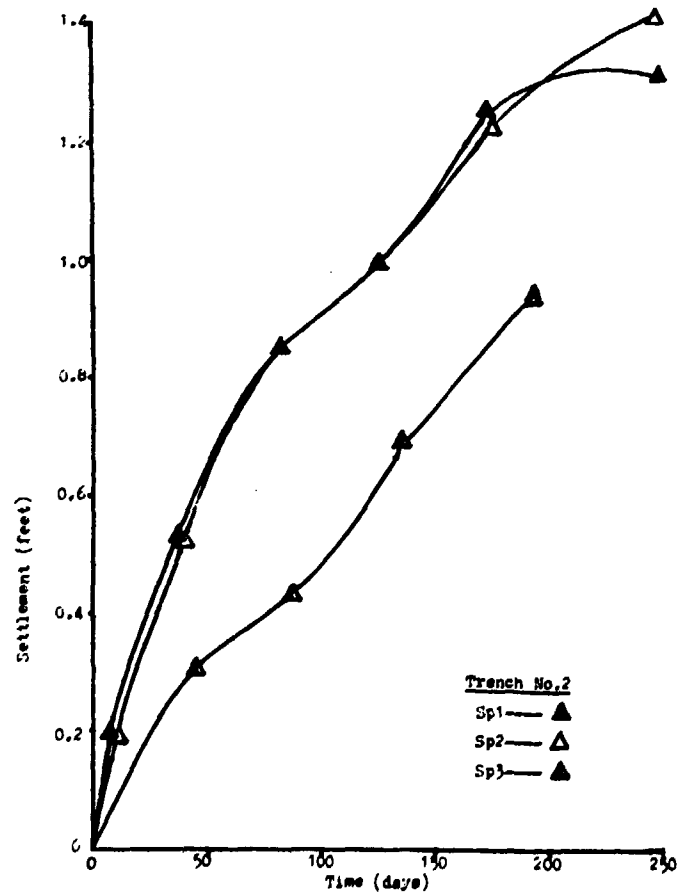


Fig. 15 Settlement Vs. Time (CCPI)

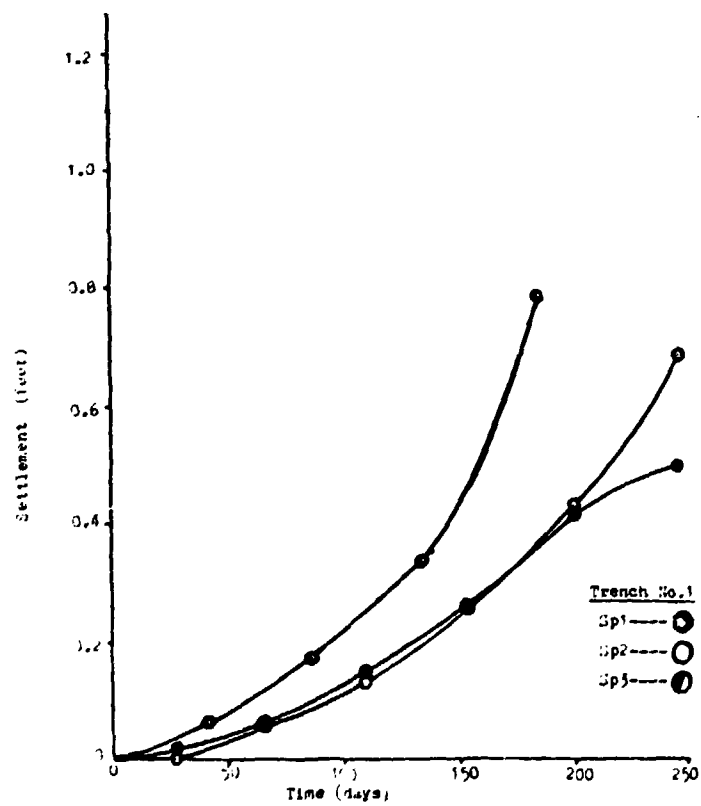


Fig. 1b Settlement vs. Time (Trench No. 1)

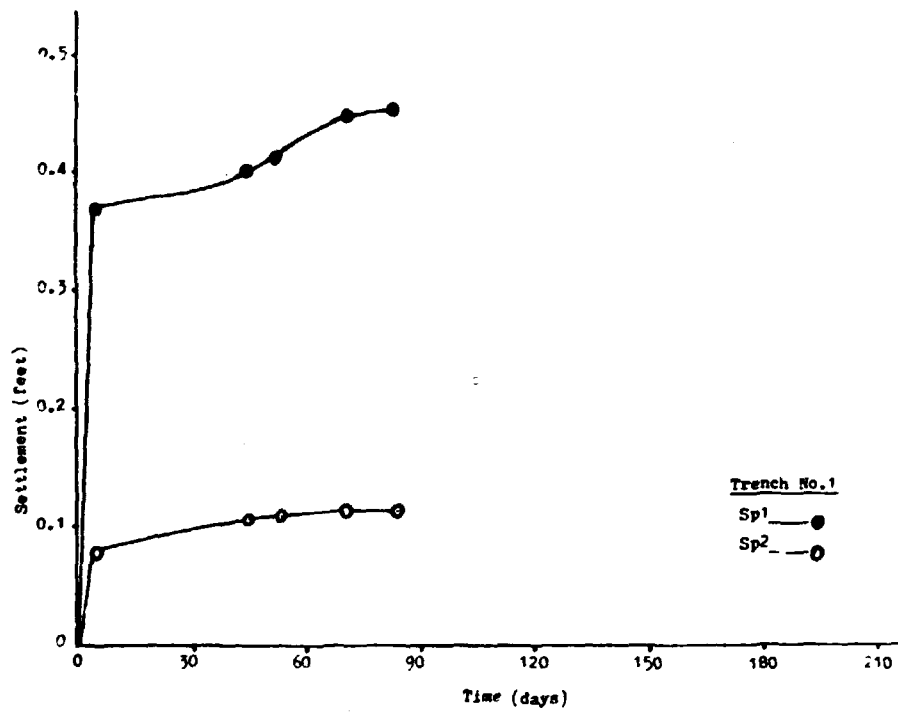


Fig. 17 Settlement Vs. Time (ML).

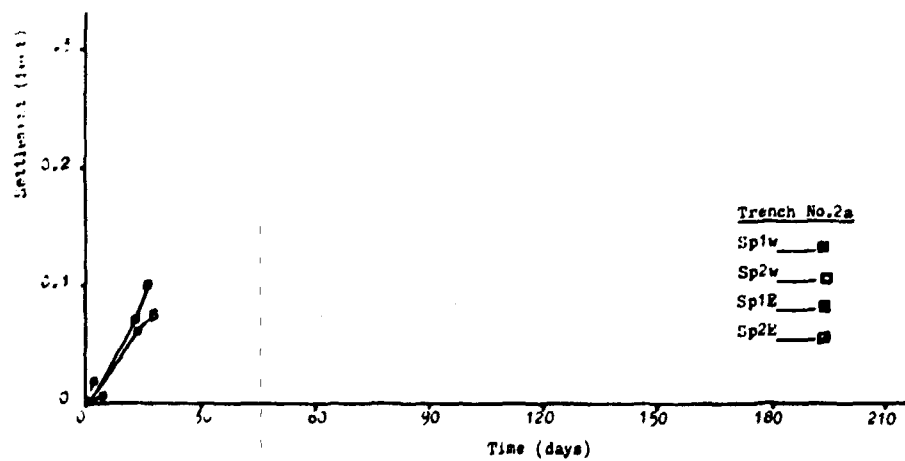


Fig. 18 Settlement Vs. Time (ML)

## HYDROLOGIC ANALYSIS AND RESULTS

Two rounds of sampling of the porous cups have been made at both the ML and CGHF sites. The samples obtained are currently being analyzed for tracer content. The successful collection of samples varied from the sites with the ML site having 11 out of 15 possible samplers with water and the CGHF site having 8 out of 24 possible samplers with water. The high percentage of dry samplers at the CGHF site is due primarily to the low rainfall at the site.

Preliminary results of the water samples have been run for trenches number ML-1, ML-2a and ML-3. Trench ML-1 shows possible movement of tracer C through pathway 4 (Fig. 13). Trench ML-2a shows possible movement of tracer A through pathway 1 (Fig. 13). Trench ML-3 also shows possible movement of tracer A through pathway 1 (Fig. 13). It is stressed that these results are from preliminary analyses and may change as updated or different analytical techniques are implemented.

## FUTURE PLANS

### HYDROLOGIC AND STRUCTURAL MONITORING

During FY 1983, the principal activity will be monitoring for the movement of tracers at both of the experimental burial grounds. Ten compounds have been applied as tracers in the experimental trenches, and these compounds will have to be looked for on a regular basis in most of the sampling points in and around the trenches.

There are approximately 40 sampling points total in the two sites. We expect to make measurements on a monthly basis at each sampling point. To economize the time and effort involved in monitoring, only the tracers that have a reasonable possibility of being detected at a given sampling point will be investigated; otherwise, the analytical load will outrun our manpower and equipment capabilities.

Structural monitoring will consist of establishing settlement plate elevations bimonthly for all trenches at both the CGHF site and the ML site. Settlements for each trench will be plotted vs. time so that differential movements between various elevations within any given trench can be monitored and compared to similar movements within other trenches. Such plots will also enable settlement rates to be determined.

### STRUCTURAL RESPONSE COMPUTER CODES

An existing finite element computer code will be monitored to model the as-built trench configuration in order to predict the load-displacement characteristics of the proposed trench-cap-crown system. The analyses will be similar to those performed for the initial four trenches at CGHF.

### REHABILITATION OF TRENCH NO. 3 AT CGHF SITE

The collapse of Trench No. 3 at the CGHF site following explosive destruction of the support platform negates its usefulness for further study after the detailed analysis of the soil beam failure mechanisms. It is proposed to excavate the debris out of Trench No. 3 and rehabilitate it to accept the modified design shown in Fig. 19. The main difference between this modified design and the original design is the sloping trench walls extending down from the surface to approximately the platform elevation. It is felt that this configuration will cause the trench cap to act more like a plug and enhance the development of a soil arch after the platform is mechanically collapsed. The rest of the design is basically the same as that constructed previously with the exception that settlement plates and tracers feedthrough will be placed at the narrow ends of the trenches so as not to perforate or rip the geotextiles in the soil beam.

### TRENCH NO. 5 AT CGHF AND ML SITES

It is planned that a Trench No. 5 be constructed at each site with two concepts, the "soil beam" and the "wick" drain. Figure 20 contains a sketch of the proposed design. The "wick" is basically a blanket drain consisting of a filtered layer in contact with the compacted soil crown above the wick, and a coarser collector layer beneath it in contact with the top surface of the soil beam. The soil beam will be constructed to have a greater thickness in the center than at the edges. This configuration serves two purposes: first, it provides a sloping base for the wick and crown so that drainage is away from the trench and, second, it has its largest section in the center where maximum moments would occur should a void be created directly beneath the soil beam. The "skin" of the soil beam will consist of Mirafi 500N geotextiles or some suitable substitute. The cap itself will consist of local soil used as backfill and compacted to within 95% of maximum "standard" dry density (ASTM D698) at a moisture content within 4% of optimum. It will be constructed in lifts not to exceed one foot in thickness. The organics will consist of baled straw stacked in the bottom of the trench to a non-compacted initial height of not less than 5 feet. The crown above the wick shall consist of compacted native soils. The following devices will be emplaced as indicated in Fig. 20.

### Settlement Plates

Because experience has shown that rips and/or perforations severely impair the load carrying capacity of geotextiles, settlement plates will be installed along the short sides of each trench beyond the edges of the soil beam so that the rods may extend to the surface without having to pass through the soil beam itself. Sets of settlement plates will be placed at the following elevations: top of the organics, midway in the cap, immediately below the soil beam, immediately on top of the soil beam (over its center). The settlement plate design and the method of installation will be the same as was used for the first four trenches.

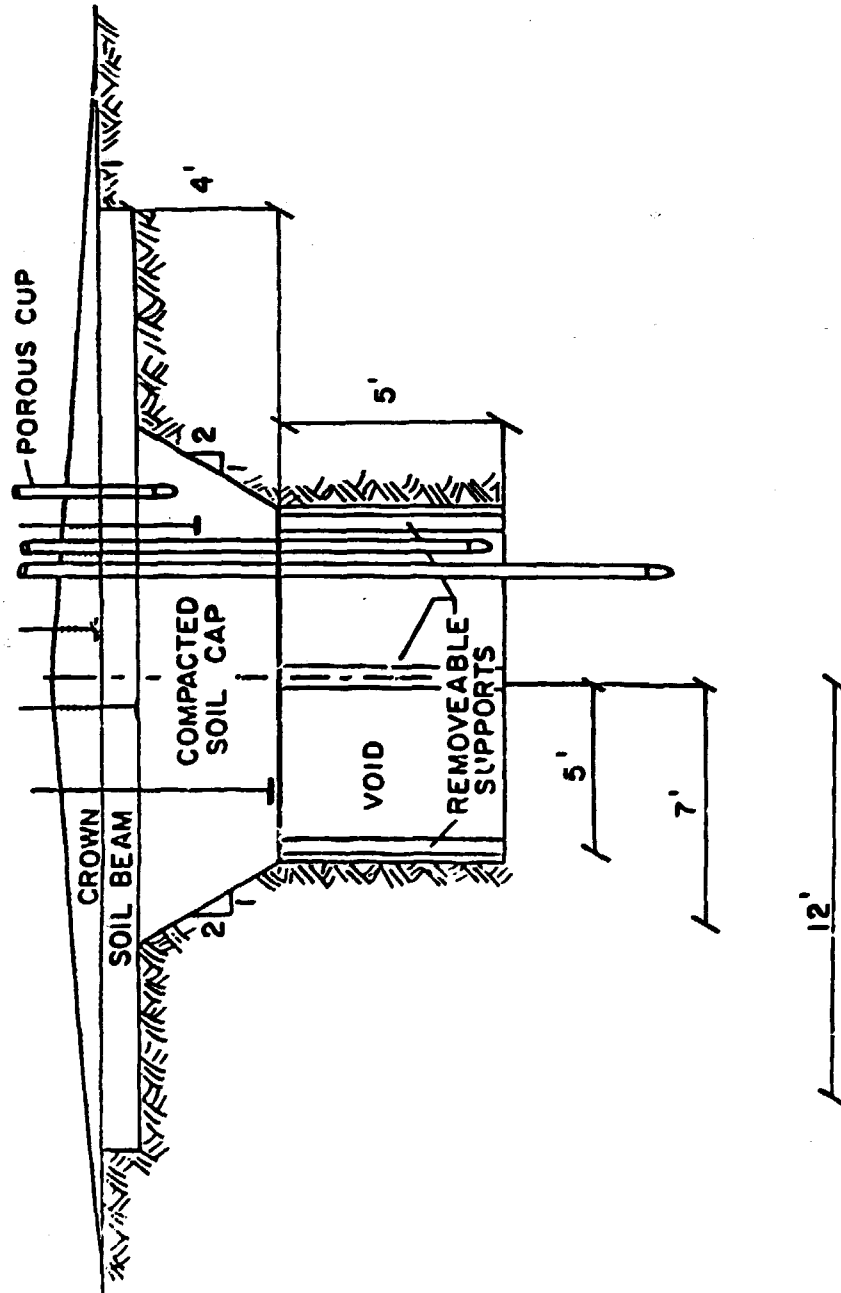


Fig. 19 Modified Trench No. 3 at CGHF Site



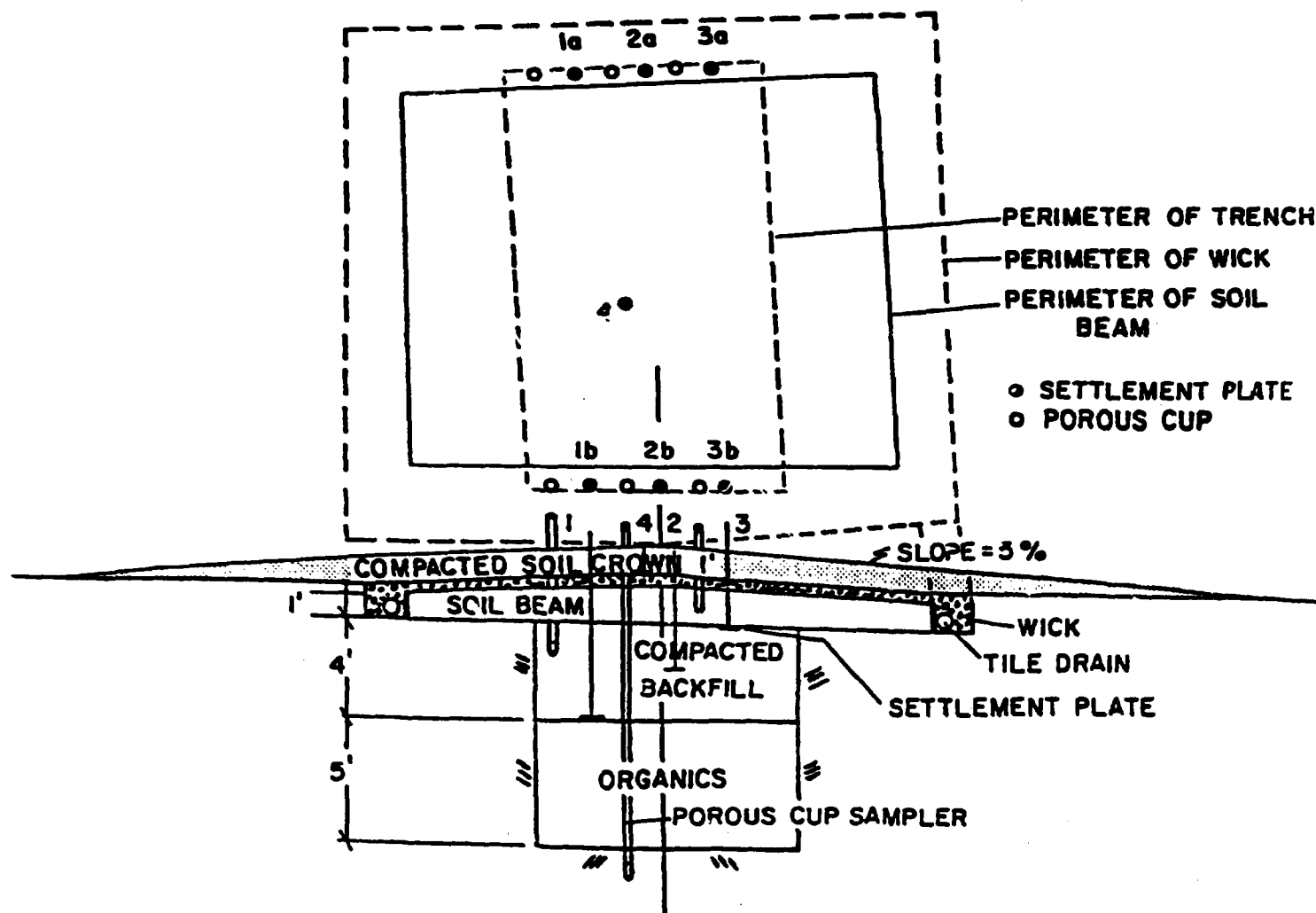


Fig. 20 Trench No. 5 Soil Beam/Wick Design

### Porous Cup Tracer Monitors

In a like manner and for the reasons explained above, the porous cup tracer monitors will be emplaced along the short side of the trench to monitor at or below the trench bottom, in the cup backfill and at the wick.

It is expected that at the completion of this experiment conclusions will be reached regarding optimized trench capping and trench monitoring techniques in both dry and humid climates. In addition, it is expected that methods will be developed to improve the prediction of relative long term stability of these designs.

Recommendations will be made as to regulations required to insure radionuclide isolation and protection of the public in so far as has been concluded in these experiments.

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## TRENCH CAP STUDIES

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## INTRODUCTION

Currently there are 11 major shallow land burial grounds for low-level radioactive wastes in the United States. Five of these are operated by the U.S. Department of Energy (DOE) and six are commercial sites. Of these, two DOE and four commercial sites are located in the humid eastern part of the country. Tritium migration has been observed at five of these sites (1). The probable cause of migration is infiltration of precipitation through trench covers and damage through loss of control of the surface water (2). The reduction of infiltration at low-level radioactive waste disposal sites appears essential to proper management in humid regions.

Radionuclide migration from low-level waste disposal sites can be reduced by limiting water contact with the wastes. Properly designed, constructed, and maintained trench caps can reduce water contact with the wastes by limiting the infiltration of precipitation. In the past, trench caps have been designed and constructed with mixed results. Material excavated from the trenches is used for the cover. The material is mounded to a minimum depth of one to three meters, then vegetated for erosion control. At sites in the humid regions, covers are compacted by heavy earth-moving equipment or by temporarily placing excavated material from adjacent trenches on the mounds. The covers at the Barnwell and Maxey Flats sites and those at the more recent trenches at the Sheffield site incorporate clay obtained from local sources.

This paper is a review of ongoing research at the Illinois State Geological Survey under contract from the Nuclear Regulatory Commission. The objectives of this research are (1) to provide basic information on the characteristics of materials that can be used in the construction of waste-trench caps, (2) to study the interaction of these materials which will limit the precipitation infiltrating through the wastes, (3) to test under field conditions the most promising trench cover designs, and (4) to recommend to the U.S. Nuclear Regulatory Commission, Division of Waste Management, criteria and procedures for review of trench cover designs.

## TASK 1: REVIEW OF PRESENT PRACTICE

The Task 1 report presents a survey of present practices in the design of trench covers and reviews the scientific literature dealing with this subject. The report includes an annotated bibliography of 208 references, prefaced by a summary review of the subject taken directly from the references. The annotations that accompany the references consist of a brief synopsis as well as an indication of relevance to trench cover design. The quality of the references is not discussed.

This literature search located very little information dealing specifically with the design of covers for low-level radioactive waste sites. Most of the information relative to radioactive waste sites is site specific and contains little quantitative information. The generic cover improvements that have been suggested generally come from the disciplines of sanitary waste disposal and soil physics. The following is a short summary of findings from this report.

Two general types of covers have been proposed: man-made "impermeable" covers (concrete, asphalt, and impermeable membranes), and covers using natural materials (clay barriers or layered designs). Increasing the thickness of existing covers has also been suggested, but this alone may not solve the problem in humid regions. In a study at the West Valley site, Kelleher (3) showed that, for erosion control, individual trench covers are superior to a single cover over the entire site. SCS Engineers (4) suggests that the uppermost one meter of all trenches should be filled with excavated material, followed by a compacted, mounded and sloped cover to reduce infiltration and the effects of waste decomposition and settling. The recommended optimal slope is 5 percent.

Arora (5) concluded that the use of rigid barriers, such as concrete, concrete asphalt, soil cement, and metallic sheets, is largely precluded by persistent subsidence, as has been observed in many trenches. (Flexible, synthetic membranes are excluded because of short expected lifetimes.)

Clays currently used in trench covers are generally native clays. For a clay cover to be effective, the U.S. EPA (6) has recommended a minimum of 6 inches of clay, with a hydraulic conductivity equal to, or less than,  $1 \times 10^{-7}$  cm/sec, covered by 19 inches of topsoil. SCS Engineers (4) recommended a minimum of 2 feet of soil over the clay to prevent cracking and noted that clay can be combined with a plastic liner. Inexpensive soil sealants may also be used as a backup system for clay. Data indicate that the clay covers at Maxey Flats have not prevented tritium migration, whereas at Barnwell, where a minimum of three meters of soil is placed over a clay barrier, no tritium migration has been noted.

Several researchers have suggested designs based on the wick effect. Farb (7) suggested a sequence of topsoil, sand, and clay as a means for upgrading hazardous waste disposal sites. Dragonette et al. (8) suggested a wick cover over a series of trenches and suggested adding a drainage pipe in the coarse layer. The optimal design suggested by Dragonette et al. (8) was a wick cover over a series of trenches, with or without perimeter drains around the site. Cherry et al. (9), Frind et al. (10), and Rancon (11), have suggested taking the wick concept one step further and using it around an entire trench. This, of course, cannot be used on trenches as a remedial measure. It is being tested (9) at Chalk River in Canada. To the best of our knowledge, no other wick covers are being tested.

## TASK 2: LABORATORY EVALUATION AND COMPUTER MODELING OF TRENCH COVER DESIGN

The Task 2 report includes initial results of laboratory tests to determine the properties of selected natural earth materials to be used in subsequent laboratory and field experiments and in computer modeling of proposed trench cover designs; describes the gamma-ray attenuation system for determination of water content and density in laboratory soil columns and of the preliminary results from the initial experiment involving layered soils; discusses the application and availability of mathematical models and computer programs describing the flow of moisture through variably saturated materials; and presents results of field-scale and laboratory column scale computer modeling of infiltration through alternative cover designs.

### LABORATORY TESTS

The hydraulic behavior of unsaturated, compacted, fine-grained materials affects the performance of several types of engineering facilities, including covers for waste disposal sites. Unsaturated moisture movement may play a significant part in infiltration through trench covers. In one of the first studies of the hydraulic conductivity of compacted, fine-grained soils, Lambe (12) pointed out that the saturated hydraulic conductivity was influenced by the soil composition, the chemistry of the pore fluid, the void ratio (or porosity), the soil structure and the degree of saturation.

The influence of soil structure on the hydraulic conductivity of fine-grained soils is one of the more interesting factors. Several investigations (12, 13, 14) have observed much lower values of saturated hydraulic conductivity for samples compacted wet of optimum moisture content than for samples compacted dry of optimum. Our study extends this work to an investigation of the unsaturated behavior of soils.

Desaturation tests were conducted to determine moisture content-retention relationship (soil-moisture characteristic curve) for each

material using tempe cells or a pressure membrane extractor. Samples compacted at moisture contents above optimum retain more moisture at higher pressures than samples compacted at moisture contents below. The smaller pores of the samples compacted wet of optimum are not desaturated as easily as the larger pores of the samples compacted dry of optimum.

Hydraulic conductivity calculations were performed for samples compacted wet and dry of optimum for the loess and the till. The figures indicate an increasing difference in the hydraulic conductivities of soils compacted wet and dry of optimum as the soils are desaturated.

## COMPUTER MODELING

The evaluation of alternative cover designs that will limit the infiltration of moisture into waste disposal sites involves the use of predictive techniques. Using specified boundary conditions and the predicted relationship between  $K$  and  $\theta$  for each material, a mathematical model can simulate the flow of moisture through individual materials and layered combinations of materials to investigate a wide range of hypothetical problems.

### One-dimensional Model

Initially, infiltration into layered covers was studied by using a one-dimensional finite difference computer program entitled MOMOLS. Initial and surface boundary conditions were found to be important factors in the system. Higher values of initial capillary pressure (or tension) increase moisture entry. Although the soil has a lower degree of saturation (and therefore a lower hydraulic conductivity) when the capillary pressure is higher, the potential infiltration rate is much higher when the degree of saturation is lower. This last effect apparently overrides the influence of the difference in conductivities.

Moisture entry is also influenced by the water content of the compacted soil when fine-grained soils are used in the surface layer. Soils compacted wet of optimum exhibit much lower moisture entry and a much slower rate of moisture front advance than soils compacted dry of optimum moisture.

Moisture redistribution can be investigated with the model by changing the surface boundary condition from a constant head to a constant surface flux equal to zero. After a simulated rainfall, the moisture front advances deeper for the soil compacted dry of optimum (about 11 cm as opposed to only 4 cm for the soil compacted wet of optimum).

If the texture of the second layer is much coarser than the top layer, moisture will build up in the top layer before entering the bottom layer. The water content in a layer of compacted Peoria Loess overlying a coarse gravel with a saturated hydraulic conductivity of 1 cm/sec builds up significantly prior to breakthrough at 2600 hours of continued surface ponding. A much more limited moisture retention occurs

in compacted Peoria Loess overlying medium sand with a hydraulic conductivity much less than the gravel. In this case, moisture breakthrough occurs after only 940 hours.

The efficiency of the "wick effect" is influenced by the initial conditions and the compaction water content of the top layer. The higher the initial capillary pressure, the more effective is the top layer in storing moisture. Soils compacted dry of optimum exhibit much higher effective storage values, primarily because of soil moisture retention characteristics. Soils compacted wet of optimum possess much less potential storage because they are more highly saturated than are soils compacted dry of optimum.

To summarize, this modeling indicated that in layered systems a coarse, unsaturated material was able to impede moisture movement. In all simulations, progress of the wetting front temporarily or permanently ceased once the interface between the gravel and the upper fine-grained material was reached. The lack of moisture movement across the interface caused pressure potential and moisture contents to build up in the overlying layer. Pressure and moisture content increased until either a static pressure profile was established (and all moisture movement ceased) or interfacial pressures reached a critical maximum value,  $\psi_c$ . If  $\psi_c$  was reached at the interface, the moisture front moved rapidly into the coarse layer. Rapid infiltration into the gravel reduced moisture contents directly above the interface and pressures at the interface quickly dropped from  $\psi_c$  to a value which was more or less maintained thereafter as moisture movement across the boundary continued.

#### Two-dimensional model

The finite element model, UNSAT2, developed by Neuman (15) and Neuman et al. (16) was used to simulate two-dimensional moisture movement through laboratory columns and field-scale cover designs. The model uses a finite element method to simulate two-dimensional nonsteady flow of water in incompressible, saturated-unsaturated soils.

Laboratory Scale. A soil column 29 cm long and 3 cm wide was chosen to simulate laboratory gamma column experiments. In most simulations, the column consisted of 11 cm of Peoria Loess, overlain by 9 cm of a medium sand and 9 cm more of Peoria Loess. The model consisted of 116 rectangular elements with dimensions of 1 cm by 2 cm. For all simulations, the sides of the column were considered no-flow boundaries and all but the top and bottom nodes had initial pressure heads of -100 cm. The relationship between moisture content,  $\theta$ , and pressure heads,  $\psi$ , was obtained from laboratory tests.

The effects of several different boundary conditions on infiltration were studied using this model. Boundary conditions tested included: (1) constant ponding at the surface and constant pressure head of -100 cm at the bottom, (2) constant ponding at the surface and watertable at the bottom, (3) constant ponding at the surface and a seepage face at the bottom, (4) constant surface flux of  $2 \times 10^{-4}$  cm/sec and a seepage



face at the bottom, and (5) a single layer of Peoria Loess.

In general, these simulations showed that very little moisture build-up occurs in the upper layer of loess before breakthrough into the sand below. In light of the one-dimensional modeling, this finding indicates that the textural contrast between layers must be significantly greater than the 3 orders of magnitude used in this simulation. In order for the wick effect to work, the material must also be dry initially. Our initial pressure head of -100 cm for each node in the simulation may have not been enough suction for the desired effect. These problems will be studied in the future.

Field Scale. Infiltration and moisture movement through a hypothetical field-scale layered cover design was simulated using UNSAT2. The particular design chosen initially for modeling, consists of three layers each 60 cm thick. A single layer cover 180 cm thick was also simulated. The cover is assumed to be 6 m wide with a 5 percent slope of each layer from the center to the sides. It is anticipated that this will be the approximate configuration of one or more pilot-scale covers to be tested in the field. Since the hypothetical cover design is symmetrical, moisture movement through only one-half of the cover was modeled.

The materials used in the field-scale cover simulations consisted of compacted Peoria Loess and gravel. The saturated hydraulic conductivities of the loess and gravel were  $2.0 \times 10^{-6}$  cm/sec and 1.0 cm/sec, respectively. The effects of hysteresis are not included in these simulations.

Initial conditions simulated at this scale were similar to those simulated in the laboratory scale study. However, in the field-scale cover simulations, pressure and moisture build-up before breakthrough is substantial. When the bottom is held at a constant pressure of -100 cm, breakthrough into the gravel occurs after 382 hours of constant surface ponding. When the bottom condition is a seepage face, breakthrough occurs after 171 hours of constant surface ponding. The success of this model in demonstrating the wick effect depends on the large textural contrast afforded by the gravel underlying the loess.

The effect of a slope on the layers was also studied in these simulations. In general, a lateral component of flow in the upper loess prior to breakthrough was apparent. However, this component was virtually eliminated after the moisture front penetrated the gravel layer below.

## VERIFICATION

The computer simulation was verified using a laboratory soil column of approximately the same dimensions and composition as that used in the simulation. The column was constructed of 2 plexiglas segments, each 10 cm wide and 15 cm high. Moisture movement through the sand layer was

measured using a gamma-ray attenuation device. Standardization of the device was obtained by regularly passing the gamma beam through an aluminum block supported at the top of the column. Initial runs were made before the gamma device was completely calibrated, so that reliable measurements could not be made of moisture movement through the loess which contains a substantial amount of expansive clays.

The column was maintained in a condition of constant surface ponding for over 255 hours. A vertical moisture profile was established after 67 hours as predicted in the simulation. However, the laboratory column maintained this profile much longer than predicted before the base of the sand unit began to approach saturation. This difference may be due to the slightly different characteristics of the materials actually used in the lab and those used in the simulation.

### TASK 3: CRITERIA FOR EVALUATING TRENCH COVERS

The Task 3 report was written in response to the need for criteria with which to judge trench covers on existing and future low-level radioactive waste burial sites. The technology is essentially the same as that used in sanitary and toxic waste landfills, and thus can be used in these applications. Our general approach has been to determine the unsaturated flow of the combination of two or more materials with significantly different properties.

A protocol was established for sample handling and testing to provide reproducible results. The study was limited to the properties of material common to the Quaternary deposits of Illinois. Engineering tests include: Atterburg limits, particle size, moisture-density relationship, dispersion, and shrink and swell. Hydraulic tests include: falling head permeability and moisture content-capillary pressure. A short study of the applicability of geofabrics is also included.

Infiltration into layered covers was studied using a one-dimensional finite difference computer solution to the equations governing the unsaturated movement of moisture. The computer program (MOMOLS) was written by Stu Rojstaczer; explanation of the theory and numerical analysis is included in the Interim Task II Report, September 30, 1981.

It was observed at the outset of this study that soils compacted wet of optimum moisture content exhibited much different saturated hydraulic conductivities and soil moisture retention characteristics than soils compacted dry of optimum. Therefore, the simulations were carried out by using the physical characteristics associated with soils compacted wet and dry of optimum to see what the influence is on infiltration and percolation. Simulations were limited chiefly to one- and two-layer systems. Subjects which have been investigated include: surface boundary conditions; moisture entry into the top layer; moisture redistribution in the top layer and the influence of the second layer on

moisture breakthrough from the first. A summary of the results from this investigation are presented in a previous section of this paper.

A study was made of the use of the water balance method, focusing especially on how the method might be applied to estimate percolation through compacted soil covers over trenches containing low-level radioactive waste. A discussion of the method and an example of its application are presented.

There are many advantages and disadvantages to the method. One of the most attractive advantages is that it is very easy to do and yields an estimate of percolation without much computation effort. It would be quite expensive to simulate the behavior of a cover with a finite element or finite difference model for a whole year. Another advantage is that it allows evapotranspiration to be taken into account; this is important, because not all the water that infiltrates the cover percolates through it. Probably the largest disadvantage of this method is that it has never been tested and verified in the field. No one can really say how good the estimates are because there are no field results with which to compare them.

The theory and technology of monitoring of water movement in unsaturated materials was investigated. A summary of this theory as well as an overview of present practice in the measurement of storage properties and rate of movement is presented.

The task 3 report summarizes some of the important geotechnical engineering considerations for construction of compacted earth trench covers. The characteristics of certain soils which could lead to inadequate performance of the trench cover are briefly discussed.

#### TASK 4: FIELD TESTING

Designs for trench covers are to be tested in field scale plots. The test site is located adjacent to the Sheffield low-level disposal site and initial monitoring equipment has been installed. Final construction plans are being prepared.

#### FUTURE RESEARCH

##### LABORATORY

Continuing analysis of different materials will provide greater flexibility in computer modeling. We will especially work on obtaining hydraulic conductivity-moisture content relationships for coarse-textured materials.

The laboratory gamma-ray attenuation device will undergo further testing to enable the simultaneous measurement of bulk density changes and moisture content changes. After this capability is attained, we will run soil column tests duplicating the conditions of the field plots which are to be installed this autumn.

#### COMPUTER MODELING

Two-dimensional modeling will continue, especially at the smaller laboratory column scale. Different materials and designs will be tested.

#### FIELD STUDY

It is anticipated that four test covers will be constructed during the late summer and early autumn of 1982. The covers will be approximately 6 m wide, 20 m long, and 1.5 to 3 m thick. Several layered designs will be compared to a one-layer cover of compacted clay or loess. Instrumentation, including neutron density loggers and tensiometers will be installed. A concurrent study by investigators from the University of Arizona will monitor moisture movement through the covers using tracer techniques.

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## RADIONUCLIDE MIGRATION IN GROUNDWATER

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## ABSTRACT

The migration rates and mechanisms of a large suite of radionuclides in groundwater created by the disposal of low-level aqueous effluents from the Hanford N-Reactor are being characterized. These studies are developing information which can be applied to understanding and predicting the movement of radionuclides in shallow-land burial sites which may become saturated with groundwater. The movement of the various radionuclides in groundwater at the N-Reactor disposal site is being investigated as a function of their physicochemical forms. Those radionuclides most readily migrating include tritium and anionic species of  $^{131}\text{I}$ ,  $^{133}\text{I}$ ,  $^{60}\text{Co}$ ,  $^{103}\text{Ru}$ ,  $^{106}\text{Ru}$ ,  $^{122}\text{Sb}$ ,  $^{124}\text{Sb}$ ,  $^{125}\text{Sb}$ ,  $^{99\text{m}}\text{Tc}$ , and  $^{99}\text{Tc}$ . Also migrating are traces of  $^{238}\text{Pu}$ ,  $^{239}\text{Pu}$ , and  $^{240}\text{Pu}$  in an oxidized (V or VI) anionic form(s). Various parameters affecting the physicochemical speciation of the radionuclides, including the major and trace ions, organic constituents, colloidal materials, and soil chemistry are being characterized to determine their relative roles in the migration processes. The data generated in this study will be extremely valuable in formulating guidelines for the operation of existing low-level waste disposal sites, in determining what long-term surveillance and maintenance needs will be required, and in determining essential requirements for remedial action.

## INTRODUCTION

One of the major gaps in understanding the environmental behavior of the long-lived radionuclides with low- and high-level waste is an adequate knowledge of the mechanisms by which these radionuclides are mobilized and transported in groundwaters. Previous predictions of the rate of movement of radionuclides in groundwaters which were based on laboratory-derived  $K_d$  values can lead to erroneous conclusions concerning the groundwater mobility of various radionuclides. This is because the laboratory-derived  $K_d$  values were obtained using ideal soluble tracers which were probably not in the same physicochemical form(s) as the mobile radionuclide species known to be present in some groundwaters. In addition, most  $K_d$  experiments do not consider

the long-term leaching of soil-adsorbed radionuclides by ligands and complexing substances contained in groundwaters. To better understand the rates and mechanisms of groundwater transport of radionuclides it is necessary to examine such transport in actual situations and compare the observed behavior with the predicted rate of movement.

A unique opportunity exists at the Hanford 1301-N crib trench to study, under natural conditions, the transport of a large number of radionuclides in oxygenated groundwaters. At the 1301-N crib site, low level radioactivity contained in Hanford N-Reactor effluent water is continually discharged into a seepage crib consisting of a rock-lined basin, approximately 3300 m<sup>2</sup> in area which is connected to a narrow, unlined trench several hundred meters long. Approximately 8000 l/min. of effluent water are continuously discharged from the Hanford N-Reactor to this retention system which lies parallel to the Columbia River (see Figure 1). This effluent contains some primary coolant water and drainage waters from a number of other sources, including fuel storage basin water. The trench is situated on a small bluff overlooking the Columbia River. The soil bank between the trench and the river is composed of unconsolidated fluvial silt, sand, gravel, and rock. The effluent water percolates by gravity through the soil bank to the river and some of this water emerges along the riverbank as seepage springs. This system serves to remove most of the radioactivity in the effluent water during transit through the soil bank. However, movement through the soil results in selective removal of the more reactive physicochemical species, and the more mobile chemical forms percolate through the soil.

#### HISTORICAL BACKGROUND AND DESCRIPTION OF STUDY SITE

The Hanford N-Reactor is a dual purpose (power + plutonium) reactor located at Hanford, Washington on a bluff overlooking the Columbia River. The reactor began operation in 1964 and has operated somewhat continuously since startup. The reactor is presently operated for the Department of Energy by United Nuclear Industries, Incorporated (UNI), and steam is sold to the Washington Public Power Supply System.

A unique aqueous waste disposal system at the reactor site was designed and built to accommodate radioactive aqueous effluents released from the reactor. This system, known as the 1301-N facility, is located northeast of the 105-N Reactor Building on a riverbank bluff 18 m above and 265 m back from the river. The crib is a 38 m by 86 m basin connected to a 490 m by 15 m extension trench. The system is designed such that the maximum crib water level is well below the surface of a three-foot layer of large boulders lining the crib bottom. The extension trench was formerly covered with wire mesh to prevent access by birds and larger animals. Last year the trench was completely covered with preformed concrete slabs.

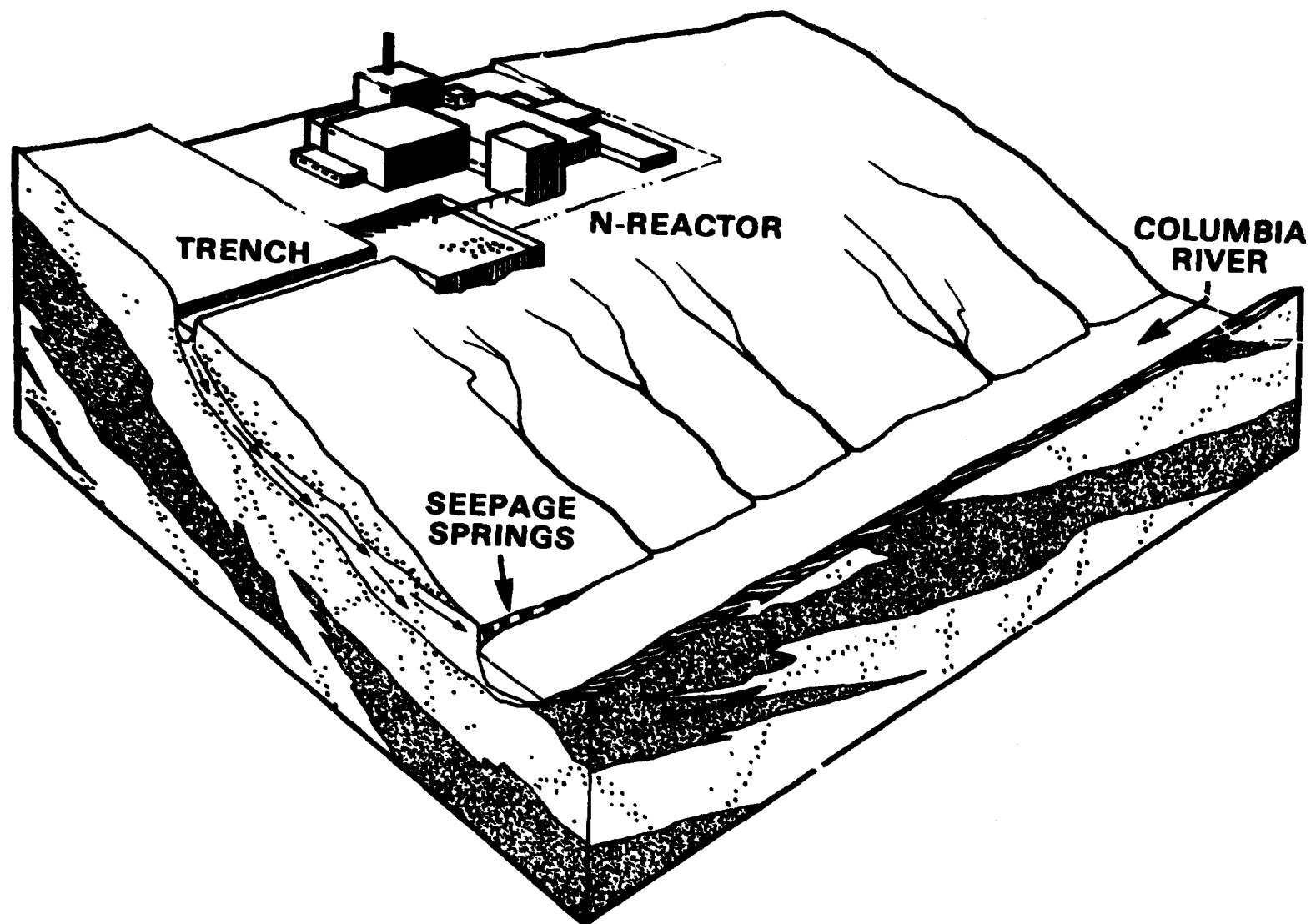


Figure 1. Schematic Cross-Sectional View of N-Reactor Liquid Effluent Disposal System



The annual releases of radioactivity to the 1301-N crib during most of the period of its existence have been monitored by UNI, Incorporated, the operators of the reactor. Table 1 presents the 1978 average and peak concentrations of twenty-eight radionuclides in crib and spring waters, and the total curies of these radionuclides released to the 1301-N crib from N-Reactor and that released to the Columbia River via the seepage springs (USDOE, 1980).

The annual release data has been utilized to develop an estimate of the total cumulative inventory of some of the long-lived radionuclides in the 1301-N crib. These estimates are given in Table 2. Not included in this estimate is undoubtedly a large inventory of other long-lived radionuclides including  $^{55}\text{Fe}$ ,  $^{63}\text{Ni}$ ,  $^{99}\text{Tc}$ ,  $^{125}\text{Sb}$ ,  $^{129}\text{I}$ , and  $^{152,154,155}\text{Eu}$ . Substantial amounts of  $^{93}\text{Zr-Nb}$  and  $^{94}\text{Nb}$  may also be present.

TABLE 2  
Radionuclide Burden of 1301-N Facility at End of 1978  
(from USDOE, 1980)

| <u>Radionuclide</u> | <u>Facility Burden<br/>(Ci)</u> |
|---------------------|---------------------------------|
| $^{60}\text{Co}$    | 3100                            |
| $^{90}\text{Sr}$    | 1400                            |
| $^{137}\text{Cs}$   | 1800                            |
| $^{239}\text{Pu}$   | 11                              |

The soil bank acts as a relatively efficient sorption bed for a large number of radionuclides, whereas others migrate with the groundwater. The annual average transmission factors for a number of radionuclides migrating from the crib to the springs is given in Table 3.

TABLE 1  
Discharge of Radionuclides to 1301-N Facility and  
Seepage to River During CY 1978 (from USDOE, 1980)

| Radionuclide          | To the 1301-N Facility              |                                  |                           | Seepage from Facility to<br>River Via Springs <sup>(a)</sup> |                                  |                           | Decontamination<br>Factor <sup>(b)</sup> |
|-----------------------|-------------------------------------|----------------------------------|---------------------------|--------------------------------------------------------------|----------------------------------|---------------------------|------------------------------------------|
|                       | Average<br>Concentration<br>(pCi/l) | Peak<br>Concentration<br>(pCi/l) | Total<br>Released<br>(Ci) | Average<br>Concentration<br>(pCi/l)                          | Peak<br>Concentration<br>(pCi/l) | Total<br>Released<br>(Ci) |                                          |
| <sup>3</sup> H        | 6.9x10 <sup>4</sup>                 | 2.9x10 <sup>5</sup>              | 3.3x10 <sup>2</sup>       | 5.8x10 <sup>4</sup>                                          | 1.5x10 <sup>5</sup>              | 3.3x10 <sup>2</sup>       | 1                                        |
| <sup>32</sup> P       | 1.2x10 <sup>4</sup>                 | 6.6x10 <sup>4</sup>              | 5.6x10 <sup>1</sup>       | 1.3x10 <sup>1</sup>                                          | 2.1x10 <sup>2</sup>              | 2.9x10 <sup>-2</sup>      | 1,900                                    |
| <sup>51</sup> Cr      | 7.9x10 <sup>4</sup>                 | 2.1x10 <sup>5</sup>              | 3.7x10 <sup>2</sup>       | 6.4x10 <sup>1</sup>                                          | 7.7x10 <sup>1</sup>              | 1.4x10 <sup>-1</sup>      | 2,600                                    |
| <sup>54</sup> Mn      | 1.7x10 <sup>5</sup>                 | 2.8x10 <sup>5</sup>              | 8.1x10 <sup>2</sup>       | ND                                                           | ND                               | ND                        | ---                                      |
| <sup>59</sup> Fe      | 1.6x10 <sup>5</sup>                 | 7.4x10 <sup>2</sup>              | 7.6x10 <sup>2</sup>       | ND                                                           | ND                               | ND                        | ---                                      |
| <sup>58</sup> Co      | 1.4x10 <sup>4</sup>                 | 3.1x10 <sup>4</sup>              | 6.4x10 <sup>1</sup>       | 7.8x10 <sup>0</sup>                                          | 8.9x10 <sup>0</sup>              | 1.7x10 <sup>-2</sup>      | 3,800                                    |
| <sup>60</sup> Co      | 2.0x10 <sup>5</sup>                 | 5.6x10 <sup>5</sup>              | 9.4x10 <sup>2</sup>       | 1.8x10 <sup>2</sup>                                          | 3.9x10 <sup>2</sup>              | 4.1x10 <sup>-1</sup>      | 2,300                                    |
| <sup>65</sup> Zn      | 8.3x10 <sup>3</sup>                 | 1.3x10 <sup>4</sup>              | 3.9x10 <sup>1</sup>       | ND                                                           | ND                               | ND                        | ---                                      |
| <sup>89</sup> Sr      | 1.1x10 <sup>5</sup>                 | 1.8x10 <sup>6</sup>              | 5.2x10 <sup>2</sup>       | 1.8x10 <sup>2</sup>                                          | 4.8x10 <sup>2</sup>              | 4.0x10 <sup>-1</sup>      | 1,300                                    |
| <sup>90</sup> Sr      | 2.5x10 <sup>4</sup>                 | 9.6x10 <sup>4</sup>              | 1.2x10 <sup>2</sup>       | 5.9x10 <sup>2</sup>                                          | 1.5x10 <sup>3</sup>              | 1.3x10 <sup>0</sup>       | 90                                       |
| <sup>95</sup> ZrNb    | 1.3x10 <sup>5</sup>                 | 1.1x10 <sup>6</sup>              | 1.2x10 <sup>3</sup>       | ND                                                           | ND                               | ND                        | ---                                      |
| <sup>99m</sup> MoTc   | 3.3x10 <sup>5</sup>                 | 5.6x10 <sup>6</sup>              | 1.6x10 <sup>3</sup>       | 5.9x10 <sup>2</sup>                                          | 1.4x10 <sup>3</sup>              | 1.3x10 <sup>0</sup>       | 1,200                                    |
| <sup>103</sup> Ru     | 6.3x10 <sup>4</sup>                 | 5.6x10 <sup>5</sup>              | 3.0x10 <sup>2</sup>       | 1.4x10 <sup>2</sup>                                          | 4.3x10 <sup>2</sup>              | 3.1x10 <sup>-1</sup>      | 970                                      |
| <sup>106</sup> Ru     | 7.0x10 <sup>4</sup>                 | 1.6x10 <sup>5</sup>              | 3.3x10 <sup>2</sup>       | 2.2x10 <sup>2</sup>                                          | 4.4x10 <sup>2</sup>              | 4.9x10 <sup>-1</sup>      | 670                                      |
| <sup>124</sup> Sb     | 1.3x10 <sup>4</sup>                 | 1.2x10 <sup>4</sup>              | 6.2x10 <sup>1</sup>       | 2.8x10 <sup>1</sup>                                          | 8.0x10 <sup>1</sup>              | 6.2x10 <sup>-2</sup>      | 1,000                                    |
| <sup>125</sup> Sb     | 7.6x10 <sup>2</sup>                 | 7.5x10 <sup>2</sup>              | 4.1x10 <sup>0</sup>       | 7.7x10 <sup>1</sup>                                          | 3.1x10 <sup>2</sup>              | 1.7x10 <sup>-1</sup>      | 20                                       |
| <sup>129</sup> I      | 6.7x10 <sup>-3</sup>                | ---                              | 1.4x10 <sup>-5</sup>      | 3.6x10 <sup>-3</sup>                                         | ---                              | 8.2x10 <sup>-6</sup>      | 2                                        |
| <sup>131</sup> I      | 1.2x10 <sup>5</sup>                 | 1.0x10 <sup>6</sup>              | 5.6x10 <sup>2</sup>       | 1.7x10 <sup>3</sup>                                          | 2.7x10 <sup>4</sup>              | 3.8x10 <sup>0</sup>       | 150                                      |
| <sup>133</sup> Xe     | 1.1x10 <sup>5</sup>                 | 8.3x10 <sup>5</sup>              | 5.2x10 <sup>2</sup>       | 2.6x10 <sup>3</sup>                                          | 8.3x10 <sup>3</sup>              | 5.7x10 <sup>0</sup>       | 90                                       |
| <sup>134</sup> Cs     | 1.4x10 <sup>4</sup>                 | 4.6x10 <sup>4</sup>              | 6.7x10 <sup>1</sup>       | ND                                                           | ND                               | ND                        | ---                                      |
| <sup>137</sup> Cs     | 7.3x10 <sup>4</sup>                 | 2.4x10 <sup>5</sup>              | 3.4x10 <sup>2</sup>       | 7.1x10 <sup>0</sup>                                          | 1.0x10 <sup>1</sup>              | 1.6x10 <sup>-2</sup>      | 21,000                                   |
| <sup>140</sup> BaLa   | 7.3x10 <sup>5</sup>                 | 1.4x10 <sup>7</sup>              | 7.1x10 <sup>3</sup>       | ND                                                           | ND                               | ND                        | ---                                      |
| <sup>141</sup> Ce     | 1.0x10 <sup>5</sup>                 | 1.2x10 <sup>6</sup>              | 4.9x10 <sup>2</sup>       | ND                                                           | ND                               | ND                        | ---                                      |
| <sup>144</sup> CePr   | 1.4x10 <sup>5</sup>                 | 5.6x10 <sup>5</sup>              | 6.6x10 <sup>2</sup>       | ND                                                           | ND                               | ND                        | ---                                      |
| <sup>154</sup> Eu     | 1.2x10 <sup>4</sup>                 | 1.3x10 <sup>4</sup>              | 5.5x10 <sup>1</sup>       | ND                                                           | ND                               | ND                        | ---                                      |
| <sup>155</sup> Eu     | 7.3x10 <sup>3</sup>                 | 7.3x10 <sup>4</sup>              | 3.4x10 <sup>1</sup>       | ND                                                           | ND                               | ND                        | ---                                      |
| <sup>238</sup> Pu     | 5.5x10 <sup>1</sup>                 | 1.2x10 <sup>2</sup>              | 2.6x10 <sup>-1</sup>      | 5.7x10 <sup>-2</sup>                                         | 3.3x10 <sup>-1</sup>             | 1.3x10 <sup>-4</sup>      | 2,000                                    |
| <sup>239/240</sup> Pu | 2.7x10 <sup>2</sup>                 | 5.6x10 <sup>2</sup>              | 1.3x10 <sup>0</sup>       | 1.1x10 <sup>-1</sup>                                         | 1.3x10 <sup>0</sup>              | 3.0x10 <sup>-4</sup>      | 4,000                                    |

(a) Estimated Flow = 900 gpm.

(b) DF =  $\frac{\text{Total Released Facility}}{\text{Total Released Springs}}$

ND = Radionuclide not detected in springs but were measured in effluent to the facility.

TABLE 3  
1301-N Facility Transmission Factors-Percent (from USDOE, 1980)

| <u>Radionuclide</u>  | <u>1974</u> | <u>1975</u> | <u>1976</u> | <u>1977</u> | <u>1978</u> |
|----------------------|-------------|-------------|-------------|-------------|-------------|
| <sup>3</sup> H       | 100         | 100         | 100         | 100         | 100         |
| <sup>60</sup> Co     | 0.06        | 0.08        | 0.11        | 0.07        | 0.04        |
| <sup>90</sup> Sr     | 0.47        | 0.43        | 0.32        | 0.83        | 1.11        |
| <sup>95</sup> Zr-Nb  | 0.01        | (a)         | (a)         | (a)         | (a)         |
| <sup>99</sup> Mo     | 0.03        | 0.06        | 0.02        | 0.06        | 0.08        |
| <sup>106</sup> Ru    | 0.63        | 0.45        | 0.33        | 0.29        | 0.15        |
| <sup>131</sup> I     | 0.47        | 0.42        | 0.22        | 0.77        | 0.67        |
| <sup>137</sup> Cs    | 0.03        | (b)         | (a)         | 0.01        | (b)         |
| <sup>140</sup> Ba-La | 0.05        | (a)         | (a)         | (a)         | (a)         |

(a) Radionuclide detected at crib but not at springs.

(b) Not significant.

## RESULTS AND DISCUSSIONS

Crib and spring waters have been sampled on a periodic basis to determine the physicochemical partitioning of radionuclides between particulate forms and soluble cationic, anionic, and nonionic species. A special sampling technique is utilized in which water is pumped directly from the crib inlet and from the springs through assemblies containing membrane filters, ion exchange resins and activated aluminum oxide beds.<sup>(2)</sup>

The filters remove the particulate forms of the radionuclides, the resins remove soluble cationic and anionic species and the activated aluminum oxide removes nonionic species. Five to ten liters of crib water and up to 2000 liters of spring water are typically processed through these assemblies.

Radionuclide concentrations observed in particulate, anionic, cationic, and nonanionic forms in crib and spring waters are shown in Table 4. Most of the radionuclides enter the crib from the reactor associated with particulate forms. The soluble radionuclide species present in the crib water are predominantly cationic forms, the exceptions being <sup>131</sup>I and <sup>125</sup>Sb. <sup>60</sup>Co, <sup>54</sup>Mn, <sup>137</sup>Cs, <sup>144</sup>Ce, <sup>90</sup>Zr, <sup>95</sup>Nb,

**TABLE 4**  
**Radionuclide Concentrations and Physicochemical Forms in Crib and Spring Waters**

| PHYSICOCHEMICAL FORM | RADIONUCLIDE CONCENTRATIONS - pCi /liter |                  |                  |                  |                  |                  |                  |                  |                  |                  |                   |                   |                   |                   |                  |                   |                   |                   |                   |                   |
|----------------------|------------------------------------------|------------------|------------------|------------------|------------------|------------------|------------------|------------------|------------------|------------------|-------------------|-------------------|-------------------|-------------------|------------------|-------------------|-------------------|-------------------|-------------------|-------------------|
|                      | <sup>46</sup> Sc                         | <sup>51</sup> Cr | <sup>54</sup> Mn | <sup>57</sup> Co | <sup>58</sup> Co | <sup>59</sup> Fe | <sup>60</sup> Co | <sup>65</sup> Zn | <sup>95</sup> Zr | <sup>95</sup> Nb | <sup>103</sup> Ru | <sup>106</sup> Ru | <sup>124</sup> Sb | <sup>125</sup> Sb | <sup>131</sup> I | <sup>134</sup> Cs | <sup>137</sup> Cs | <sup>140</sup> Bs | <sup>141</sup> Ce | <sup>144</sup> Ce |
| N-CRIB WATER         |                                          |                  |                  |                  |                  |                  |                  |                  |                  |                  |                   |                   |                   |                   |                  |                   |                   |                   |                   |                   |
| 3/10/81              |                                          |                  |                  |                  |                  |                  |                  |                  |                  |                  |                   |                   |                   |                   |                  |                   |                   |                   |                   |                   |
| PARTICULATE          | <13                                      | <90              | 22,200           | <3               | <13              | <30              | 10,500           | 60.4             | 67.1             | 338              | <15               | 448               | 11.4              | 66.7              | <14              | 3630              | 25,900            | <70               | <7                | 1086              |
| ANIONIC              | <0.2                                     | 4.2              | 6.3              | 0.45             | <0.2             | <0.5             | 34.8             | 1.1              | 14               | 48               | 4.7               | 230               | 1.6               | 121               | <0.2             |                   | <0.2              | <1                | 0.48              | 94.9              |
| CATIONIC             | 31.4                                     | <90              | 14,110           | 52.7             | 148              | 575              | 70,690           | 940              | 1768             | 4899             | 135               | 11,340            | <36               | 147               | <13              |                   | 1020              | <80               | 45.7              | 19,350            |
| NONIONIC             | <0.07                                    | 2.4              | 0.14             | <0.02            | <0.05            | <0.1             | 2.5              | <0.1             | 2.0              | 0.93             | 0.2               | 16                | 1.3               | 72.4              | <0.09            |                   | <0.1              | <0.3              | <0.1              | 4.2               |
| 6/17/81              |                                          |                  |                  |                  |                  |                  |                  |                  |                  |                  |                   |                   |                   |                   |                  |                   |                   |                   |                   |                   |
| PARTICULATE          | ---                                      | ---              | 4220             | 38.7             | 297              | 4697             | 21,670           | 222              | 760              | 11,190           | 546               | 2790              | ---               | 71.4              | ---              | 47.4              | 475               | ---               | ---               | 7690              |
| ANIONIC              | 1.5                                      | 16.1             | 151              | 2.5              | 4.0              | 121              | 132              | 1.5              | 96.9             | 231              | 98.3              | 445               | 17.1              | 209               | 166              | 0.6               | 2.9               | 23.2              | 39                | 440               |
| CATIONIC             | <25                                      | <280             | 24,400           | 16.6             | <27              | <69              | 12,000           | 96.5             | 107              | 335              | 40.6              | 290               | <25               | 105               | <140             | 7090              | 58,860            | 2110              | 141               | 1780              |
| NONIONIC             | <0.2                                     | 4.8              | 0.89             | 0.28             | <0.2             | 1.5              | 7.8              | <0.4             | 18               | 9.1              | 6.2               | 40.6              | 12.1              | 136               | <0.6             | <0.1              | <0.1              | <1.7              | 4.7               | 55                |
| 9/22/81              |                                          |                  |                  |                  |                  |                  |                  |                  |                  |                  |                   |                   |                   |                   |                  |                   |                   |                   |                   |                   |
| PARTICULATE          | ---                                      | ---              | 3940             | 11.7             | 632              | 3240             | 12,120           | 289              | 2030             | 9490             | 273               | 1780              | ---               | 62.2              | ---              | 27.1              | 138               | ---               | 151               | 2930              |
| ANIONIC              | <3.5                                     | <75              | 490              | 6.8              | <4.1             | 97.9             | 555              | 40.8             | 38.7             | 170              | <7.6              | 81.9              | 5.9               | 37.4              | 65,630           | 6.8               | 14                | 4590              | <35               | <123              |
| CATIONIC             | <13                                      | <110             | 14,220           | 5.9              | 2240             | 42.3             | 4800             | 334              | <23              | 36.5             | 20.0              | 160               | 44.9              | 48                | 2960             | 2440              | 31,250            | 97,880            | <10               | 602               |
| NONIONIC             | <0.8                                     | 7.7              | 8.8              | <0.3             | <0.8             | <1.8             | 14.7             | <2.2             | 9.1              | 1.6              | <0.8              | 31.9              | 5.84              | 73.6              | 27               | <0.9              | <0.8              | 33                | <0.8              | <3                |
| N-SPRING WATER       |                                          |                  |                  |                  |                  |                  |                  |                  |                  |                  |                   |                   |                   |                   |                  |                   |                   |                   |                   |                   |
| 3/3/81               |                                          |                  |                  |                  |                  |                  |                  |                  |                  |                  |                   |                   |                   |                   |                  |                   |                   |                   |                   |                   |
| PARTICULATE          | <0.003                                   | <0.02            | <0.002           | <0.001           | <0.001           | <0.004           | 0.31             | <0.004           | <0.003           | <0.002           | <0.002            | 0.19              | <0.005            | 0.05              | <0.002           | <0.002            | <0.002            | <0.008            | <0.001            | 0.10              |
| ANIONIC              | <0.04                                    | <0.22            | <0.04            | <0.01            | <0.03            | <0.09            | 23.0             | 0.17             | <0.07            | 0.14             | 1.51              | 89.2              | 0.32              | 18.7              | <0.05            | <0.03             | <0.02             | <0.1              | <0.02             | <0.06             |
| CATIONIC             | <0.02                                    | <0.15            | <0.03            | <0.007           | <0.02            | <0.07            | 22.9             | <0.08            | <0.03            | <0.02            | 0.08              | 9.6               | <0.02             | 0.41              | <0.02            | <0.02             | <0.01             | <0.1              | <0.01             | 0.11              |
| NONIONIC             | <0.006                                   | 3.2              | <0.01            | <0.004           | <0.004           | <0.01            | 2.7              | <0.01            | <0.01            | <0.006           | 0.06              | 3.5               | 1.50              | 78.0              | <0.02            | <0.004            | <0.004            | <0.06             | <0.01             | <0.03             |
| 6/17/81              |                                          |                  |                  |                  |                  |                  |                  |                  |                  |                  |                   |                   |                   |                   |                  |                   |                   |                   |                   |                   |
| PARTICULATE          | <0.003                                   | <0.04            | <0.001           | <0.0004          | <0.002           | <0.01            | 0.30             | <0.003           | <0.004           | <0.005           | <0.004            | 0.29              | <0.004            | <0.008            | <1.2             | <0.001            | <0.002            | <0.3              | <0.04             | 0.07              |
| ANIONIC              | <0.02                                    | 1.1              | <0.01            | <0.004           | 0.027            | <0.04            | 23.5             | <0.07            | <0.02            | <0.02            | 1.30              | 65.5              | <0.02             | 6.2               | 15.9             | <0.01             | <0.02             | <0.1              | <0.01             | <0.03             |
| CATIONIC             | <0.02                                    | <0.17            | <0.02            | <0.004           | 0.037            | <0.04            | 23.7             | <0.05            | <0.02            | <0.03            | 0.21              | 9.0               | <0.01             | 0.32              | 0.15             | <0.01             | <0.01             | <0.1              | <0.01             | <0.02             |
| NONIONIC             | <0.02                                    | 4.0              | <0.01            | <0.004           | <0.008           | <0.03            | 4.9              | <0.03            | <0.01            | <0.006           | 0.08              | 5.4               | 0.49              | 76.0              | 0.5              | <0.006            | <0.007            | <0.07             | <0.01             | <0.03             |
| 9/22/81              |                                          |                  |                  |                  |                  |                  |                  |                  |                  |                  |                   |                   |                   |                   |                  |                   |                   |                   |                   |                   |
| PARTICULATE          | <0.02                                    | <0.2             | <0.009           | <0.004           | <0.01            | <0.04            | 0.42             | <0.02            | <0.02            | <0.02            | <0.02             | <0.1              | <0.03             | 0.10              | 2.5              | <0.01             | <0.02             | <0.5              | <0.02             | <0.004            |
| ANIONIC              | 0.14                                     | 3.0              | <0.1             | <0.01            | <0.08            | <0.2             | 162.9            | 0.98             | <0.1             | <0.1             | 2.31              | 112.2             | 0.31              | 22.4              | 2499             | 0.09              | 0.28              | 3.2               | <0.06             | <0.08             |
| CATIONIC             | <0.13                                    | <0.8             | <0.1             | <0.03            | <0.04            | 0.59             | 64.8             | 0.74             | <0.1             | <0.1             | 0.63              | 28.1              | <0.1              | 0.58              | 60               | <0.1              | 0.30              | <0.4              | <0.06             | <0.2              |
| NONIONIC             | <0.02                                    | 4.6              | <0.02            | <0.02            | <0.02            | <0.06            | 8.3              | <0.1             | <0.03            | <0.03            | 0.16              | 7.6               | 1.33              | 117               | 5.4              | <0.02             | <0.04             | <0.1              | <0.06             | <0.04             |

$^{106}\text{Ru}$ , and  $^{59}\text{Fe}$  are the predominant gamma-emitters. When the reactor is operating a number of short- to intermediate-lived radionuclides are discharged to the crib, including the radioiodines ( $^{131}\text{I}$ ,  $^{132}\text{I}$ ,  $^{133}\text{I}$ ),  $^{140}\text{Ba}$ - $^{140}\text{La}$  and  $^{99}\text{Mo}$ .

As the crib water percolates through the soil bank to the riverbank springs the particulate and cationic species are efficiently removed by filtration, sorption, and ion exchange mechanisms by the soil. The emerging radionuclides in the riverbank springs are primarily those which exist in anionic or nonionic species, i.e.,  $^{51}\text{Cr}$ ,  $^{60}\text{Co}$ ,  $^{106}\text{Ru}$ ,  $^{125}\text{Sb}$ , and  $^{131}\text{I}$ .

The percentages of these mobile radionuclides in their various physicochemical forms observed in the crib and spring waters are shown in Table 5.

The  $^{60}\text{Co}$  in the crib water is mainly in a particulate or soluble cationic form, whereas in the spring water it is found partitioned mainly between anionic and cationic forms.

Ruthenium-106 is present in the crib water in highly variable forms. During the March 10, 1981 sampling, the  $^{106}\text{Ru}$  was present mainly in a cationic form, whereas on June 17, 1981 and September 22, 1981 the particulate associated  $^{106}\text{Ru}$  was prevalent, and the anionic and cationic forms were approximately equivalent. However, the  $^{106}\text{Ru}$  observed in the spring waters was always present in a predominantly anionic form.

The  $^{125}\text{Sb}$  in the crib water was frequently partitioned in all measured forms in nearly equal amounts. In the spring water, 80-92% of the  $^{125}\text{Sb}$  was in a nonionic form, the remainder being present as an anion.

Chromium-51 could not be detected in the gamma-ray spectra of the crib waters due to masking effects from the Compton continuum from the relatively high concentrations of other higher energy radionuclides. However, in the spring waters the  $^{51}\text{Cr}$  was present mainly (54-89%) in a nonionic form, with smaller percentages (21-36%) in an anionic form.

The  $^{131}\text{I}$ , as expected, was present in both crib and spring waters mainly (96-97%) in an anionic form.

Rai and Serne(3) have predicted from thermodynamic data the nature of the soil solution species of a number of elements in various weathering environments. The major predicted species of antimony expected under oxygenated conditions and the pH observed for crib and spring waters is  $\text{HSbO}_2^\circ$ ,  $\text{Sb(OH)}_3^\circ$ , and  $\text{SbO}_2^-$  which is in good agreement with our analytical results. The predicted ruthenium species are given as  $\text{RuO}_4^-$ ,  $\text{RuO}_4^{2-}$ , and  $\text{RuO}_4^\circ$ , which again is in good agreement with our empirical data. The predicted major species for cobalt are  $\text{Co}^{+2}$ ,  $\text{CoOH}^+$ , and  $\text{CoSO}_4$ . These could explain the cationic form(s) of  $^{60}\text{Co}$  in the spring water. The only anionic species predicted by Rai and Serne of any significance is  $\text{Co(OH)}_2^-$ , but this was predicted to be present at

concentrations many orders of magnitude below the major cation and nonionic cobalt species.

#### DISSOLVED ORGANIC MATTER

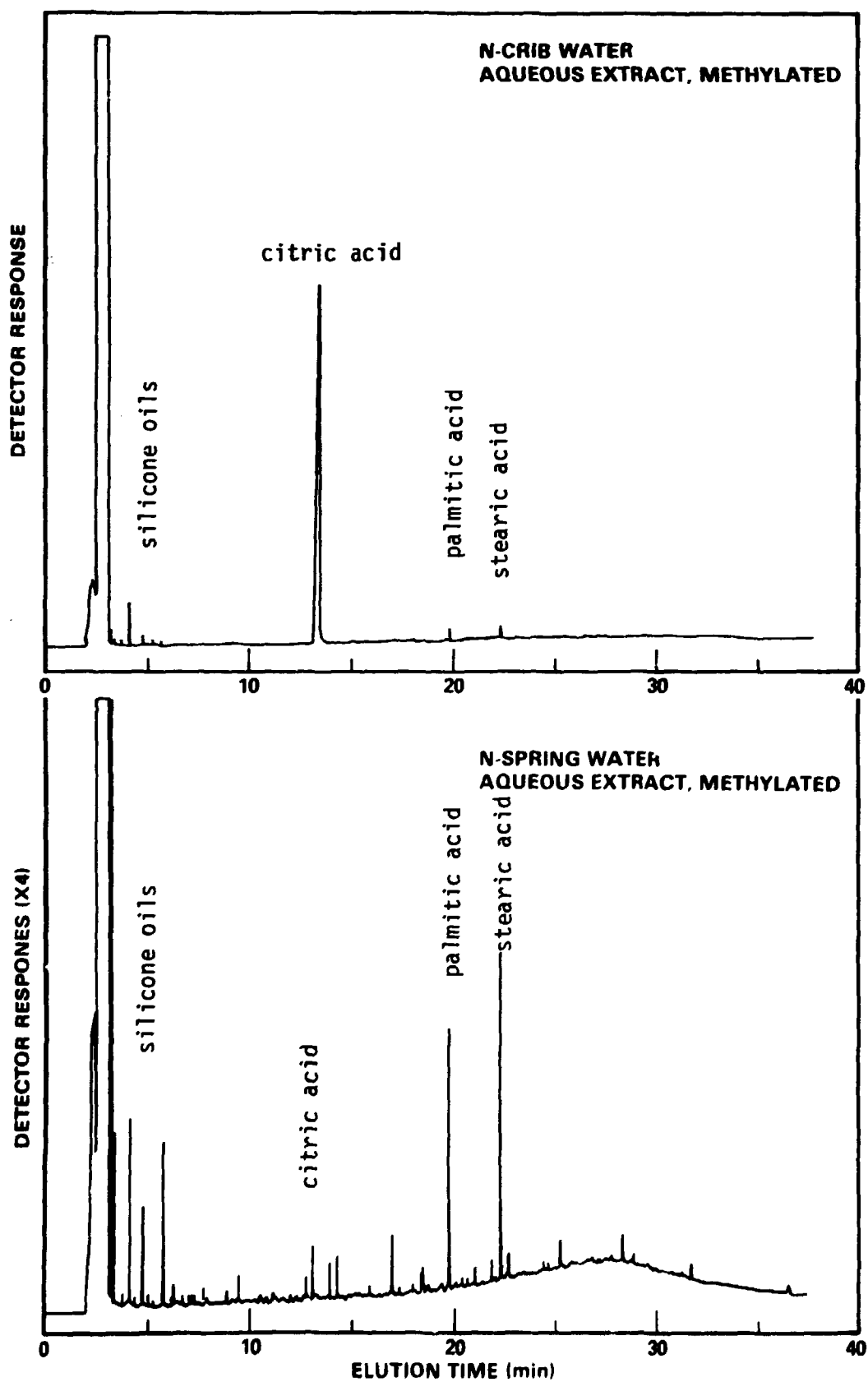
The dissolved organic carbon concentrations in the crib and spring waters were typical of natural surface waters, generally ranging between 2 to 3 mg/l. These concentrations are considered sufficiently high to complex metal radionuclide species and thus affect their mobility in groundwaters. Therefore, a comprehensive identification and quantification of the major and minor dissolved organic constituents in the crib and spring water was initiated utilizing gas chromatograph-mass spectrometry (GC-MS).

The results of the GC-MS analyses are shown in Figure 2. In the crib water sample one major peak, identified as citrate, was present. Also, two minor constituents identified as palmitic and stearic acids were present, as well as trace amounts of silicone oils. In the spring water the citrate concentration was very low compared to the palmitic and stearic acids and the silicone oils. The citrate in the crib water sample was obviously added to the reactor effluent water at some point in the reactor building. Citrate is a common decontamination reagent and could have entered the effluent water line from a variety of reactor building drains. Citrate is also a good complexing agent for a number of the transition metal radionuclides and could be an effective agent in enhancing the mobility of several radionuclides in the groundwater system at the 1301-N site.

The C-16 and C-18 carboxylic acid groups, palmitic and stearic acid, respectively, were the major dissolved organic constituents in the spring water. These acids can be formed naturally by decomposition of vegetable matter and are categorized in the so-called humic acid group. The 1301-N trench does contain substantial amounts of decaying organic matter, which could be the source of these acids. However, palmitic and stearic acids are also used in soaps and detergents and this could also be a source of these substances. The humic acids have the ability to complex iron and other transition metal radionuclides, and could enhance their groundwater mobility at the 1301-N site. We are continuing to characterize the organic matter in the 1301-N system to determine its role in the migration of radioactivity in the groundwater at this site.

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## INSTRUMENTATION FOR THE VERIFICATION OF THE RADIONUCLIDE CONTENT OF WASTE PACKAGES

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The objective of this research program is to organize, apply, and develop as required, instrumentation for assaying the radionuclide content of waste packages received by land burial operators, both to verify the data provided by the generator/shipper and to provide the most accurate records possible of buried materials. This DOE National Low Level Waste Management Program effort will provide technology for monitoring the accurate inventory of wastes interred at a low level disposal site, both shallow land and greater confinement. It will also ensure that all packages conform to the proper disposal categories prescribed by regulation, most notably those proposed in 10 CFR 61. Finally, an accurate inventory of the radionuclide content of a disposal site will provide a usable source term for modeling efforts attempting to predict the effectiveness of confinement or the magnitude of potential releases.

The waste package parameters and measurement requirements have been described in the previous edition of this series<sup>(1)</sup> including the existing applicable instrumentation capabilities. The major efforts this past year have concentrated on assembling a field test instrument package, planning a field test on actual commercial waste packages, development of data reduction techniques for interpreting data obtained both in these field tests and with the final verification instrumentation system, and preparation of a set of "standard" waste packages containing known concentrations of radionuclides in typical matrix materials for certification and calibration of waste assay systems.

Obviously, it would be ideal if all radionuclides listed in 10 CFR 61 could be analyzed simultaneously, quantitatively, and nondestructively in a sealed waste package. Just as obviously, that is not possible. Consequently, a modified MEGAS system with a simple transuranic neutron detection attachment has been assembled for use in the field tests to determine exactly what radioactivities are measurable in commercial waste packages and what modifications may be employed to measure additional radioisotopes. In order to certify the concentrations of all activities which may be present, either for transportation or inventory requirements, it will be necessary to perform analog studies on typical waste streams to establish upper limits for the concentration of those activities which cannot be measured based on the concentration of those which can. Such studies are currently planned by both EPRI

and the NRC. In terms of disposal site operation and confinement effectiveness, the most important radionuclides are  $^3\text{H}$ ,  $^{14}\text{C}$ ,  $^{90}\text{Sr}$ ,  $^{99}\text{Tc}$ ,  $^{129}\text{I}$ ,  $^{237}\text{Np}$ , and the transuranics. This is based on their potential health hazards, mobility, and long half life. Since non-destructive analysis techniques only exist currently for  $^{90}\text{Sr}$ ,  $^{237}\text{Np}$ , and the transuranics, the field test assay system has been assembled with particular attention to these isotopes and those gamma-ray emitting radionuclides which may be measured quantitatively.

The test equipment will utilize state-of-the-art high resolution germanium diode gamma-ray spectrometric equipment to measure the photon emitting radioisotopes based on the intensity of their characteristic emission lines and to measure  $^{90}\text{Sr}$  based on the intensity of its bremsstrahlung radiation. Rather elaborate computer data reduction techniques are necessary to determine the absolute intensity of the bremsstrahlung radiation. Basically, the procedure involves first identifying and quantifying the gamma-ray emitting isotopes present in the spectrum based on their full energy photopeaks and then "stripping out" each component utilizing an intensity normalized reference library spectrum leaving only  $^{90}\text{Sr}$  bremsstrahlung radiation as the residual component. The  $^{90}\text{Sr}$  concentration is then determined by comparison to a library reference spectrum on the basis of intensity. The entire process requires acquisition of the net reference library of spectra from isotopically pure standards so that their shape duplicates the corresponding component in the unknown spectrum, gain normalizing all spectra so that the photon energies in the unknown and reference spectra coincide, intensity normalization so that subtraction of the various components is quantitative, and an unnecessary but cosmetic smoothing function for displaying a more typical final product. The details of this data handling process are discussed in another paper in this document<sup>(2)</sup> and will not be repeated here.

An assembly of  $^{10}\text{BF}_3$  proportional counters will be used as active neutron detectors to measure the transuranic generated neutron flux emitted by the waste package. Determination of the flux is straightforward, and conversion to absolute transuranic concentrations is done by comparison to pure standards and by use of a computer generated calibration factor which includes the variable parameters of relative transuranic isotope composition, chemical composition of the matrix, residual moisture, the waste form, and a geometry factor. This calibration program has also been discussed elsewhere<sup>(3)</sup> and will not be repeated here.

In order to facilitate the above required calibrations and to provide a referenceable set of standard waste packages for calibration or certification of future assay systems, individual sources of standardized  $^3\text{H}$ ,  $^{14}\text{C}$ ,  $^{57}\text{Co}$ ,  $^{60}\text{Co}$ ,  $^{90}\text{Sr}$ ,  $^{99}\text{Tc}$ ,  $^{106}\text{Ru}$ ,  $^{125}\text{Sb}$ ,  $^{129}\text{I}$ ,  $^{133}\text{Ba}$ ,  $^{134}\text{Cs}$ ,  $^{137}\text{Cs}$ ,  $^{144}\text{Ce}$ ,  $^{152}\text{Eu}$ ,  $^{226}\text{Ra}$ ,  $^{228}\text{Th}$ ,  $^{237}\text{Np}$ , and  $^{241}\text{Am}$  have been ordered. Fabrication of waste packages containing these isotopes, both individually and in combination, in common matrix materials, will be completed in the near future.

( One setback has been experienced in the program during the past year. Arrangements were made with the U. S. Ecology Inc., Richland disposal site operator to test the instrumentation and procedures on random samples of actual commercial waste packages received at that facility. Permission to proceed was verbally requested and granted by the U. S. Ecology, Louisville home office who indicated that an authorization letter would be sent to both PNL and their site operator. That letter was never received and after three months of follow-up phone calls and letters, the initial offer was withdrawn. Alternate field tests will now be planned, and Chem Nuclear Systems in Barnwell, South Carolina will be contacted first.

Prepared for the U.S. Department of Energy under Contract DE-AC06-76RLO 1830

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RELEVANCE OF BIOTIC PATHWAYS TO THE REGULATION  
OF NUCLEAR WASTE DISPOSAL

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ABSTRACT

Alternatives for licensing and regulating low-level waste burial sites need to be selected on the basis of reasonable evaluations of risk. With regard to low-level sites, biotic transport processes must be evaluated along with other transport processes (e.g. groundwater, agricultural food chains) if the consequences of alternative courses of action are to be understood. Biotic transport (amounts and rates of radionuclide transport by biota) at waste facilities is poorly understood when compared to current understanding of routes and movements of radionuclides through agricultural food chains.

This presentation includes work on Phase I of this research program conducted for the U. S. Nuclear Regulatory Commission (1). Our objective was to assess current biotic transport modeling. Our efforts have focused on evaluating the relevance of biotic transport to the assessment of impacts of waste disposal on humans and their environment. Our assessment considered available dose pathway analysis computer models and the potential applications of biotic transport models at low-level waste (LLW) disposal sites. An identification and review of biotic transport mechanisms and processes for terrestrial systems was provided in Reference (1). Current efforts are to estimate the comparative magnitude of dose to man resulting from biotic transport. The final report for Phase I will contain: a) a critical review of available modeling techniques; b) an assessment of the possible significance of biotic transport of radionuclides from low-level waste disposal sites to man; and c) if appropriate, a summary of additional investigations needed to apply biotic transport modeling to regulatory concerns effectively.

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## INTRODUCTION

Concern for the impacts from human exposure to radioactivity from LLW disposal has resulted in a series of regulatory guidelines, environmental assessments, and management practices. The need to consider and integrate a large number of processes and mechanisms that may contribute to radiation dose has resulted in a number of computer programs to make the required calculations. Additionally, evaluations of radionuclide behavior and movements for extended time periods of up to 500 years have been needed.

In-depth studies of radionuclide movement through the biota associated with commercial low-level waste burial sites have not been conducted, though a number of specific examples of biotic transport have been observed. In addition, laboratory experiments and field observations have demonstrated biotic mechanisms and processes which transport radionuclides (1). The inferences for biotic transport potential are relatively clear. What is not known, and has not been assessed, is the magnitude of the transport over longer times (up to 500 years) and the potential consequences associated with it.

This presentation contains a technical assessment of the methods used to assess the contribution to human radiation dose from biotic transport mechanisms and processes. This presentation provides an evaluation of the relevance of biotic transport in the assessment of the impact of waste disposal on man and his environment. Evidence of the occurrence of biotic transport of radionuclides is provided from low-level waste management sites. Finally, the preliminary results of our "order of magnitude" model comparing the dose impacts of human intrusion and biotic transport 100 years after LLW burial ground closure are briefly discussed.

## LITERATURE REVIEW OF BIOTIC TRANSPORT CONSIDERATIONS

Commercial low-level waste (LLW) burial grounds are located primarily on the basis of regional requirements for radioactive waste disposal. Final siting decisions rely on many considerations, though hydrogeologic and economic concerns are often most important. Existing information regarding biotic transport was reviewed and evaluated to determine the potential importance of biotic transport on siting and operating commercial LLW burial grounds (1). This included a review of biotic pathway considerations in recently published LLW management reports and in radiation exposure pathway models.

## RECENT LLW MANAGEMENT REPORTS

Recent studies on LLW disposal include discussions of impact analysis, waste classification, environmental assessment, and

decommissioning. No studies were found, however, that specifically address biotic transport mechanisms. The studies reviewed utilized environmental models designed to evaluate transport by biota directly related to the human food chain.

Murphy and Holter (2) provided safety and cost information for the conceptual decommissioning of commercial low-level waste burial grounds. They considered both an arid western site and a humid eastern site as the location for reference facilities. Typical hydrology, climate, and geology conditions of arid western and humid eastern locations were used in the conceptual analysis. Direct root uptake of radionuclides by farm products was the only biological release mechanism considered; the authors did not consider biotic transport mechanisms. In fact, Volume 1 of the final report contains the statement:

"ecological pathways involving the movement of waste material by waterfowl, burrowing animals, blowing weeds, etc., are not considered in this study" (2)

No attempt was made to model or determine the potential magnitude of biotic transport as a means of dispersing radioactive material.

As part of the National Environmental Policy Act of 1969 (NEPA), the U. S. Nuclear Regulatory Commission has issued a draft environmental impact statement (EIS) on the proposed new regulation 10CFR61 (3). In this draft EIS, the NRC demonstrated the decision processes applied to the development of new regulations. The impacts associated with low-level waste disposal were estimated for a base case and several alternatives; estimated impacts principally involve long-term radiological exposures and costs. The authors stated that long term exposure pathways could involve human intrusion, potential leaching and groundwater transport, dispersion by animals, long-term erosion of the overburden, air and water transport, and gaseous decomposition. Plant and animal intrusion and dispersion of buried low-level wastes were included in a discussion of other long-term pathways. The draft EIS contains the following statement:

"NRC staff believes, however, that the most significant pathway is ground-water migration. ... Impacts from plant and animal intrusion are site-specific and can be reduced through engineering designs applied to reduce ground-water migration and potential intruder exposures" (3)

The waste classification reported by the NRC (3) was based on dose results from two human intrusion scenarios: the intruder-construction scenario and the intruder-agriculture scenario. The authors made no attempt to quantify or model the potential impacts of biotic transport mechanisms beyond brief statements.

Little et al. (4) reported on an interim version of the PRESTO (Prediction of Radiation Effects from Shallow Trench Operations) computer program. This program was being developed for the U. S.

Environmental Protection Agency to predict health impacts to static population groups over a 1000-year period following burial operations. Data were being developed for three specific sites: Barnwell, South Carolina; Beatty, Nevada; and West Valley, New York. The models in the computer program were intended to assess radionuclide transport, resulting exposures, and health impacts. The basic exposure scenarios considered in the PRESTO models included: routine operations with routine spillage; human intrusion; site reclamation (farming); and atmospheric suspension of radionuclides from an eroded trench. No mention was made about the potential impacts of biotic transport mechanisms during the 1000-year analysis period, and no attempts were made to model or otherwise quantify them.

#### RADIATION EXPOSURE PATHWAY MODELS

Existing radionuclide transport and radiation exposure pathway models were reviewed to determine the current status of biotic transport modeling. Models chosen for review were identified through discussions with personnel at the Pacific Northwest Laboratory who had participated in model development efforts, and through model compilation studies (1). The review resulted in a list of 23 computer programs used for radionuclide transport and dose to man pathway calculations.

The radionuclide transport and radiation exposure pathway computer programs identified considered only atmospheric and surface water transport mechanisms. None of the programs directly considered biotic transport mechanisms. The exposure pathway models calculated concentrations in human food stuffs, with no attempts to model radionuclide transport by non-food chain biota. All of the computer programs were developed to conduct radiation dose analyses following radionuclide releases from operating nuclear facilities or fallout. These programs did not consider potential long-term impacts on the integrity of buried waste containers in LLW trenches or the increase in accessibility of the buried wastes resulting from biotic mechanisms.

#### THE POTENTIAL ROLE OF BIOTIC TRANSPORT MODELS

The long-term behavior of material buried at LLW burial grounds is a topic of prime concern in site-specific environmental assessments. Typically, the radiotoxicity of low-level wastes is evaluated for up to about 500 years. Such a time frame is valid since most radionuclides in low-level wastes have relatively short half-lives. However, some radionuclides disposed of in LLW burial grounds, such as C-14, Ni-59, Tc-99, I-129, transuranic radionuclides, and radionuclides in the uranium decay chain, have half-lives much longer than 500 years. In an attempt to account for the long-term behavior of these radionuclides in the environment adequately, much effort has been devoted to modeling long-term ground-water migrations. However, no corresponding effort has been made to account for the potential long-term impacts of plants and



animals on either the wastes, waste containers, or the trench material overlying the wastes.

Actions of plants or animals on the waste forms or trenches may enhance radionuclide transport by air or water. For example, plant or tree roots may penetrate the overburden and eventually permit direct rainwater contact with the waste zone. Additionally, organisms that accelerate the decomposition of waste disposal containers and forms of waste may be present in the waste trenches, making the waste more easily transported by ground water.

At this time, it is difficult to estimate the impacts of biotic transport mechanisms adequately, since biotic transport mechanisms have not been modeled or quantified. Thus, models are being developed in this study to estimate the quantities and concentrations of materials directly removed from burial sites by the actions of plants and animals. Since most of the plant and animal species of concern here are not directly related to the food chain, additional formulations are being developed to link the new model to the food chain computer programs.

While the magnitude and type of potential biotic transport problems may be highly site-specific, it is inappropriate to continue ignoring the potential long-term impacts of biotic transport. Our modeling efforts are designed to help determine the importance of biotic transport relative to other long-term mechanisms (such as human intrusion of ground-water transport) currently addressed in environmental impact assessments.

## SUMMARY AND PRELIMINARY RESULTS

Licensing decisions regarding low-level waste (LLW) management are, of necessity, influenced by the results of predictive modeling. Our review of recent LLW literature reveals that a long-term scenario analysis approach is typically used in predictive modeling exercises (1). In such an approach, radionuclide transport models are coupled with radiation dose evaluation models through the use of long-term human exposure scenarios. The resulting doses or estimates of human health impacts are then used as input to decisions about siting, trench design, burial ground operation, waste packaging, waste form, and ultimately decommissioning.

All of the modeling exercises reviewed were based on the assumption that burial ground performance was most strongly affected by the long-term actions of air and water. None of the models considered biotic mechanisms and processes other than those that were directly linked to man through the human food chain (1).

Biotic transport has not been evaluated beyond the qualitative level in current radioactive waste disposal impact assessments (1). However, statements about the lack of concern for significant effects

from biotic transport are not supported by operational experience or by an evaluation of biotic transport mechanisms and processes (1). Based on our review, we have established an "order of magnitude" model of biotic transport.

We find that all current dose pathway models are essentially the same and deficient in that they do not include biotic transport. We conclude that the results of radiological impact assessments are highly dependent on the scenario examined (1). Additionally, current assessment procedures may not be ideally suited for use with the long-term scenarios that are of concern.

An example of a typical pathway scenario currently considered is the crop root penetration scenario. This scenario is considered in several current impact assessments (2, 3, 4). In this scenario, surface erosion is assumed to occur over a period of time, typically about 100 to 500 years. At the end of this erosion period, farm crop roots are assumed to penetrate a clean (uncontaminated) trench cover and enter a homogeneously contaminated waste zone. Only a small fraction of the roots are assumed to enter the waste zone. The resulting food crops are then assumed to be ingested by an individual, and radiation doses from ingestion are calculated. The waste profile normally postulated for this scenario is graphically illustrated in Figure 1. Based on the preliminary findings of our "order of magnitude" modeling, biotic transport over a 100 year period has the potential for producing a soil concentration profile similar to the one in Figure 2. This profile could be established by the accumulated intrusion and active transport by plants or animals over this long time period. Although the short-term effects of biotic transport may be small, its accumulated effects may be to enhance the amount of waste that the roots eventually contact. Radionuclides brought up near the surface may lead to other human exposure pathways (e.g. inhalation of resuspended surface contamination). Additionally, the more mobile radionuclides may reconcentrate near the surface as a result of the cumulative action of plants over long time periods. Thus, the mixture of radionuclides near the surface resulting from biotic transport may be significantly different than the mixture in the waste zone.

The results of our preliminary modeling efforts indicate that the critical organ doses from a biotic transport scenario may be of the same order of magnitude as the doses resulting from human intrusion scenarios. Both doses were evaluated at 100 years after site closure. This result clearly indicates that more effort should be devoted to modeling the potential impacts of biotic transport and determining the parameters most critical to the modeling process. A further evaluation is also necessary to determine potential site licensing or design impacts.

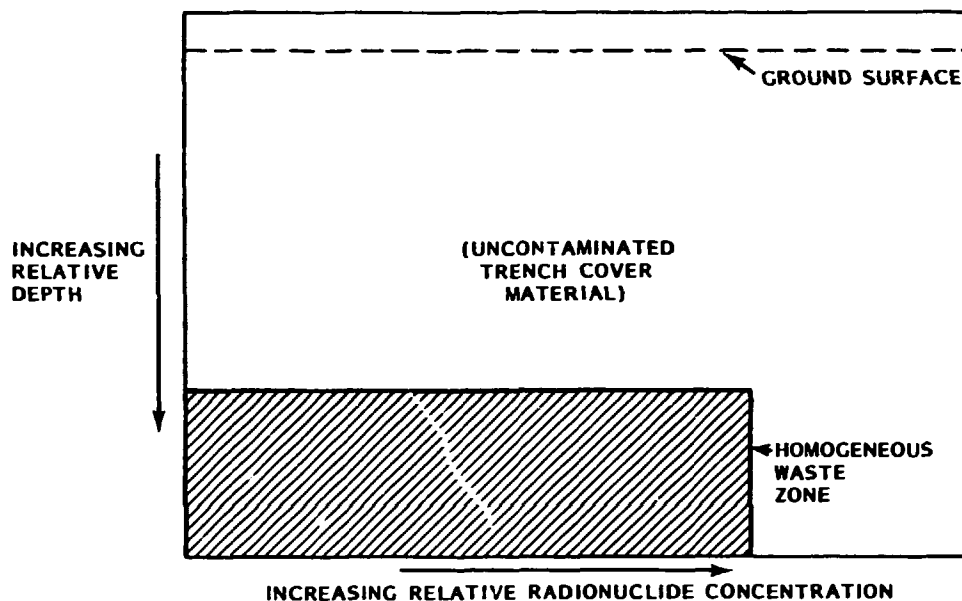


**FIGURE 1.** Waste Concentration Profile for a Typical Crop Root Penetration Scenario

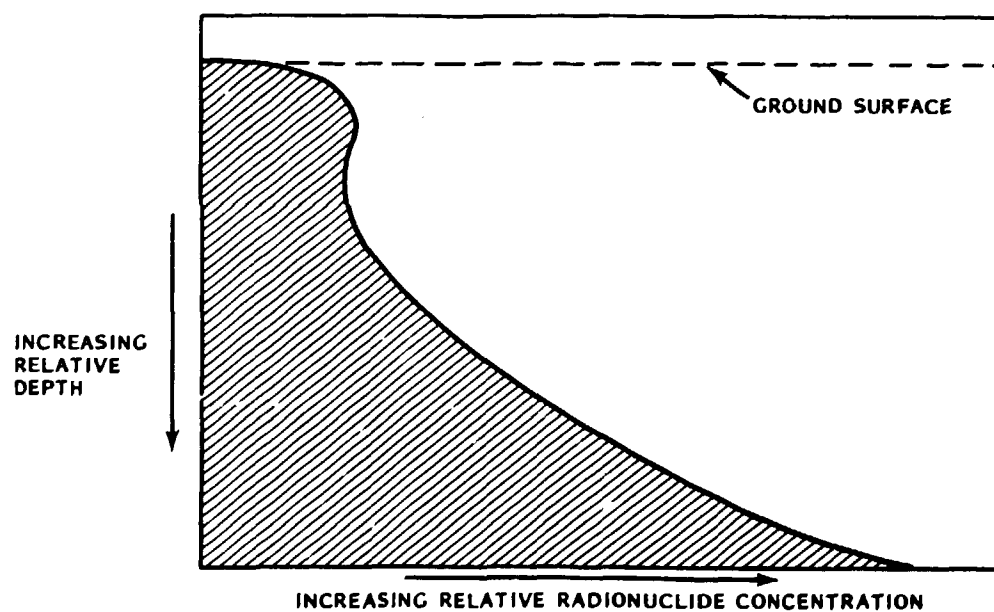


**FIGURE 2.** Potential Waste Concentration Profile Resulting from a Biotic Transport Scenario

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## PREPARATION OF THE CORRECTIVE MEASURES MANUAL

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### INTRODUCTION

The Corrective Measures Manual is part of a series of handbooks designed to help document low-level radioactive waste technology. The goal of the series is to provide the basic information needed for the safe and cost-effective management of low-level radioactive waste. Within that series, the role of the Corrective Measures Manual is to guide the operator in selecting effective corrective actions to correct problems which arise after waste disposal units have been closed. It is anticipated that the handbook will also 1) assist in recognizing undesirable conditions before problems result (or fully develop) and 2) encourage some improvements in operating practices.

Preparation of the handbook has involved compilation and review of a wide range of technical information to develop a cohesive, usable document. Reported on today are 1) the conceptualization of the document using the compiled and reviewed information, 2) the organization of the information, and 3) current development toward completion.

### CONCEPTUALIZATION OF INFORMATION

Radionuclides from disposed waste may be released into the uncontrolled environment through hydrologic, atmospheric, and biologic modes of transport. In Table 1, these transport modes are linked to problems concerning isolation of waste that have been experienced at disposal sites. During our review of experience at the major disposal sites, it became apparent that the most effective corrective actions administered were those developed from an understanding of the disposal site conditions which were linked to the problems. As a result, it was considered particularly important that the handbook assist and support the processes of associating problems with causative conditions and the directing of corrective actions at those conditions. Table 2 shows how three of the problems, water erosion, infiltration of surface water, and dispersion by wind, are connected to causal conditions. Review of these causal conditions should determine the application of appropriate corrective actions in response to the problems.

As an example, two of the conditions shown in Table 2, unstable trench cover and intrusion by deep-rooted plants, have associated potential actions shown in Table 3. This overall conceptualization of the handbook around causal conditions

TABLE 1. RELATIONSHIP OF PROBLEMS IN WASTE ISOLATION TO MODES OF RADIONUCLIDE TRANSPORT

| MODE        | PROBLEM CATEGORY | PROBLEM                                                                                                                 |
|-------------|------------------|-------------------------------------------------------------------------------------------------------------------------|
| Hydrologic  | Surface Water    | -Water erosion<br>-Ponding<br>-Flooding                                                                                 |
|             | Subsurface water | -Infiltration of surface water<br>-Intrusion of ground water<br>-Solute and particulate mobilization<br>-Gas generation |
| Atmospheric | ---              | -Wind erosion<br>-Dispersion by wind                                                                                    |
| Biologic    | Plant            | -Plant uptake                                                                                                           |
|             | Animal           | -Dispersion by animals                                                                                                  |

TABLE 2. EXAMPLES OF RELATIONSHIP OF PROBLEMS TO CAUSAL CONDITIONS

| PROBLEM                       | CONDITION                                                                                                 |
|-------------------------------|-----------------------------------------------------------------------------------------------------------|
| Water erosion                 | -Unstable trench cover<br>-Subsidence                                                                     |
| Infiltration of surface water | -Permeable trench cover                                                                                   |
| Dispersion by wind            | -Unstable trench cover<br>-Intrusion by deep-rooted plants<br>-Chemical and physical conditions in trench |

TABLE 3. EXAMPLES OF MEASURES TO CORRECT CONDITIONS

| CONDITION                       | CORRECTIVE MEASURES                                                                                                                                                                                 |
|---------------------------------|-----------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------|
| Unstable trench cover           | <ul style="list-style-type: none"> <li>-Increase control over surface water</li> <li>-Fortify surface</li> </ul>                                                                                    |
| Intrusion by deep-rooted plants | <ul style="list-style-type: none"> <li>-Remove and dispose</li> <li>-Encourage desired vegetation</li> <li>-Limit maturation of problem vegetation</li> <li>-Discourage all plant growth</li> </ul> |



- Directs the user to the cause of the problems.
- Promotes a systematic approach to applying corrective actions.
- Tunes the user to unanticipated problems or to those which are not yet apparent.

#### ORGANIZATION OF INFORMATION

The organization of technical information for handbook users must be responsive to the following questions:

- 1) What conditions cause problems?
- 2) What problems result?
- 3) How are they apparent?
- 4) How are the conditions corrected?

Based on review of experience at the major disposal sites, the following conditions have resulted in problems:

- Unstable trench cover
- Permeable trench cover
- Subsidence
- Intersection with Ground Water System
- Intrusion by Deep-Rooted Plants
- Intrusion by Burrowing Animals
- Chemical and Physical Conditions in trench

These conditions form the backbone for organization of the information concerning corrective measures.

#### CURRENT DEVELOPMENT

For the user of the handbook, an important part of preparing a response to a problem is review of the previous experience of others and the ideas available from related research and development. These provide approaches to correcting problems, design considerations, and in some cases, evidence of effectiveness.

A compilation of information on experience and research and development in applying corrective measures at the major low-level radioactive waste disposal sites in the United States (Rose, et al., 1982) has been drafted and sent out for the peer review process. It was based on a review of relevant literature and on visits to eleven disposal sites. Included were facilities operated by the Department of Energy,

- Hanford Reservation
- Idaho National Engineering Laboratory
- Los Alamos National Laboratory
- Nevada Test Site
- Oak Ridge National Laboratory
- Savannah River Plant

and those operated for commercial disposal,

- Barnwell Chem-Nuclear Systems, Inc.
- Beatty, U.S. Ecology, Inc.
- Maxey Flats, National Waste Management Services, Inc.
- Richland, U.S. Ecology, Inc.
- West Valley, Nuclear Fuel Services, Inc.

Reference was also made to studies conducted at the Sheffield, Illinois site and at Brookhaven National Laboratory and Sandia National Laboratories.

The information was organized into the seven conditions discussed earlier, for compatibility with the overall structure of the handbook. In discussion of each, the condition was briefly introduced and related to disposal site problems. It was followed by an itemization of actions taken to correct the condition, both those that have actually been applied and those being studied and developed.

The disposal sites examined are located throughout the country, so the information compiled spans a range of climatologic, geologic, hydrologic, and biologic characteristics. Consequently, the severity of problems associated with the conditions varies significantly among the sites. The result is variation in the types of information concerning corrective measures. Some conditions have received a large amount of attention, such as those associated with water-related problems at some sites. Corrective measures for other conditions, such as subsidence and

intrusion by burrowing animals, have not been as fully developed, but research and development activities are underway. Still other conditions are currently being evaluated with respect to the nature and extent of associated problems. Examples are the conditions associated with gas production and permeation.

The peer review process for this document will be concluded before the completion of the draft of the Corrective Measures Manual. Consequently, it is anticipated that feedback from this review will assist in preparing a handbook containing effective guidance for the user.

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## SOURCE TERMS OF RADIONUCLIDES IN SHALLOW LAND BURIAL SITES\*

*Day*

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## ABSTRACT

A research program is under way to provide source term data on the chemical and radionuclide contents of trench waters at low level radioactive waste disposal sites and to provide information on the geochemical behavior of radionuclides under simulated field conditions. Trench waters were sampled at four commercial waste burial sites and analyzed for chemical and radionuclide constituents. The sorption, diffusion and precipitation behavior of radionuclides were studied using site specific sediments and trench waters. Radionuclide leach rates from solidified waste, under simulated trench conditions, were also determined.

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INTRODUCTION

A field and laboratory program was designed to study the existing commercial low level radioactive waste disposal sites. The prime consideration for continued use of shallow land burial practices for the disposal of low level radioactive waste is the containment of radionuclides from the environment. This investigation provides source term data for radionuclides and other solutes in trench waters at the sites and will describe physical and chemical properties of the geochemical system that control the movement of radionuclides. These objectives were pursued in four task areas: (1) Trench water chemistry; (2) Characteristics of sediments from representative lithostratigraphic units at disposal sites; (3) Radionuclide migration studies; (4) Waste form leaching under simulated trench environments.

A summary of the past and present activities of the Source Term of Radionuclides project is shown in Table 1 and described in earlier reports. (1-6)

\*Work performed under the auspices of the United States Nuclear Regulatory Commission.

Table 1

## Past and Present Activities of the Source Term of Radionuclides Project

| Disposal Sites | Trench Water Chemistry | Sediment Characterization | Radio-nuclide Migration   | Leaching in Simulated Trench Environment |
|----------------|------------------------|---------------------------|---------------------------|------------------------------------------|
| Maxey Flats    | 1976-1979, 1981        | IP                        | Sorption<br>Precipitation |                                          |
| West Valley    | 1977, 1978,            | IP                        | Sorption<br>Precipitation |                                          |
| Barnwell       | 1977-1979, 1980        | IP                        | Sorption<br>Diffusion     | IP                                       |
| Sheffield      | 1979, 1982             | IP                        | Sorption<br>Diffusion     |                                          |

IP = in progress.

## TRENCH WATER CHEMISTRY

Since 1976, water samples have been obtained periodically from trenches and wells at the commercial waste disposal sites located in Maxey Flats, KY; West Valley, NY; Barnwell, SC; and Sheffield, IL. Sampling procedures were developed to maintain inert conditions while measuring pH, Eh, conductance, dissolved oxygen and temperature in the field. Figure 1 shows the equipment used to sample trench water at the disposal sites. In the laboratory, the trench waters were analyzed for dissolved radionuclides ( $^{60}\text{Co}$ ,  $^{137}\text{Cs}$ ,  $^3\text{H}$ ,  $^{90}\text{Sr}$ , and Pu isotopes) and major inorganic constituents ( $\text{Na}^+$ ,  $\text{K}^+$ ,  $\text{NH}_4^+$ ,  $\text{Ca}^{+2}$ ,  $\text{Mg}^{+2}$ ,  $\text{Ba}^{+2}$ , total Fe,  $\text{Cl}^-$ ,  $\text{SO}_4^{-2}$ ,  $\text{HCO}_3^-$ ). In addition, selected trench water samples are being analyzed for chelating agents and carrier cations such as  $\text{Sr}^{+2}$ ,  $\text{Cs}^+$ , and  $\text{Co}^{+2}$ .

Tritium is the dominant radionuclide in the trench waters at all the sites. The concentrations of  $^{90}\text{Sr}$ ,  $^{238}\text{Pu}$ ,  $^{239}\text{Pu}$ , and  $^{240}\text{Pu}$  at Maxey Flats and West Valley exceed those at Barnwell and Sheffield by several orders of magnitude (Table 2).

As an example, ternary plots displaying the major cation and anion compositions of Maxey Flats trench waters are shown in Figures 2 and 3.



Figure 1. Portable anoxic collection system used to sample trench water from low level radioactive waste disposal sites.

Table 2

Maximum Observed Radionuclide Concentrations  
in Trench Waters From Low Level Waste Disposal Sites

|                       | Concentration<br>(pCi/L) |                   |                      |                      |
|-----------------------|--------------------------|-------------------|----------------------|----------------------|
|                       | Maxey<br>Flats           | West<br>Valley    | Barnwell             | Sheffield            |
| Gross Alpha           | $5.9 \times 10^5$        | $1.3 \times 10^5$ | $1.6 \times 10^1$    | $8.6 \times 10^2$    |
| Gross Beta            | $5.3 \times 10^6$        | $3.8 \times 10^7$ | $9.6 \times 10^2$    | $5.6 \times 10^4$    |
| Tritium               | $5.1 \times 10^9$        | $3.7 \times 10^9$ | $7.8 \times 10^8$    | $6.2 \times 10^5$    |
| $^{238}\text{Pu}$     | $2.1 \times 10^5$        | $1.6 \times 10^5$ | $4.6 \times 10^0$    | $1.4 \times 10^0$    |
| $^{239/240}\text{Pu}$ | $2.1 \times 10^4$        | $3.4 \times 10^2$ | $9.3 \times 10^{-1}$ | $9.3 \times 10^{-1}$ |
| $^{90}\text{Sr}$      | $5.4 \times 10^5$        | $2.4 \times 10^7$ | $4.8 \times 10^1$    | $2.2 \times 10^3$    |
| $^{134}\text{Cs}$     | $1.7 \times 10^3$        | $5.0 \times 10^3$ | $3.4 \times 10^1$    | $1.4 \times 10^4$    |
| $^{137}\text{Cs}$     | $2.3 \times 10^4$        | $1.0 \times 10^6$ | $5.1 \times 10^2$    | $3.0 \times 10^4$    |
| $^{60}\text{Co}$      | $5.0 \times 10^4$        | $1.8 \times 10^3$ | $3.7 \times 10^2$    | $2.4 \times 10^4$    |
| $^{241}\text{Am}$     | $4.5 \times 10^4$        | $3.6 \times 10^2$ | ---                  | ---                  |

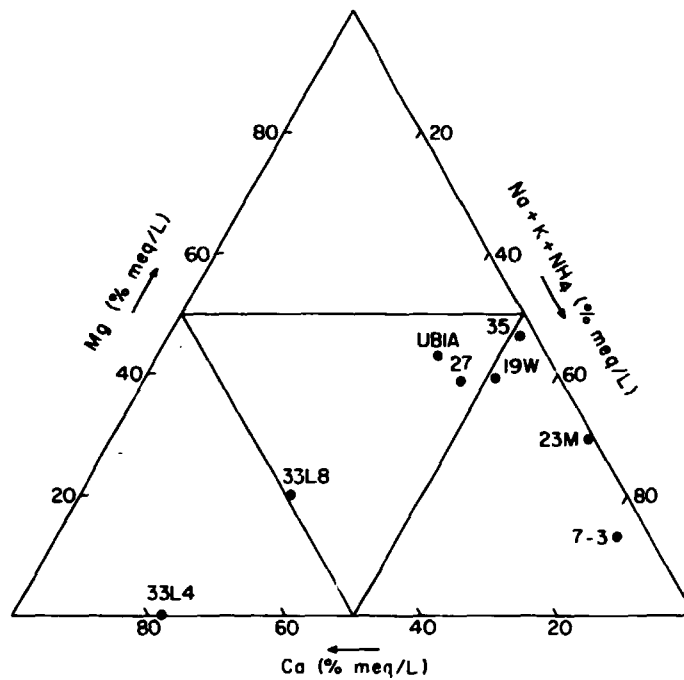


Figure 2. Trilinear diagram displaying major cation composition of Maxey Flats trench waters. Data for well water UB1A are also shown.

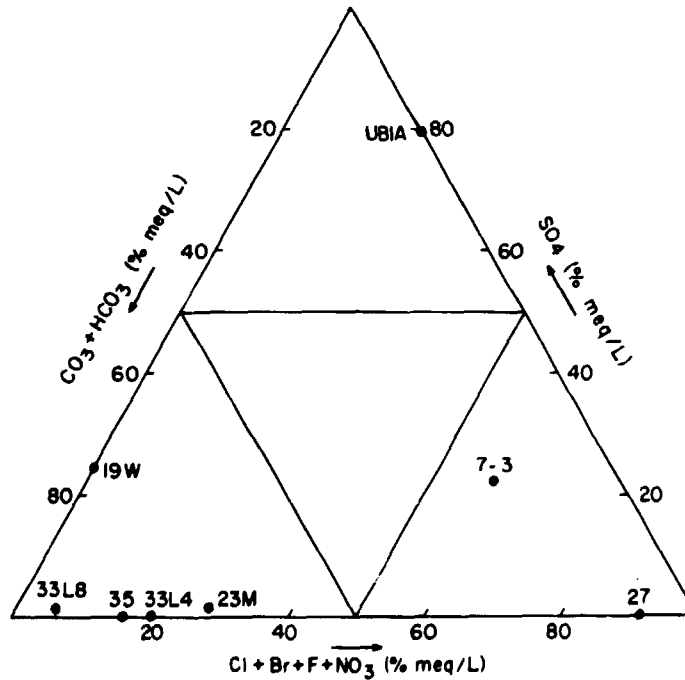


Figure 3. Trilinear diagram displaying major anion composition of Maxey Flats trench waters. Data for well water UB1A are also shown.

## SEDIMENT CHARACTERISTICS

Basic information on the physical and chemical characteristics of site specific sediments is essential to the evaluation of their radionuclide retention properties. The characterization included particle size distribution, cation exchange capacity, surface area, organic matter, extractable iron and total carbonate. In addition, parameters relevant to radionuclide migration studies, such as bulk density, porosity and hydraulic conductivity, were determined on repacked sediment cores from the Hawthorne formation at Barnwell.

## RADIONUCLIDE MIGRATION TESTS

The principal processes which control radionuclide migration are transport processes and retardation. Transport processes include advection, mechanical dispersion and diffusion. Some reactions such as complex formation with inorganic ligands and with chelating agents can enhance radionuclide mobility. The processes which retard radionuclide migration consist of a large group of reactions that occur between dissolved radionuclides and the sediments and are collectively known as sorption processes. Sorption is the primary process by which radionuclide transport is retarded. Precipitation-coprecipitation reactions and radionuclide scavenging by hydrous oxide/hydroxide precipitates can also retard radionuclide migration.

In this program, we have considered diffusion and sorption as the principal radionuclide migration processes. For this purpose, quantitative measurements of radionuclide diffusion coefficients ( $D$ ) and sorption coefficients ( $K_d$ ) using site specific sediments and trench waters are being made. In addition, radionuclide retardation factors and the extent of radionuclide scavenging by iron oxide/hydroxide precipitation upon oxidation of anoxic trench waters are being determined.

## BATCH SORPTION

Mathematical models used to calculate radionuclide transport by groundwater generally assume ideal chromatographic behavior for radionuclide sorption. The distribution coefficient,  $K_d$ , must be experimentally determined to account for the partitioning of each radionuclide between the solution and the solid phases. Sorption isotherms for selected radionuclides were determined using site specific sediments and radionuclide spiked trench waters.



## DIFFUSION

The diffusion behavior of selected radionuclides is being studied by direct measurement of radionuclide penetration into a sediment plug. The diffusion of tritium through clay rich sediments is of particular interest to low level disposal sites because of the abundance of tritium in the trench waters. In addition, experiments are in progress to determine apparent diffusion coefficients for reactive tracers such as  $^{137}\text{Cs}$ ,  $^{60}\text{Co}$ ,  $^{85}\text{Sr}$ , etc. The diffusion measurements for reactive tracers represent both diffusion and sorption. Tritium migration, on the other hand, reflects sorption-free diffusion.

## RETARDATION

Experiments were initiated to measure radionuclide retardation using site specific sediments and simulated trench water. The retardation of a radionuclide is defined conceptually as the ratio of the velocity ( $V$ ) at which the radionuclide moves through the porous medium divided by the linear velocity.

$$R_f = V/V_1$$

The linear groundwater velocity,  $V_1$ , is equal to the velocity at which a nonsorbed tracer (tritiated water) passes through a porous sediment. A schematic for the apparatus for radionuclide retardation testing is shown in Figure 4.

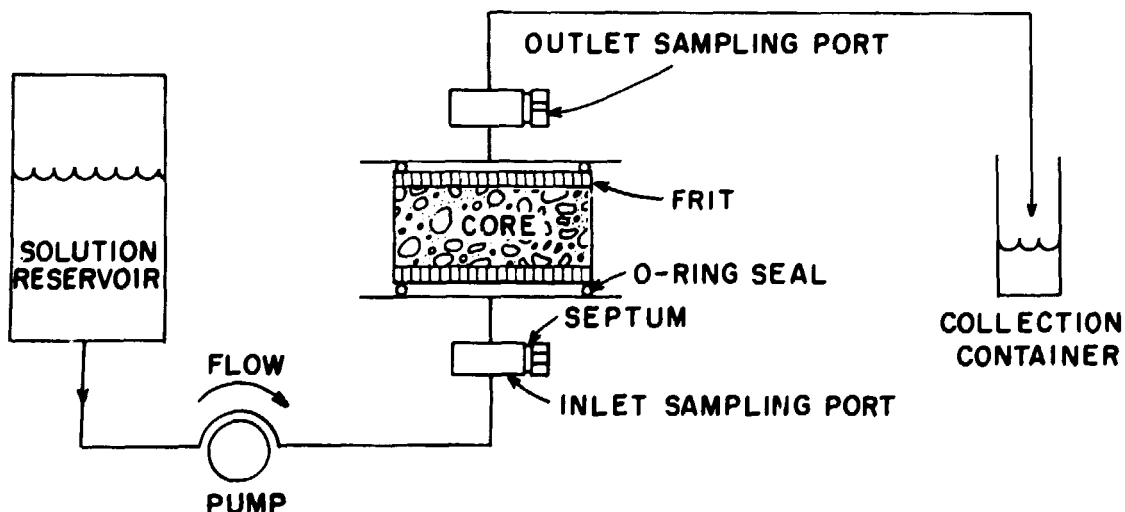


Figure 4 A schematic of the apparatus for radionuclide retardation testing by continuous water flow through sediment core.

## PRECIPITATION-COPRECIPITATION

Precipitation-coprecipitation reactions and radionuclide scavenging by precipitates can result in significant retardation of radionuclide migration in both unsaturated and saturated zones. Tests to measure the extent of coprecipitation of radionuclides by iron oxyhydroxide as a result of the oxidation of anoxic trench waters are under way. Scoping experiments to study the scavenging of radionuclides by metal sulfide and carbonate precipitates in anoxic trench waters have been initiated.

## LEACHING OF WASTE FORM IN SIMULATED TRENCH ENVIRONMENTS

Data on the leaching behavior of ion exchange resins and boric acid waste streams solidified in cement have been generated using a modified IAEA leach test at BNL.<sup>(7)</sup> Such tests, although useful in making relative comparisons of waste forms, represent a worst case leaching condition in a water saturated environment. In a realistic situation, the trench environments are only partially saturated and the water moves away from the waste at a rate controlled by the groundwater infiltration rate and the permeability of the sediment. Experiments are being conducted to leach solidified waste surrounded by site-specific soil in a flowing, partially water saturated environment (Figure 5). The waste form consists of Na<sup>+</sup> form resin beads, loaded with <sup>137</sup>Cs and <sup>85</sup>Sr, and solidified in Portland I cement. Simulated trench water is used as a leachant. For sake of comparison, parallel experiments are being performed using high density polyethylene beads as an inert, porous medium to determine leachability of the waste forms with minimal sorption of the radionuclides onto the surrounding medium.

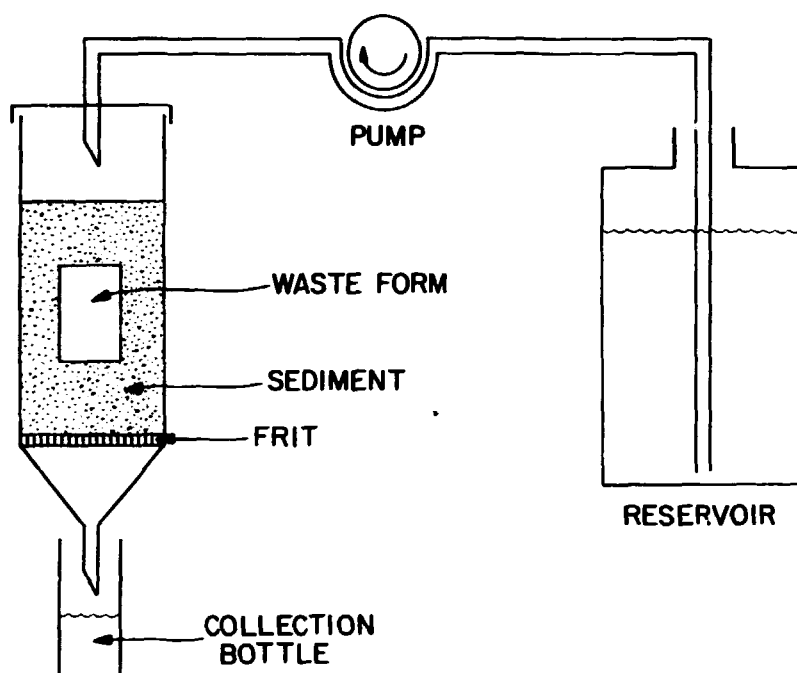


Figure 5 A schematic of the apparatus to measure leaching of solidified waste in a porous solid medium with continuous unsaturated leachant solution flow.

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## MICROBIAL EFFECTS ON RADIOACTIVE WASTES AT SLB SITES

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### INTRODUCTION

A significant fraction of DOE and commercially generated low-level radioactive waste consists of carbonaceous compounds including paper, wood, cloth, animal carcasses, scintillation mixtures and chelating agents from decontamination solutions. These materials are subject to degradation by microorganisms present in the shallow land burial environment and may contribute to enhanced migration of radionuclides through the formation of gases, mobile complexes and bioaccumulation. Microorganisms may directly attack the waste package promoting container corrosion and waste form degradation. Problems relevant to subsidence and trench deterioration have also been attributed to microbial degradation of organic wastes.

While several mechanisms for enhanced radionuclide migration as a result of microbial degradation have been proposed, there is no consensus concerning which mechanisms predominate or whether they are even significant.

The objectives of this study are to determine the significance of microbial degradation of organic wastes on radionuclide migration in shallow land burial for humid and arid sites, establish which mechanisms predominate and ascertain the conditions under which these mechanisms operate.

A literature study was conducted in FY 82 to determine the present state of knowledge dealing with radionuclide movement attributed to microbial activities at shallow land burial sites.

The results of this study are given in detail in reports entitled Assessment of Microbial Processes on Gas Production at Radioactive Low-Level Waste Disposal Sites, BNL-51557, May 1982 and Assessment of Microbial Processes on Radionuclide Mobility in Shallow Land Burial, BNL-51574, July 1982.

This work was identified as a major data need in a report entitled, Unresolved Technical Issues in Land Burial of Low-Level Radioactive Wastes, ORNL/NFW-79/62 (October 1979). The program is in direct support of DOE/LLW Management Program Alpha Milestone B, "Develop technology for waste treatment, handling, and packaging for shallow land burial site disposal," Milestone C, "Develop technology and documentation to support a shallow land burial site," and Milestone D, "Develop remedial action technology for shallow land burial sites."

### MICROBIAL GENERATION OF GASES

Factors controlling gaseous emanations from low-level radioactive waste disposal sites are assessed. Importance of gaseous fluxes of methane, carbon dioxide and possibly hydrogen from the site stems from the inclusion of tritium and/or  $^{14}\text{C}$  into the elemental composition of these compounds. In that the primary source of these gases is the biodegradation of organic components of the waste materials, primary emphasis of the study involved an examination of the biochemical pathways producing methane, carbon dioxide and hydrogen, and the environmental parameters controlling the activity of the microbial community involved. Initial examination of the data indicates that the ecosystem is anaerobic. Hence, reaction sequences of importance are fermentation of carbonaceous substrates to fatty acids, alcohols, ammonia, carbon dioxide and hydrogen. The fatty acids and alcohols are further metabolized to acetate plus carbon dioxide. With a sufficiently reducing environment, acetate, carbon dioxide and hydrogen are converted to methane by the methanogenic bacteria. As the result of the complexity of the pathway leading to methane production, factors such as substrate availability, which limit the initial reaction in the sequence, greatly affect the overall rate of methane evolution.

Although the methane and carbon dioxide production rate indicates the degradation rate of the organic substances in the waste, it does not predict the methane evolution rate from the trench site. Methane fluxes from the soil surface are equivalent to the net synthesis minus the quantity oxidized by the microbial community as the gas passes through the soil profile. Methane metabolized within the soil profile will be converted to carbon dioxide, water and cell components. The  $^{14}\text{C}$  in the methane converted to carbon dioxide would leave the ecosystem with carbon dioxide produced within the confines of the trench. The  $^{14}\text{C}$  entering cell components would remain within the soil for a period of time dependent upon the turnover rate of the cell carbon.

### FIELD STUDIES OF GASEOUS RELEASES

Gas studies were performed at three of the six commercial low-level radioactive waste disposal sites during the period 1976-1978. The first was an exploratory study performed at West Valley, New York, in June 1976 by personnel from USGS, NYGS, and the N.Y. State Health Department (1-3). Additional gas samples were collected at West Valley during periods in 1977 and 1978. In August 1978, a limited study was conducted at the Beatty, Nevada, disposal site (4,5) and in December, 1978, a similar study was made at Maxey Flats, Kentucky (6). The results of these studies are summarized in Tables 1-3.

Table 1  
Radionuclide Concentrations in Field Air and Soil Gas Samples  
Above Trench Soil Covers at West Valley, New York (3)  
( $\mu\text{Ci}/\text{cm}^3 \pm$  error in parentheses)

| Over-Trench                    | Date    | $^3\text{HH}$ | $^{14}\text{CH}_4$                                           | $^3\text{HCH}_3$ | $^{14}\text{C}(\text{HC})^a$ | $^3\text{H}(\text{HC})^a$ | $^{14}\text{CO}_2$ | $^3\text{HHO}$ |
|--------------------------------|---------|---------------|--------------------------------------------------------------|------------------|------------------------------|---------------------------|--------------------|----------------|
| 14 (1 in.)                     | 6/15/77 | -             | <8 E-8                                                       | <2 E-7           | -                            | -                         | <1.6 E-7           | -              |
| 11 (1 in.)                     | 6/15/77 | -             | <1.1 E-7                                                     | <1.4 E-7         | -                            | -                         | <1.3 E-7           | -              |
| 9 (1 in.)                      | 9/21/77 | -             | <3 E-8                                                       | <1.0 E-7         | -                            | -                         | -                  | -              |
| 5 (1 in.)                      | 9/21/77 | -             | <5 E-8                                                       | <1.7 E-7         | -                            | -                         | -                  | -              |
| 9 (1 in.)                      | 4/26/78 | <6 E-8        | <1.4 E-8                                                     | 7 E-8(83)        | -                            | -                         | <1.3 E-7           | -              |
| 5 (5 ft.)                      | 4/26/78 | -             | <1.3 E-12                                                    | 8 E-13(51)       | -                            | -                         | <4 E-11            | 2.5 E-12(17)   |
| 14 (5 ft.)                     | 4/26/78 | -             | <9 E-13                                                      | 4 E-12(34)       | -                            | -                         | <1.3 E-11          | < 4E-13        |
| 3 (Over-fracture)              | 6/04/76 | <1.3 E-9      | $\underbrace{7 \text{ E-7(11)} \quad 1.1 \text{ E-7(10)}}_b$ |                  | -                            | -                         | 6.7 E-9(12)        | -              |
| 5 (F) Flux <sup>c</sup><br>Box | 9/21/77 | 4.1 E-9(40)   | 3.3 E-7(5)                                                   | 9.0 E-6(5)       | 1.1 E-8(21)                  | 1.1 E-6(5)                | 4.1 E-7(7)         | <1.6 E-9       |
| 5(UF) Flux <sup>d</sup><br>Box | 9/21/77 | <3 E-10       | 9.1 E-9(5)                                                   | 2.3 E-6(5)       | <2 E-9                       | 4.8 E-7(5)                | 9.3 E-7(5)         | 1.2 E-8        |

<sup>a</sup>As hydrocarbons above methane ( $\text{C}_2\text{H}_6$ ,  $\text{C}_3\text{H}_8$ , and  $\text{C}_4\text{H}_{10}$ )

<sup>b</sup>Total methane in gas sample < 0.02%

<sup>c</sup>Fractured

<sup>d</sup>Unfractured



Table 2  
Radionuclide Concentrations and Gas Compositions  
of Trench and Soil Gases at Beatty, Nevada [4]

| Location             | N <sub>2</sub><br>(% Vol.) | O <sub>2</sub><br>(% Vol.) | CO <sub>2</sub><br>(ppm) | CH <sub>4</sub><br>(ppm) | Trace H <sub>2</sub> O<br>(mg/L) | Gaseous<br>Tritium<br>( $\mu$ Ci/cm <sup>3</sup> ) | <sup>3</sup> H <sub>2</sub> O<br>( $\mu$ Ci/cm <sup>3</sup> ) | Gaseous<br>C-14<br>( $\mu$ Ci/cm <sup>3</sup> ) |
|----------------------|----------------------------|----------------------------|--------------------------|--------------------------|----------------------------------|----------------------------------------------------|---------------------------------------------------------------|-------------------------------------------------|
| <u>Background</u>    | 78                         | 22                         | 300                      | 10                       | 1.95                             | <1.4 E-10                                          | 5.52 E-8                                                      | <9.01 E-11                                      |
| <u>Trench Gas</u>    |                            |                            |                          |                          |                                  |                                                    |                                                               |                                                 |
| Observation Well R-1 | 78                         | 20                         | 1860                     | 7.7                      | 2.81                             | 2.68 E-7                                           | 8.22 E-8                                                      | 5.90 E-8                                        |
| Observation Well R-2 | 78                         | 17                         | 8100                     | 19                       | 2.74                             | 5.68 E-9                                           | 3.08 E-8                                                      | 3.60 E-8                                        |
| Observation Well R-3 | 74                         | 15                         | 1270                     | 41                       | 2.55                             | 4.73 E-8                                           | 2.93 E-8                                                      | 3.04 E-8                                        |
| Observation Well R-6 | 78                         | 22                         | 600                      | 7                        | --                               | 2.11 E-7                                           | --                                                            | 1.46 E-9                                        |
| Sump Well Tr-14      | 76                         | 21                         | 530                      | 16                       | 2.88                             | 9.86 E-8                                           | 5.52 E-6                                                      | 2.59 E-8                                        |
| Sump Well Tr-16      | 64                         | 17                         | 2200                     | 9.8                      | 3.59                             | 9.82 E-9                                           | 9.35 E-8                                                      | 8.56 E-9                                        |
| <u>Soil Gas</u>      |                            |                            |                          |                          |                                  |                                                    |                                                               |                                                 |
| Canister Tr-2        | 73                         | 21                         | 300                      | 5.1                      | 1.77                             | <1.4 E-10                                          | 5.18 E-8                                                      | <9.0 E-11                                       |
| Canister Tr-3        | 77                         | 22                         | 300                      | 7.1                      | 2.11                             | 1.99 E-7                                           | 1.90 E-8                                                      | 2.21 E-10                                       |
| Canister Tr-4        | 76                         | 21                         | 300                      | 6.5                      | 2.08                             | 1.17 E-7                                           | 7.55 E-8                                                      | 1.35 E-10                                       |
| Canister Tr-5        | 73                         | 21                         | 300                      | 6.5                      | 2.02                             | 7.75 E-8                                           | 1.27 E-8                                                      | 5.27 E-9                                        |
| Canister Tr-14       | 79                         | 21                         | 300                      | 6.5                      | 2.02                             | 1.53 E-7                                           | 3.72 E-8                                                      | 5.41 E-10                                       |
| Canister Tr-16       | 78                         | 22                         | 300                      | 7.3                      | 2.58                             | 1.98 E-7                                           | 1.62 E-8                                                      | 8.11 E-10                                       |

Table 3  
Radionuclide Concentrations and Gas Compositions  
of Trench and Soil Gases at Maxey Flats, Kentucky [6]

| Location          | N <sub>2</sub><br>(% Vol.) | O <sub>2</sub><br>(% Vol.) | CO <sub>2</sub><br>(ppm) | CH <sub>4</sub><br>(ppm) | Trace H <sub>2</sub> O<br>(mg/L) | Gaseous<br>Tritium<br>( $\mu\text{Ci}/\text{cm}^3$ ) | <sup>3</sup> HHO<br>( $\mu\text{Ci}/\text{cm}^3$ ) | Gaseous<br>C-14<br>( $\mu\text{Ci}/\text{cm}^3$ ) |
|-------------------|----------------------------|----------------------------|--------------------------|--------------------------|----------------------------------|------------------------------------------------------|----------------------------------------------------|---------------------------------------------------|
| <u>Background</u> |                            |                            |                          |                          |                                  |                                                      |                                                    |                                                   |
| North Fence       | 77                         | 22                         | 98                       | 2                        | 0.95                             | 4.18 E-8                                             | 2.48 E-9                                           | N.D. <sup>a</sup>                                 |
| South Fence       | 77                         | 22                         | 168                      | 2                        | 2.11                             | 1.28 E-8                                             | 8.33 E-9                                           | N.D.                                              |
| <u>Trench Gas</u> |                            |                            |                          |                          |                                  |                                                      |                                                    |                                                   |
| Tr-7 Sump         | 77                         | 22                         | 148                      | 2                        | 1.48                             | 1.29 E-8                                             | 6.31 E-9                                           | N.D.                                              |
| Tr-11S Sump       | 77                         | 22                         | 470                      | 114                      | 2.60                             | 1.58 E-8                                             | 1.48 E-8                                           | 4.95 E-11                                         |
| Tr-18 Sump        | 78                         | 20                         | 12000                    | 6720                     | 2.56                             | 1.12 E-7                                             | 2.25 E-8                                           | 7.88 E-8                                          |
| Tr-26 Sump        | 77                         | 22                         | 29                       | 2                        | 2.22                             | 1.58 E-7                                             | 7.77 E-9                                           | 2.57 E-10                                         |
| Tr-35 Sump        | 77                         | 22                         | 186                      | 8                        | 1.38                             | 1.61 E-8                                             | 2.25 E-9                                           | N.D.                                              |
| <u>Soil Gas</u>   |                            |                            |                          |                          |                                  |                                                      |                                                    |                                                   |
| Tr-11S            | 77                         | 22                         | 255                      | 1                        | 1.04                             | 2.84 E-9                                             | 8.67 E-10                                          | N.D.                                              |
| Tr-18             | 77                         | 21                         | 2580                     | 12                       | 1.33                             | 9.01 E-10                                            | 6.42 E-10                                          | 4.50 E-11                                         |
| Tr-20             | 78                         | 22                         | 881                      | 1.4                      | 1.47                             | 1.13 E-8                                             | 2.59 E-9                                           | N.D.                                              |
| Tr-26             | 77                         | 22                         | 819                      | 2                        | 2.12                             | 5.81 E-9                                             | 4.95 E-10                                          | N.D.                                              |
| Tr-35             | 77                         | 22                         | 1410                     | 4                        | 2.83                             | 9.50 E-9                                             | 1.80 E-9                                           | N.D.                                              |

<sup>a</sup> N.D. - Not detected

The concentrations in methane and carbon dioxide are orders of magnitude higher at West Valley, while the levels at Maxey Flats are comparable to Beatty. It would be expected that microbial activity at the humid West Valley site would produce more methane and carbon dioxide than at the arid Beatty site, but the observation that the levels of carbon-14 in soil gas are approximately the same at Beatty and Maxey Flats (humid site) is noteworthy. The observation that gaseous tritium and tritiated water vapor in soil gas samples at Beatty are higher than at Maxey Flats may be caused by (1) the higher evapotranspiration at Beatty, (2) the higher permeability of the trench covers at Beatty, or (3) the differences in atmospheric pressure at the time the samples were obtained.

The concentrations of tritium and carbon-14 measured in soil gas compared to maximum permissible concentrations (MPC) for release to unrestricted areas listed in Table II of 10 CFR Part 20 are a factor of 200 below MPC for tritium and a factor of 190 below MPC for carbon-14 (7). These calculations were based on the maximum tritium and carbon-14 concentration observed at the Beatty facility, viz.,  $2 \times 10^{-7}$   $\mu\text{Ci}/\text{cm}^3$  gaseous tritium and  $5.27 \times 10^{-6}$   $\mu\text{Ci}/\text{cm}^3$  carbon-14. MPC values of  $4 \times 10^{-5}$   $\mu\text{Ci}/\text{cm}^3$  and  $1 \times 10^{-6}$   $\mu\text{Ci}/\text{cm}^3$  for submersion doses were used for tritium and carbon-14, respectively. The same comparisons for the highest tritium and carbon-14 soil gas emanations measured at Maxey Flats ( $1.1 \times 10^{-8}$   $\mu\text{Ci}/\text{cm}^3$  [Tritium] and  $9 \times 10^{-7}$   $\mu\text{Ci}/\text{cm}^3$  [carbon-14]) indicate less than MPC releases for each radionuclide. The highest emanations measured at West Valley, in the flux boxes above Trench 5, were a factor of 4.4 below MPC for tritium and a factor of 1.1 below MPC for carbon-14. The actual concentrations in unrestricted areas would be much less due to atmospheric dispersion. Emanation rates at the facility boundary are expected to be considerably less than 25 mrem/yr whole body criteria proposed (7).

Additional field studies should be performed to verify these preliminary investigations, taking into consideration the effects of differences in site climate, soil conditions, and measurement techniques.

#### BIOACCUMULATION OF RADIONUCLIDES

The questions which must be considered to determine the impact of bioaccumulation on radionuclide mobility are: a) Does bioaccumulation occur in the trench ecosystem? b) Would bioaccumulation enhance or retard radionuclide mobility? c) What would the effect of radionuclide decay be on microbial growth and development? d) Upon death of the microbial cell would mineralization of the cell constituents and subsequent mobility of the radionuclides be greater or less than if bioaccumulation had not occurred? Due to the lack of data concerning occurrence of this phenomenon in the disposal trench, conclusions must be based on extrapolation of experimental results collected during study of alternate ecosystems (8-10).

The major physical manifestation of bioaccumulation is that soluble radionuclide molecules are converted to particulate substances of sizes approximating those of growing microbial cells. Thus, factors controlling mobility of the nuclide will essentially be those relating to microbial mobility in general. Within the trench confines, the microbes may be suspended in the trench waters or immobilized on surfaces of particulate substances. The latter microbes are of little consequence in radionuclide loss from the trench. The question, thus, becomes: Will the bacteria suspended in trench waters be able to penetrate the soil matrix and migrate from the trench? Considerable research has been conducted to evaluate movement of fecal coliforms contained in sewage effluent applied to soil surfaces or entering soils through septic systems. These data are useful in evaluating migration of microbes from the waste disposal trench. Both mineral and organic soils are effective filters of microbial cells from sewage and septic system effluents (11-17). Few difficulties are encountered, except in the cases of channeling and with saturated water flow. Thus, it can be concluded that construction of a compacted trench base and placement of the trench in regions of low soil water tables would minimize problems associated with bioaccumulation. Placement of trenches on sandy soils may create some difficulties in that microbial movement is greater in that soil type than in less permeable soils.

Material accumulated on growing microbial cells would be expected to remain attached to the cell until the death and subsequent decay of the cell constituents. Longevity of the microbes surrounded with the radionuclides may be anticipated to be shorter than that of the microbe adsorbing a nonradioactive metal, such as iron. Wildung and Garland (18) observed a significant decline in aerobic spore forming and anaerobic bacteria in soil amended in hydrolyzable plutonium ( $^{239}\text{Pu}(\text{NO}_3)_4$ ) at concentrations as low as 1  $\mu\text{g/g}$  soil. Thus, stability of the radionuclide-microbe complex could be threatened by the radiation induced death of the microbe. Unfortunately, the effect of the attached radionuclide molecules on the mineralization of the dead cell debris is unknown. It could be hypothesized that sufficient radionuclide concentrations could be bound to the cell debris to inhibit or prevent its subsequent biodegradation. Thus, should this be the case, retention of the radionuclides in non-mobile forms would be extended. Research to elucidate this possibility is suggested.

#### CHELATION EFFECTS ON RADIONUCLIDES

Problems with chelators in trench waters are divisible into two major areas: 1) chelators originating with the waste materials, and 2) chelators synthesized by the microbial community indigenous to the trench. Although augmented radionuclide mobility is the major problem associated with each group, the importance and impact of each source is unique. Thus, the two types of chelators will be discussed separately.

## CHELATORS ORIGINATING WITH THE WASTE MATERIALS

Chelators contained within the waste materials are primarily EDTA (ethylenediamine tetraacetic acid), DTPA (diethylenetriaminepentaacetic acid), and NTA (nitrilotriacetic acid). These compounds convert insoluble radionuclides into water soluble structures, which may migrate from the trench confines with trench waters (19). EDTA, DTPA and NTA are all biodegradable under aerobic conditions. Tiedje (20) reported the degradation of EDTA in soils and sediments under aerobic conditions and by microorganisms in mixed liquid cultures. Equivalent degradation rates were found for the copper, cadmium, zinc, manganese, calcium and iron salts in soil, whereas the nickel salt was degraded more slowly. No radionuclide salts were examined. Similar results were found by Belly et al. (21) with EDTA decomposition in an aerated lagoon. No degradation was found under anaerobic conditions.

Reports of DTPA degradation are less common. Wildung and Garland (18) demonstrated, however, that DTPA complexed with  $^{238}\text{Pu}$  and  $^{239}\text{Pu}$  were degraded by about 20% over a 95-day period when incubated aerobically.

Studies of NTA decomposition are more abundant. Tiedje and Mason (22) found NTA decomposition in a variety of agricultural soils receiving sewage effluent. Degradation was observed with NTA concentrations of 10 to 600 ppm and temperatures of 2 to 24°C, but not under anaerobic conditions. Similarly, Barnhart et al. (23) found no NTA decomposition under anaerobic conditions in the absence of nutrient supplements. Enfors and Molin (24,25) have reported anaerobic degradation of this chelator; nitrate was used as terminal electron acceptor. A wide variety of bacteria have been isolated that are capable of converting NTA to carbon dioxide and ammonia in the presence of oxygen (26-31).

These data demonstrate that DTPA, NTA and EDTA degradation occurs slowly under aerobic conditions and that little decomposition can be expected under anaerobic conditions. The sole case of significant anaerobic decomposition indicated a requirement for nitrate as a terminal electron acceptor. Nitrate would not be expected to occur in significant concentrations in low level radioactive waste disposal trenches. Thus, it is reasonable to conclude that reduction of solubility of nuclide chelates via microbial action would be minimal in the low level radioactive waste disposal trench. A more probable reduction of radionuclides complexing with these chelates would result from exchange of radionuclides with metals such as iron, which have a higher affinity for binding to the chelator.

## CHELATORS SYNTHESIZED BY MICROORGANISMS

A wide variety of compounds that may solubilize radionuclides are produced by the microbial community. These include hydroxamates (produced by bacteria for solubilizing iron), fatty acids and metabolic products, as well as humic and fulvic acids, and metabolic intermediates

such as 2-ketogluconic acid. As indicated above, fatty acids are produced during anaerobic decomposition of the carbonaceous waste substituents in the disposal trenches. Fulvic and humic acids are soil organic matter components, which result from microbial and chemical transformations of the more biodegradation resistant carbon residues (32). These residues include the lignin common in plant debris and products from plants such as paper. Because of the abundance of acidic groupings on these molecules, solubilization of cationic nuclide residues will occur. Complexing with humic acid would be of little consequence in that these acids are not water soluble. Significance of the other complexes will depend upon the capability of iron, and other metals known to be present in trench waters, to displace the radionuclides, and the biodegradability of radionuclide-acid complexes. Further research is suggested to elucidate this problem.

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Corrective Measures Technology for Humid Sites<sup>1</sup>

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## ABSTRACT

Projects in the humid region corrective measures technology task at ORNL include a passive ground water diversion demonstration, a trench grouting demonstration, a trench chemical treatment demonstration, and a ground water seep chemical treatment demonstration. The passive drainage demonstration involves the construction of a French drain around a group of 49 disposal trenches which are seasonally intruded by ground water due to lateral recharge. The technique is generically applicable to any situation where the water table is maintained from an adjoining recharge area which may not necessarily be used for waste disposal. The trench grouting demonstration employs sodium silicate and polyacrylamide grouts to hydrologically isolate buried waste in previously backfilled trenches. The technique is most applicable to those situations where an individual trench can be identified as a major source of ground water radionuclide contamination. The trench chemical treatment subtask involves the injection of caustic soda and soda ash into a burial trench with a high inventory of <sup>90</sup>Sr. The field demonstration of this technique has brought about a dramatic lowering of <sup>90</sup>Sr concentrations in the interstitial ground water within a disposal trench and the effect has persisted over two years. In a generic hazard evaluation, <sup>90</sup>Sr was the most likely radionuclide to lead to poor site performance and, hence, it warrants the most concern for shallow land burial. This conclusion is empirically supported by the operational experience of the ORNL solid waste storage areas. The ground water seep treatment subtask addresses the problem of ground water seep contamination of surface water, a situation which may mandate immediate corrective action. Burial of soda ash in the ground water flowpath leading to the seep can effectively reduce <sup>90</sup>Sr concentrations for a 90-day period, which can be repeated as often as necessary, during which time a more permanent corrective action, as discussed above, can be implemented at the source.

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<sup>1</sup>Research sponsored by U.S. Department of Energy under contract W-7405-eng-26 with Union Carbide Corp.

## INTRODUCTION

Shallow land burial (SLB) has been and will continue to be the dominant disposal method for most low-level radioactive solid waste. SLB trenches usually perform well in retaining radionuclides and in preventing their migration to uncontrolled areas. Occasionally, trenches do leak radionuclides to the extent that corrective measures seem warranted. Such poorly performing trenches have not only been observed at ORNL (Duguid, 1975), but also at Maxey Flats (Montgomery and Blanchard, 1979) and at West Valley (Kelleher, 1979). The DOE Technical Advisory Committee has identified six problems with SLB which can lead and have led to situations requiring corrective actions. These include the infiltration of surface runoff into burial trenches, ground water intrusion, trench subsidence, soil erosion, trench intrusion by plants and animals, and radionuclide migration. In areas of the country where annual precipitation exceeds evapotranspiration, like ORNL, the problems of surface runoff control, ground water intrusion, and radionuclide migration are of paramount importance while the problems of subsidence, erosion, and biointrusion can be considered secondary. The former problems demand attention early in the lifetime of a disposal trench whereas the latter problems tend to become more important in the long term if the former problems have been alleviated.

A hypothetical trench with all of these identified problems is depicted in Figure 1. The seasonal fluctuation of the water table leads to periods when the buried waste is bathed by ground water. Additional inundation is facilitated by direct infiltration of precipitation and the catching of any surface runoff due to partial subsidence of the trench backfill. Points of rapid trench subsidence or cover collapse can effectively function as drains for surface runoff leading to an accumulation of water which results in a perched water table within the trench. Obviously, the control and diversion of surface runoff can alleviate many of these inundation problems; a discussion of such a problem at ORNL was presented last year (Spalding and Huff, 1981). Trench cover sealing with bentonite or plastic sheeting has been advocated to relieve the situation of direct infiltration of precipitation (Duguid, 1976). However, trenches, like the one depicted in Figure 1, often are situated in areas where the water table is maintained by lateral ground water recharge rather than or in addition to direct infiltration of precipitation. Passive ground water interception and diversion drains provide a means to relieve such situations. Such engineered drainage alterations, however, can become economically prohibitive for large regions containing many trenches particularly when only one or a few trenches are the source of the problem. In these situations, hydrologic isolation of individual trenches can be addressed better by waste grouting. The injection of a grout throughout the void space of a burial trench, which upon grout set becomes impermeable to water, can effectively isolate buried waste from further ground water intrusion. In addition, since trench void spaces become filled, subsidence can also be effectively controlled.

## PROBLEMS ENCOUNTERED IN THE SHALLOW LAND BURIAL OF LOW-LEVEL WASTE

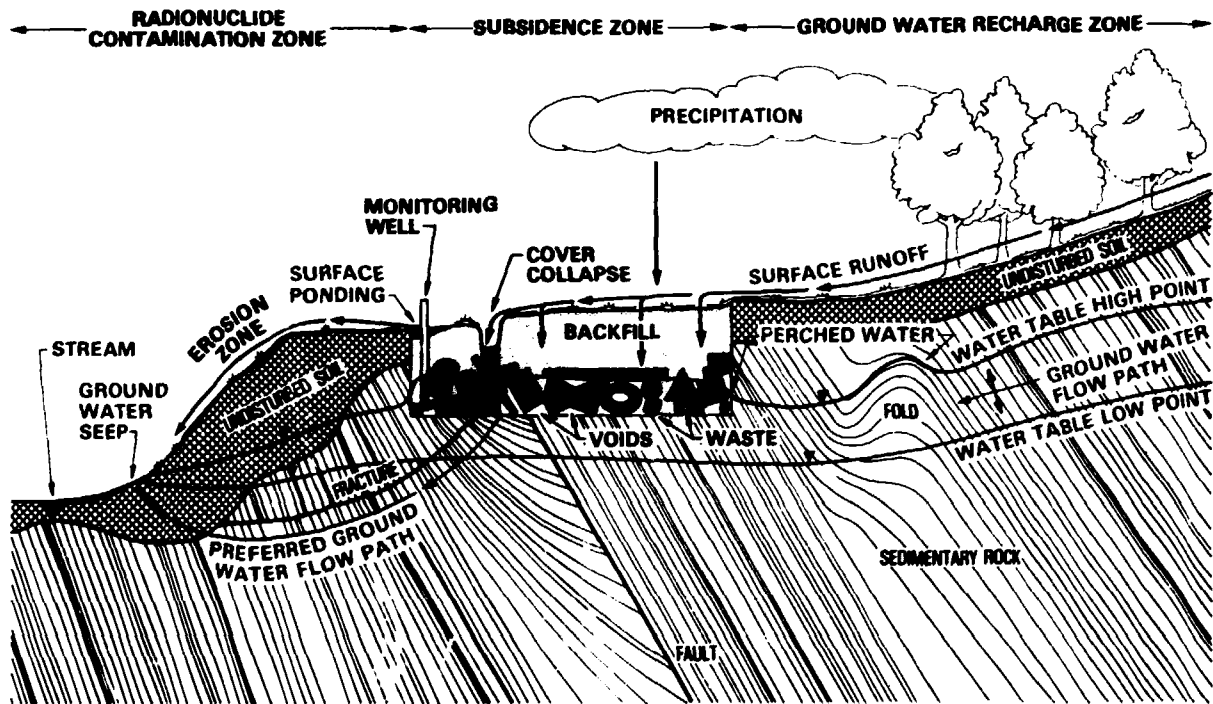


Fig. 1. Problems encountered in the shallow land burial of low-level waste to be addressed by corrective measures technology.

The ground water flow regimes in almost any existing field situation cannot be addressed under the usual simplifying assumption that soil and rock are a homogeneous porous media. The complexities of ground water flow are also depicted in Figure 1. Fractures, folds, and faults, inherited by the soil from the original rock, alter dramatically rates and directions of ground water flow. Routine ground water monitoring systems, usually based on some kind of regular grid, can often fail to intercept radionuclide migration along some geologically induced preferred route. Ground water contamination problems, therefore, often first reveal themselves in routine surface water monitoring. At such a time, corrective action may be required immediately and, therefore, effective techniques for treating ground water discharge points (seeps) need to be developed. Sources can then be treated by other corrective measures, such as drainage or grouting, which will effect a permanent solution to the radionuclide migration problem.

Considering the problem of radionuclide migration necessitates focusing on particular radionuclides so that those which might potentially compromise trench/site performance can be singled out from the many which pose significantly lower hazards. To achieve this, one must define a typical low-level radioactive solid waste and estimate the concentration of each radionuclide it contains. The variability of low-level wastes makes such a definition almost impossible but, for purposes of illustration, one waste stream has been selected for analysis. Boiling Water Reactor Filter Sludge (BWRFS) is the highest low-level waste identified by NRC (1981) and contains the broadest range of potentially hazardous radionuclides. These estimations of the concentrations in the waste, coupled with some estimate of their environmental mobility like the geologic retardation factors estimated by NRC (1981), allow radionuclides to be compared for their potential migration from a SLB trench. Using NRC's estimate for radionuclide concentrations in BWRFS and their estimate for soil retardation factors (Rd's), a hazard rating for each radionuclide can be calculated by factoring in its inherent radiological toxicity as reflected by its maximum permissible concentration (MPC) for the unrestricted use of water (NRC, 1979). Such a hazard rating, for each radionuclide, was calculated (Table 1):

$$\text{Hazard Rating} = \text{Concentration in BWRFS} / (\text{Rd} \times \text{MPC})$$

Such a hazard rating provides a quantitative method to compare radionuclides, which occur at differing concentrations in the waste, which are adsorbed to differing degrees by the soil in which they must migrate, and which have inherently different radiological hazards. This calculation provides a triage by which radionuclides ought to be considered for specific investigation. One can also see what makes a radionuclide like  $^{129}\text{I}$  result in the highest hazard rating; many other radionuclides occur at higher concentrations in BWRFS but most have much higher soil Rd's and much higher MPC's. However, monitoring of ORNL surface water discharges has never found that  $^{129}\text{I}$  makes a detectable contribution to the radiological dose of ORNL effluents; the

Table 1. A generic hazard evaluation of radionuclides encountered in the disposal of low-level radioactive waste.

| Radionuclide      | Concentration<br>in BWRFS<br>(Ci/m <sup>3</sup> ) | Soil Rd<br>(mL/g) | MPC<br>(mCi/L)       | Hazard<br>Rating <sup>4</sup> |
|-------------------|---------------------------------------------------|-------------------|----------------------|-------------------------------|
| <sup>3</sup> H    | 1.26 x 10 <sup>-2</sup>                           | 1                 | 3 x 10 <sup>-3</sup> | 4.2                           |
| <sup>14</sup> C   | 7.78 x 10 <sup>-4</sup>                           | 10                | 8 x 10 <sup>-4</sup> | 9.7 x 10 <sup>-2</sup>        |
| <sup>55</sup> Fe  | 1.44 x 10 <sup>-0</sup>                           | 2640              | 8 x 10 <sup>-4</sup> | 6.8 x 10 <sup>-1</sup>        |
| <sup>59</sup> Ni  | 1.49 x 10 <sup>-3</sup>                           | 1750              | 2 x 10 <sup>-4</sup> | 4.3 x 10 <sup>-3</sup>        |
| <sup>60</sup> Co  | 2.41 x 10 <sup>-0</sup>                           | 1750              | 5 x 10 <sup>-5</sup> | 2.8 x 10 <sup>1</sup>         |
| <sup>63</sup> Ni  | 3.25 x 10 <sup>-2</sup>                           | 1750              | 3 x 10 <sup>-5</sup> | 6.2 x 10 <sup>-1</sup>        |
| <sup>94</sup> Nb  | 4.70 x 10 <sup>-5</sup>                           | 4640              | 3 x 10 <sup>-6</sup> | 3.4 x 10 <sup>2</sup>         |
| <sup>90</sup> Sr  | 2.37 x 10 <sup>-3</sup>                           | 36                | 3 x 10 <sup>-7</sup> | 2.2 x 10 <sup>2</sup>         |
| <sup>99</sup> Tc  | 5.00 x 10 <sup>-5</sup>                           | 4                 | 3 x 10 <sup>-4</sup> | 4.2 x 10 <sup>-2</sup>        |
| <sup>129</sup> I  | 1.33 x 10 <sup>-4</sup>                           | 4                 | 6 x 10 <sup>-8</sup> | 5.5 x 10 <sup>2</sup>         |
| <sup>135</sup> Cs | 5.00 x 10 <sup>-5</sup>                           | 350               | 1 x 10 <sup>-4</sup> | 1.4 x 10 <sup>-3</sup>        |
| <sup>137</sup> Cs | 1.33 x 10 <sup>-0</sup>                           | 350               | 2 x 10 <sup>-5</sup> | 1.9 x 10 <sup>2</sup>         |
| <sup>235</sup> U  | 3.32 x 10 <sup>-7</sup>                           | 3520              | 3 x 10 <sup>-5</sup> | 3.1 x 10 <sup>-6</sup>        |
| <sup>238</sup> U  | 2.61 x 10 <sup>-6</sup>                           | 3520              | 4 x 10 <sup>-5</sup> | 1.9 x 10 <sup>-5</sup>        |
| <sup>237</sup> Np | 6.38 x 10 <sup>-11</sup>                          | 1200              | 3 x 10 <sup>-6</sup> | 1.7 x 10 <sup>-8</sup>        |
| <sup>238</sup> Pu | 4.66 x 10 <sup>-4</sup>                           | 3520              | 5 x 10 <sup>-6</sup> | 2.7 x 10 <sup>-2</sup>        |
| <sup>239</sup> Pu | 2.36 x 10 <sup>-4</sup>                           | 3520              | 5 x 10 <sup>-6</sup> | 1.3 x 10 <sup>-2</sup>        |
| <sup>241</sup> Pu | 1.15 x 10 <sup>-2</sup>                           | 3520              | 2 x 10 <sup>-4</sup> | 1.6 x 10 <sup>-2</sup>        |
| <sup>242</sup> Pu | 5.18 x 10 <sup>-7</sup>                           | 3520              | 5 x 10 <sup>-6</sup> | 2.9 x 10 <sup>-5</sup>        |
| <sup>241</sup> Am | 1.56 x 10 <sup>-4</sup>                           | 1200              | 4 x 10 <sup>-6</sup> | 3.3 x 10 <sup>-2</sup>        |
| <sup>243</sup> Am | 1.05 x 10 <sup>-5</sup>                           | 1200              | 4 x 10 <sup>-6</sup> | 2.2 x 10 <sup>-3</sup>        |
| <sup>243</sup> Cm | 2.97 x 10 <sup>-7</sup>                           | 1200              | 5 x 10 <sup>-6</sup> | 5.0 x 10 <sup>-5</sup>        |
| <sup>244</sup> Cm | 2.24 x 10 <sup>-4</sup>                           | 1200              | 7 x 10 <sup>-6</sup> | 2.7 x 10 <sup>-2</sup>        |

<sup>1</sup>BWRFS = Boiling Water Reactor Filter Sludge per NRC (1981).

<sup>2</sup>Case 1 - Soils with moderate permeability per NRC (1981).

<sup>3</sup>Maximum Permissible Concentration per NRC (1979).

<sup>4</sup>Hazard Rating = BWRFS Concentration / ( Rd x MPC ).

major contributor in our effluents, particularly from the solid waste storage areas, is  $^{90}\text{Sr}$ . Notably, this radionuclide ranks second in this generic hazard analysis (Table 1). Next in the generic rankings is  $^{137}\text{Cs}$  which appears because of its high concentration in the waste and its comparatively intermediate soil  $R_d$ ; actually, ORNL soils must contain considerably more illite than NRC's hypothetical soil since our  $R_d$ 's for  $^{137}\text{Cs}$  are about two orders of magnitude greater than those in Table 1 (Cerling and Spalding, 1981). Such information would considerably lower the Cs hazard rating for SLB at ORNL. From Table 1, it can be seen that all of the actinides are hardly worth investigation since they are orders of magnitude lower in hazard rating than most of the fission products. Such a hazard rating system merely serves to justify our focus on  $^{90}\text{Sr}$  as a particularly troublesome radionuclide for SLB, both generically and at ORNL (Duguid, 1975). Obviously, other waste streams could be selected to provide differing source concentrations of radionuclides for such an analysis, but few streams would result in a higher hazard rating for any radionuclide and most would be considerably less.

## RESULTS

The ORNL corrective measures technology program is divided into four subtasks: a passive ground water diversion-drainage demonstration, a trench grouting demonstration, a trench chemical treatment demonstration, and a ground water seep treatment demonstration.

### 1. Passive Drainage Demonstration

This subtask addresses the situation where ground water intrusion of a group of trenches is maintained by lateral recharge rather than by direct infiltration of precipitation. By constructing a ground water intercepting and diverting drain between the area to be protected and its recharge area, the ground water recharge can be eliminated. For purposes of a demonstration of this technology, a subsection of solid waste storage area (SWSA) 6 at ORNL, containing 49 burial trenches, was selected as a site for construction of a French drain (Figure 2). Other engineering options to achieve a 1.7 meter water table lowering were considered in a conceptual design/option study and included slurry trenches, drilled concrete piers, pre-fabricated caissons, steel sheet pile cutoff walls, and horizontally-drilled drains. The relative costs of the various options clearly pointed to the basic French drain as the least expensive option. The location of this proposed French drain is illustrated in Figure 2. The suite of 49 trenches, to be protected by the drain, was surface sealed with a soil-bentonite mixture in November, 1976; however, seasonally standing water still exists in many of the trenches (Arora et al., 1981). Whether or not this bentonite seal is working in this area cannot be determined from observations of the water table alone; in addition to any infiltration of precipitation, allowed by a poorly performing seal, the water table in

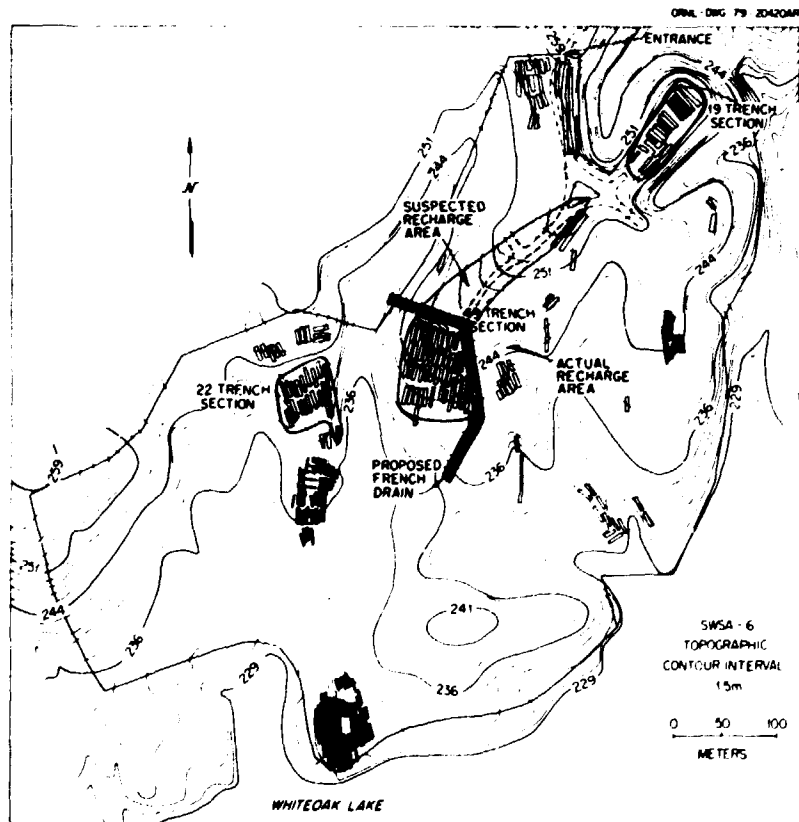


Fig. 2. Location of proposed French drain for ground water diversion demonstration in Solid Waste Storage Area 6 at ORNL.



this area is maintained by lateral ground water recharge from areas not under the seal. Interestingly, based on the surface topographic contours (Figure 2) and water table elevation data in the area (Webster et al., 1980), a traditional hydrologic analysis would indicate that the recharge area would be the shaded area in Figure 2. However, the area immediately east of these 49 trenches is probably more important as a recharge zone because it lies along the line of joint fractures in the area (Sledz and Huff, 1981) which function as preferred ground water flowpaths. Additional experimental determinations of the ground water velocities and directions along the line of the proposed drain are being collected. Regardless of the relative contributions of these two possible recharge areas, the 49 trench area will be protected by the surrounding drain. Test borings along the proposed drain have been completed and no hard rock was encountered within the 7.6 meter maximum depth of the drain. Construction will begin in the near future and, in the pre-construction interval, water table elevations will continue to be monitored to add to the existing five-year record for the area (Webster et al., 1980).

## 2. Trench Grouting Demonstration

This subtask addresses the situation where a particular trench, rather than a group of trenches, is functioning as a source of radionuclide ground water contamination and, therefore, compromising the performance of a site. By the injection of a low-viscosity and non-particulate grout into the void space of a problem trench, the buried waste can be hydrologically isolated after the grout sets to its impermeable state. Two basic problems are faced by the potential grout user for any application as a corrective measure for low-level radioactive solid waste. Firstly, of the many grout types and formulations presently available (Herndon and Lenahan, 1976), a selection must be made using criteria including a low grout viscosity similar to that of water, a set time controllable from minutes to days, an inherently low permeability after set, durability to resist the physiochemical and microbial stresses in its applied environment, low toxic hazard, and, of course, low cost. Any grout selection will, of necessity, represent a compromise among these properties. Based on existing data (Herndon and Lenahan, 1976), five grout formulations were selected for laboratory testing as indicated in Table 2; the fundamental property for this laboratory evaluation was the ability of the grout to reduce the hydraulic conductivity of several soils and wastes types. A standard saturated hydraulic conductivity determination (Klute, 1965) was performed on columns of soil before and after grouting. Both 10% polyacrylamide (Avanti Inc., 1981) and 15% sodium silicate (Diamond Shamrock Inc., 1964) performed well (Table 2) and were selected for field demonstration.

The second basic problem to be faced by any grout user for such corrective measures application is how to achieve an equitable distribution of the grout within the target trench. Since both the acrylamide and sodium silicate grout formulations are low viscosity and

Table 2. Hydraulic conductivities of grouted materials.

| Grout                         | Uranium Mill Tailings |                       |                       | Quartz Sand           | ORNL Backfill Soil     |
|-------------------------------|-----------------------|-----------------------|-----------------------|-----------------------|------------------------|
|                               | Shiprock              | Salt Lake             | Durango               |                       |                        |
| ----- cm/sec -----            |                       |                       |                       |                       |                        |
| None                          | 5.5x10 <sup>-3</sup>  | 3.8x10 <sup>-3</sup>  | 4.1x10 <sup>-3</sup>  | 3.9x10 <sup>-2</sup>  | 5.2x10 <sup>-2</sup>   |
| Polyacrylamide (10%)          | 5.4x10 <sup>-5</sup>  | <2.6x10 <sup>-7</sup> | <2.6x10 <sup>-7</sup> | <2.6x10 <sup>-7</sup> | <2.6x10 <sup>-7</sup>  |
| Bentonite (5%)                | <2.6x10 <sup>-7</sup> | <2.6x10 <sup>-7</sup> | <2.6x10 <sup>-7</sup> | <2.6x10 <sup>-7</sup> | n.d.                   |
| Resorcinol-Formaldehyde (20%) | 1.1x10 <sup>-3</sup>  | 8.7x10 <sup>-4</sup>  | <2.6x10 <sup>-7</sup> | <2.6x10 <sup>-7</sup> | n.d.                   |
| Urea-Formaldehyde (20%)       | 3.4x10 <sup>-3</sup>  | 1.4x10 <sup>-3</sup>  | 2.5x10 <sup>-3</sup>  | 2.6x10 <sup>-2</sup>  | n.d.                   |
| Sodium silicate (15%)         | 1.4x10 <sup>-6</sup>  | 7.0x10 <sup>-7</sup>  | 5.2x10 <sup>-4</sup>  | 2.6x10 <sup>-4</sup>  | 1.2x10 <sup>-5**</sup> |

\* In all cases, bentonite failed to infiltrate the soil columns and the reported low conductivities are the result of surface plugging.

\*\*In this case, 8% sodium silicate was used instead of 15%.

n.d. = not determined.

non-particulate, they should flow within trench voids as facily as intruding ground water. Therefore, grout injection could be effected with a minimum number of injection wells; because the hydraulic conductivity of a trench's waste/backfill is much greater than the surrounding unperturbed soil formation, trenches can be facily filled with fluids. To test this critical assumption for waste grouting application, three disposal trenches (Figure 3) were constructed in the suspect waste landfill area of SWSA6. These three trenches were selected to encompass the range of water table regimes to which grouting might be applied: a trench chronically inundated with ground water, a trench seasonally inundated with ground water, and a trench chronically above the water table. Trenches were filled with suspect solid waste and backfilled with soil as SLB is currently practiced at ORNL. A grid of nine wells was positioned in each trench (Figure 3) and each trench was surrounded by four additional wells within the unperturbed formation. By measuring the hydraulic conductivity of the material surrounding each of these wells through constant-head pump in tests (Bouwer and Jackson, 1974), grouting effectiveness can be determined by comparing before and after grouting measurements. The grid of wells, in addition to measuring the waste/backfill and soil formation conductivity, provides a means to determine the areal extent of grout penetration. Success of this grouting demonstration will be interpreted by a reduction in the hydraulic conductivity of the waste/backfill to an order of magnitude less than that of the unperturbed formation.

Pre-grouting pump-in tests have been completed for all of the wells within and around one of the trenches although the calculation of hydraulic conductivities from these observations has not yet been completed. This trench has subsequently been grouted with 4,500 L of 8% sodium silicate (Siroc 132, Diamond Shamrock Inc., 1964) using a Chem Grout (1981) Model CG-500 dual tank mixer and pump for grout delivery to a central injection well. An additional trench will be grouted with sodium silicate in September, 1982.

### 3. Chemical Trench Treatment Demonstration

As discussed in more detail last year (Spalding and Huff, 1981), treatment of trenches with caustic soda and soda ash can lead to a significant fixation or immobilization of  $^{90}\text{Sr}$ . As discussed earlier,  $^{90}\text{Sr}$  is likely the most compromising radionuclide to the performance of a SLB site. Laboratory studies, employing both test tube batch and soil columns, have indicated that  $\text{Ca}(^{90}\text{Sr})\text{CO}_3$  is an excellent phase to form in soil in situ by the addition of soda ash and/or caustic soda (Spalding, 1981). A number of concerns remained about the applicability of this technique to acidic soils which are inherently low in available Ca from which  $\text{CaCO}_3$  will form on the addition of soda ash. Several acidic soils (Table 3) were treated with soda ash and supplemental Ca or Ba to supply adequate amounts of alkaline earths to effect a precipitation of  $\text{CaCO}_3$  or  $\text{BaCO}_3$ , respectively. Soil columns were treated with  $^{85}\text{Sr}$  and chemicals as

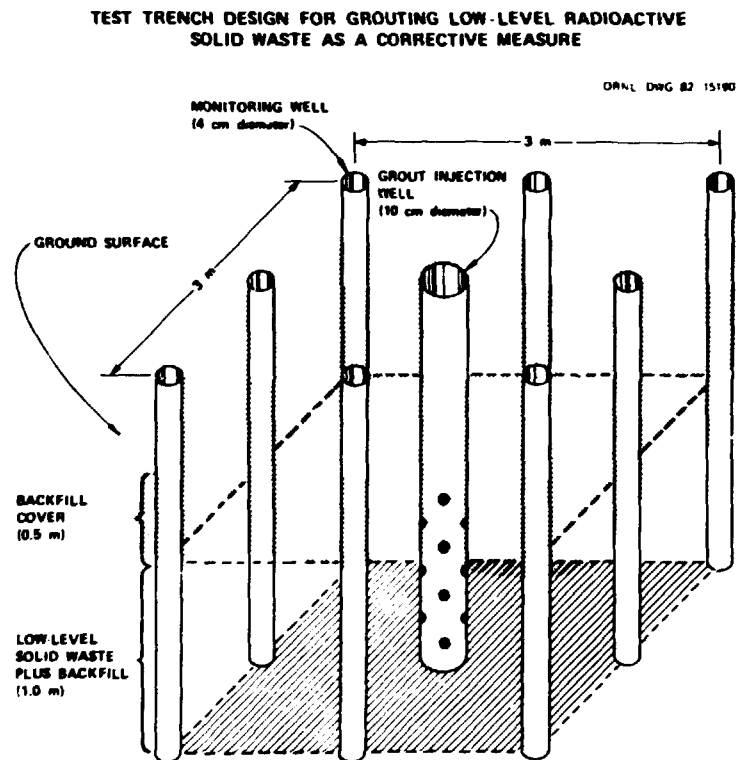


Fig. 3. Trench design for grouting technology demonstration.

Table 3. Chemical properties of four acidic soils used for soda ash treatment to immobilize radiostrontium.

| Attribute                | Units    | Soil |      |      |      |
|--------------------------|----------|------|------|------|------|
|                          |          | LSI  | LSII | FBI  | FAI  |
| Organic Carbon           | %        | <0.1 | <0.1 | <0.1 | 6.4  |
| Exchangeable:            |          |      |      |      |      |
| Acidity                  | meq/100g | 19.6 | 14.1 | 8.1  | 15.2 |
| Ca                       | meq/100g | 0.1  | 4.1  | 0.4  | 9.6  |
| Mg                       | meq/100g | 1.2  | 4.2  | 0.6  | 1.3  |
| K                        | meq/100g | 0.3  | 0.2  | 0.1  | 0.3  |
| Na                       | meq/100g | <0.1 | <0.1 | <0.1 | <0.1 |
| Cation Exchange Capacity | meq/100g | 21.3 | 22.7 | 9.3  | 26.4 |
| pH                       | -log[H]  | 3.7  | 4.1  | 3.9  | 5.1  |

<sup>1</sup>LSI and LSII = Litz-Sequoia association, C horizon, a clayey, mixed mesic typic Hapludult, parent material = Conasauga Group. FBI and FAI = Fullerton series, B and A horizons, respectively, a clayey, kaolinitic, thermic typic Paleudult, parent material = Knox dolomite.

described previously (Spalding, 1981) and eluted with dilute  $\text{CaCl}_2$  with the results depicted in Figure 4. Supplemental Ca or Ba did effect a greater fixation of  $^{85}\text{Sr}$  than soda ash alone as evidenced by the positions of the plateaus of the cumulative elution curves (Figure 4). After observations had indicated that little additional  $^{85}\text{Sr}$  was being eluted, all columns were eluted with 0.1 N HCl to remove the carbonate phase as indicated by the second step in the elution curves in Figure 4. Thus a two-step system for the chemical fixation of radiostrontium in strongly acidic soils would seem quite feasible and considerably broadens the range of soils to which the technique is applicable.

To illustrate the degree to which ORNL soils (as would any soil which contains significant amounts of illite clay) fix  $^{137}\text{Cs}$ , soil columns were labelled from the top with  $^{137}\text{Cs}$  and eluted exhaustively (i.e., >20 column volumes) with water and none of the  $^{137}\text{Cs}$  could be leached. Autoradiographs of these 10-cm soil columns were prepared which clearly show that the  $^{137}\text{Cs}$  had not moved and had remained at the top of the soil column (Figure 5). Also illustrated is the finding that neither the supplementary Ba or Ca treatments had any effect on this  $^{137}\text{Cs}$  fixation.

This chemical treatment technology also required considerable model development and adaptation to describe the long term behavior of these treatments. Simulations of radionuclide migration using the Lapidus-Amundson equation (Spalding and Hoffman, 1982) indicated that this model was adequate to describe the elution of a peak or pulse input of radionuclide under conditions where a linear adsorption isotherm is followed. Typical elution profiles of hypothetical radionuclide species of differing  $K_d$ 's, based on realistic inputs to the Lapidus-Amundson (1952) equation, are depicted in Figure 6. Experimental determination of the model input parameters using high pressure liquid soil column radiochromatography has been made with real-time radioactivity detection (Figure 7). Such experimental approaches have proven quite useful in evaluating various soils for radionuclide adsorption properties.

The field demonstration of the trench treatment with caustic soda and soda ash, reported last year (Spalding and Huff, 1981), has continued in its monitoring phases. Little change in the trends of the observations reported then has occurred to the present. Hydrologic characterization of the trench environs has been attempted with some water pump-in tests. The resulting hydrologic characteristics will be used to evaluate the chemical behavior of this trench through time and will be used to formulate a model of the trench environs from which conclusions about trench performance beyond the period of experimental observation can be based.

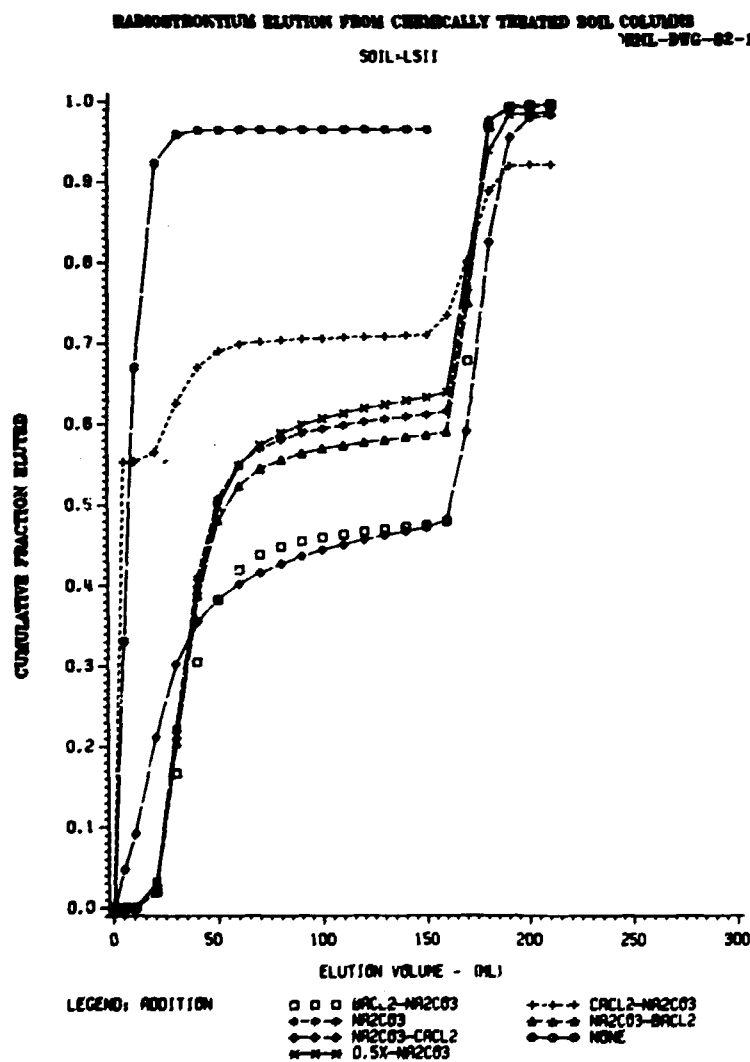


Fig. 4. Elution of  $^{85}\text{Sr}$  from an acidic soil (Litz-Sequoia II) as affected by soda ash treatments.

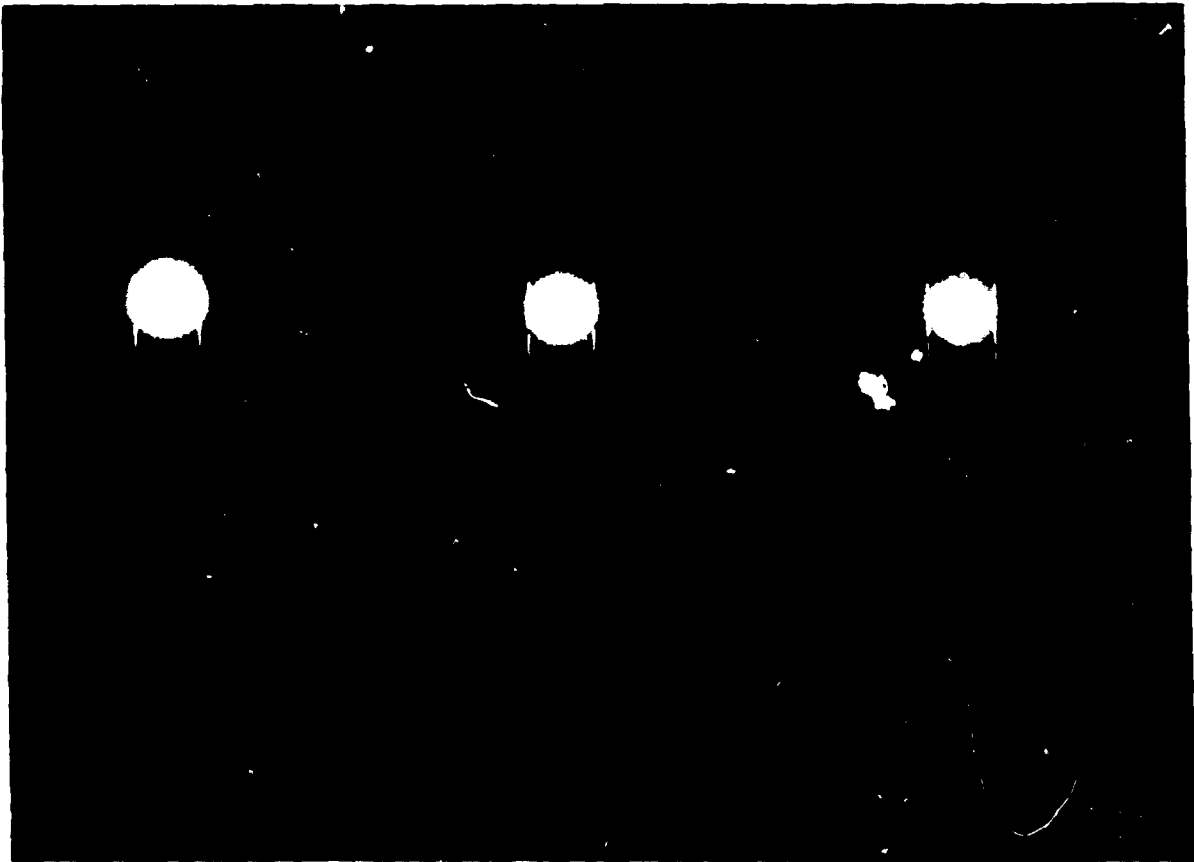


Fig. 5. Autoradiographs of  $^{137}\text{Cs}$ -labelled soil columns after leaching with greater than 20 column volumes 0.1 N  $\text{CaCl}_2$ .



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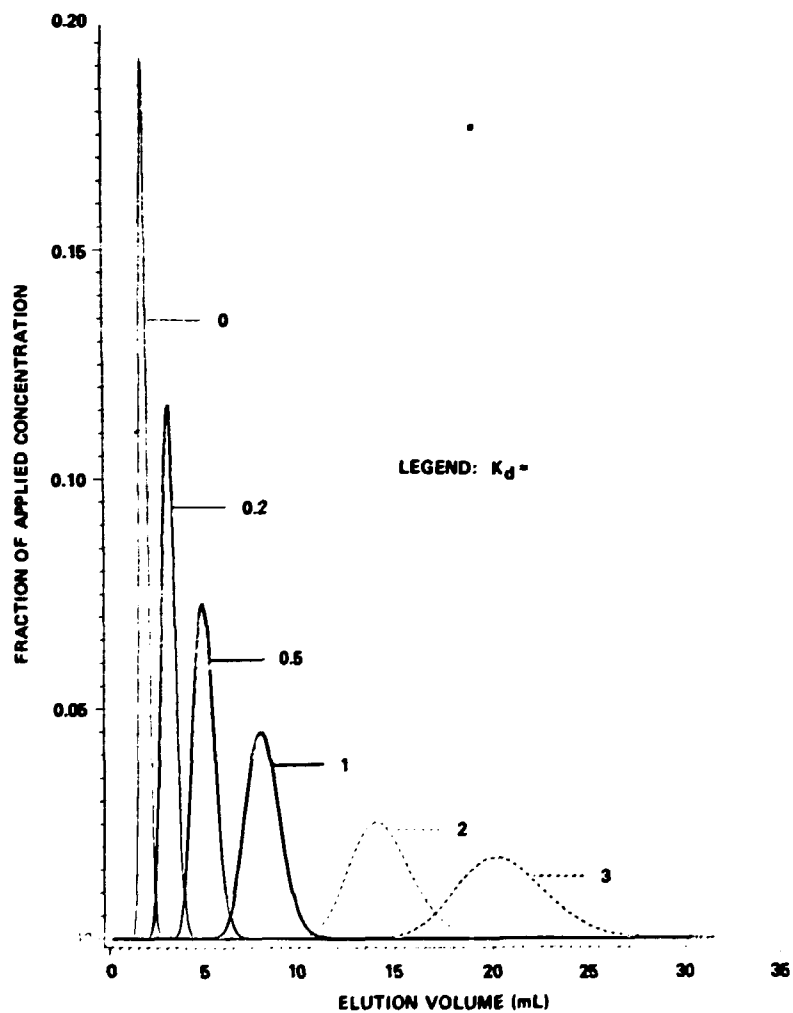


Fig. 6. Lapidus-Amundson equation simulation of soil column elution profiles of radionuclide species of increasing  $K_d$ .

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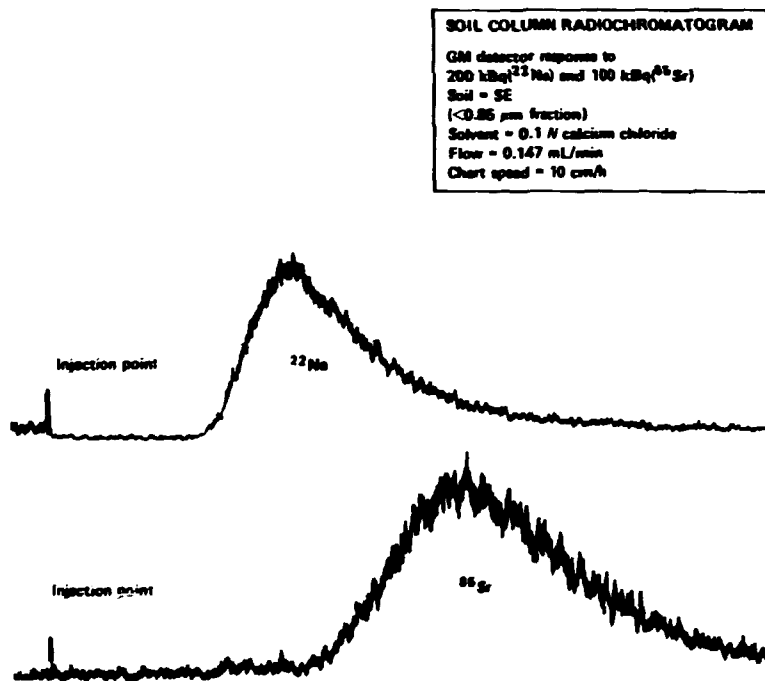


Fig. 7. Radiochromatogram of  $^{22}\text{Na}$  and  $^{85}\text{Sr}$  elution from a soil column.

#### 4. Ground Water Seep Treatment Demonstration

This subtask addresses the situation where a ground water seep is discharging radioactivity into surface water. Such a situation may require an immediate corrective action depending on the level of activity involved. Source treatments, like the three techniques discussed above, could not result in an immediate response at a seep discharge because the contamination plume, leading to the seep, could take several years to flush. Thus even interim techniques to address such situations have considerable merit. Many ground water seeps occur on the periphery of SWSA5 (Figure 8, from Duguid, 1976). One of these seeps, S-5, is still discharging a readily measureable concentration (i.e., ca. 7 Bq/mL) of  $^{90}\text{Sr}$  and was mapped out in more detail (Figure 9) after installing a weir, stage height recorder, and automatic water sampler at the site. After about a year of pre-treatment monitoring, 360 kg of soda ash were buried at the locations indicated in Figure 9, and the response of  $^{90}\text{Sr}$  and seep water quality were observed (Figure 10). Following soda ash burial pH and electrical conductivity of the seep water increased markedly while the hardness and Sr levels decreased concomitantly. An in-situ softening of this seep water, with the coprecipitation of  $^{90}\text{Sr}$  on calcite, was brought about by this treatment. The interaction of the varying flow rate at this seep with its chemical quality, probably led to much of the 'noise' in the data depicted in Figure 10. A smoothing of this 'noise' can be obtained by averaging these water quality parameters into three periods: 0 to 300 days before, 0 to 90 days after, and 91 to 310 days after soda ash burial (Table 4). Samples have also been classified in Table 4 as either grab, for those taken at one time during flow determination, or composite, for those composited over 24 to 48 hours. Roughly an order of magnitude decrease in  $^{90}\text{Sr}$  concentration was maintained for the 90-day period following soda ash burial which, in turn, was followed by a slow recovery to pre-treatment levels over the next year. Thus, the technique has merit as an interim corrective measure, which could be repeated as often as necessary, while more permanent corrective measures were carried out at the source. The major reason for its temporariness is the depletion of the soda ash via dissolution, neutralization, and calcite precipitation in the ground water flowpath. Since the incoming  $^{90}\text{Sr}$  in ground water was not stopped through a source treatment, once the soda ash was depleted, there was no mechanism for continued radiostrontium fixation.

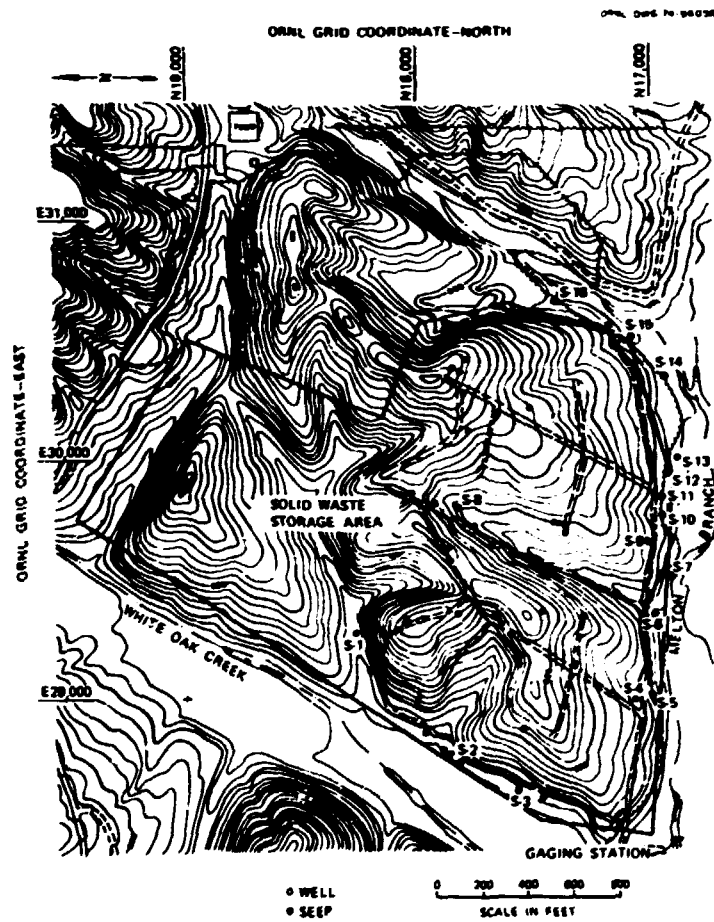


Fig. 8. Location of ground water seeps around Solid Waste Storage Area 5 at ORNL from Duguid (1975).

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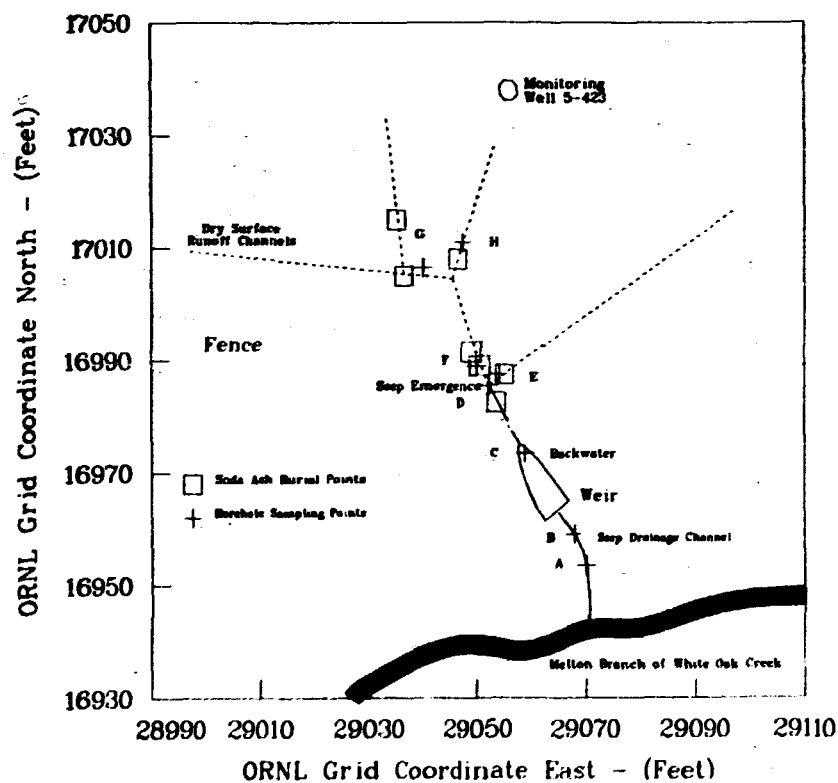


Fig. 9. Expanded map of a ground water seep in Solid Waste Storage Area 5 showing soda ash burial points.

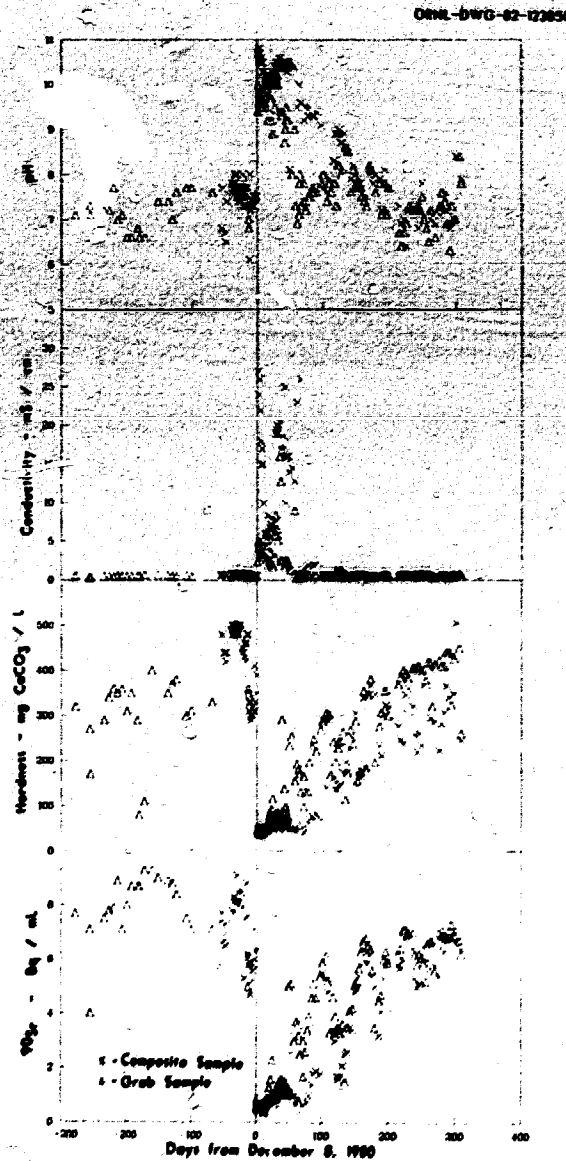


Fig. 10. Response of the  $^{90}\text{Sr}$ -contaminated seep water relative to the time of soda ash burial.

Table 4. Average chemical characteristics of seep water in the periods before, after, and during soda ash treatment.

|                   |                         | Period Relative to Soda Ash Burial |           |                 |           |                   |           |
|-------------------|-------------------------|------------------------------------|-----------|-----------------|-----------|-------------------|-----------|
|                   |                         | 0-300 days before                  |           | 0-90 days after |           | 91-310 days after |           |
| Characteristic    | Units                   | Grab                               | Composite | Grab            | Composite | Grab              | Composite |
| <sup>90</sup> Sr  | Bq/mL                   | 7.71d                              | 7.19d     | 1.75b           | 0.95a     | 5.62c             | 5.05c     |
| Hardness          | mg CaCO <sub>3</sub> /L | 309c                               | 433d      | 104a            | 61a       | 318c              | 260b      |
| Conductivity      | μS/cm                   | 564a                               | 678a      | 3300b           | 13000c    | 623a              | 703a      |
| pH                | -log[H]                 | 7.2a                               | 7.5ab     | 9.0c            | 10.0d     | 7.5ab             | 7.7b      |
| Saturation Index  | ΔpH                     | -0.06                              | 0.66      | 0.76            | 1.39      | 0.31              | 0.35      |
| Number of Samples | -                       | 22                                 | 41        | 48              | 45        | 54                | 52        |

\*Means followed by the same letter are not significantly different at the 5% level.

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## REMEDIAL ACTION TECHNOLOGY-ARID

*Aug*

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## ABSTRACT

A summary of the low-level waste remedial action program at Los Alamos is presented. The experimental design and progress is described for the experiments on second generation intrusion barriers, subsidence effects on SLB components, moisture cycling effects on chemical transport, and erosion methodologies.

The soil moisture data from both the bio-intrusion and moisture cycling experiments both demonstrate the overwhelming importance of vegetation in minimizing infiltration of water through trench covers and backfill. Evaporation, as a water loss component in trench covers, is only effective in reducing soil moisture within 40 cm of the trench cover surface. Moisture infiltrating past the zone of evaporation in unvegetated or poorly vegetated trench covers is in storage and accumulates until drainage out of the soil profile occurs. Judicious selection of vegetation species for vegetating a low-level waste site may prevent infiltration of moisture into the trench and, when coupled with other design features (i.e. trench cover slope, tilling and seeding practice), may greatly reduce problems with erosion.

Standard U.S. Department of Agriculture erosion plots, when coupled with a state-of-the-art water balance and erosion model (CREAMS), (a field scale model for Chemicals, Runoff and Erosion from Agricultural Management Systems), promises to be highly useful in screening proposed remedial action cover designs for low-level waste sites. The erosion plot configuration allows for complete accounting of the water balance in a soil profile. This feature enables the user to optimize cover designs to minimize erosion and infiltration of water into the trench.

## INTRODUCTION

This paper summarizes the status of experiments to develop and to evaluate arid site remedial action technology for low-level radioactive waste. The extensive data sets that are available for several of the remedial action subtasks have been reserved for publication elsewhere and will not appear in this summary.

Most of the problems requiring remedial action at low-level waste sites in the U.S. involve water and/or erosion of the trench cover, infiltration of water into the trench contributing to subsurface transport and subsidence, and capillary movement of water upward as a result of wetting-dry cycles (Fig. 1). Potential problems with biota include intrusion of plant roots, and burrowing animals into the waste and enhanced water related problems (i.e. increased infiltration and surface erosion) as a result of excessive animal burrowing in the cover soil and backfill.

Because most of the preceding problems either directly or indirectly involve processes interacting with the trench cover (Fig. 1), most of our remedial action studies have focused on this component of the shallow land burial system. The approach we have taken in developing remedial action technology for low-level waste sites is to recognize that physical and biological processes affecting site integrity are interdependent and, therefore, cannot be treated as separate problems. For example, vegetation cover on a site plays a vital role in regulating the behavior of water in the trench cover and backfill (i.e. erosion and infiltration). In retrospect, then, it is clear that an understanding of the nature of the interdependence of physical and biological processes is necessary to capitalize on relationships that increase waste stability.

Remedial action experiments that will be addressed in this paper are:

- (1) second generation bio-intrusion barrier testing,
- (2) subsidence effects on burial system components,
- (3) moisture cycling effects on chemical transport, and
- (4) erosion control technology.

## SECOND GENERATION BIO-INTRUSION BARRIER SYSTEMS

There are several experiments underway in the U.S. to develop bio-intrusion barrier systems for large volume solid wastes (1,2). At

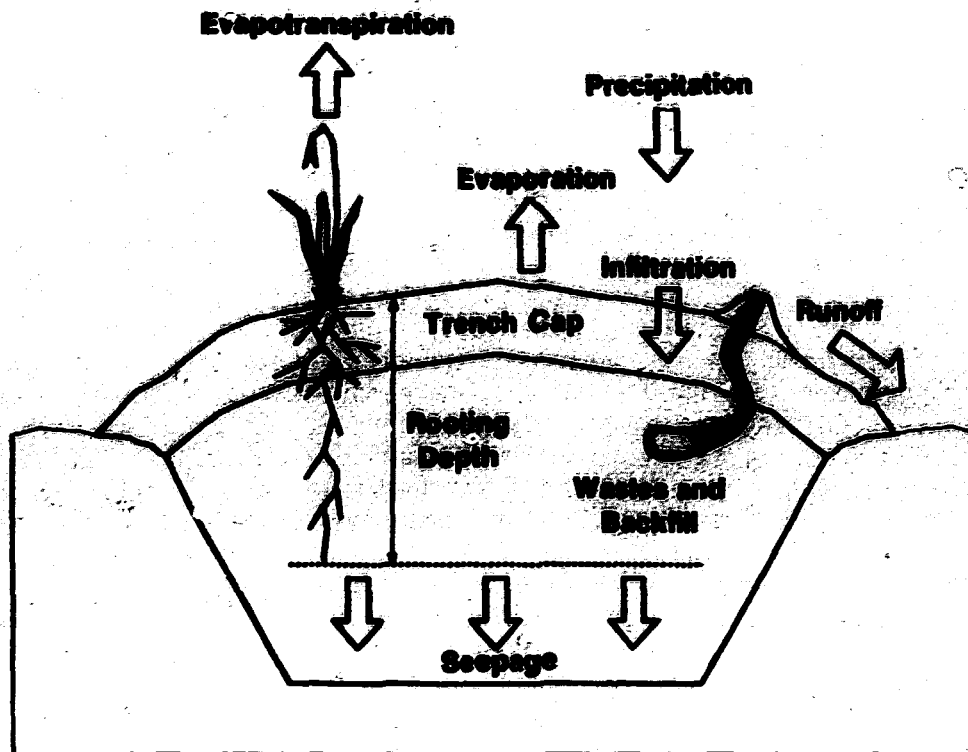


FIGURE 1. TRANSPORT PATHWAYS - SLB

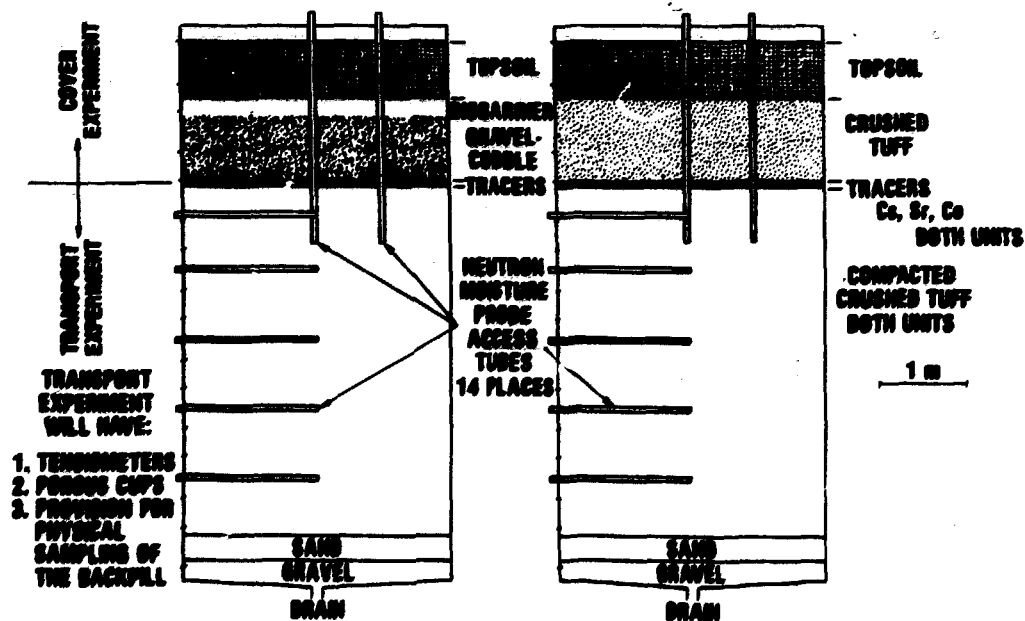


FIGURE 2. BIOINTRUSION BARRIER/TRANSPORT EXPERIMENT

Los Alamos we have pursued the feasibility of using geological materials as bio-intrusion barriers because these materials are not readily subject to decomposition and they are generally available and relatively inexpensive to apply.

Results of short-term, small scale experiments (1) conducted in 25 cm diameter lysimeters at Los Alamos Experimental Engineered Test Facility (EETF) demonstrated the effectiveness of layered cobble-gravel-soil cover systems in preventing plant root and burrowing animal intrusion into simulated waste compared with conventional cover systems. The next step in the further evaluation of cobble-gravel intrusion barriers for application to low-level waste sites requires answers to the following questions:

(1) Do cobble-gravel intrusion barriers perform satisfactorily at larger scales?

(2) What effect do layered rock barrier systems have on the water balance?

(3) How are the layered rock intrusion barrier systems affected by subsidence?

The questions about experiment scale and the effect of the rock barrier on the water balance are being addressed simultaneously in two separate experiments. An intermediate-scale experiment is underway in the large caissons at the EETF (Fig. 2), while a large scale study plot (Fig. 3) has been included in the cover system recently applied to Area B as a part of a necessary remedial action. The Area B low-level waste site was decommissioned in 1947. Both experiments include a cobble-gravel intrusion barrier along with a corresponding control (or conventional) cover treatment. Simulated waste placed in a narrow band immediately below the cover profiles will indicate root penetration through periodic sampling of vegetation for the plant-available tracers.

A cover profile consisting of either crushed tuff or cobble-gravel overlain by 60 cm of topsoil was placed over the simulated waste in the caisson experiments. In the Area B plots, the tracer was applied directly to the old waste cover surface. In the control plot, 90 cm of topsoil was applied over the tracer, similar to the treatment that the remainder of Area B received. The other plot consisted of the cobble-gravel intrusion barrier overlain by 60 cm of topsoil.

A topsoil depth of 60 cm was used on most of the plots as a compromise between optimum topsoil depth and cost. Simulations with a water balance model [CREAMS (3)] indicated that 60 cm of topsoil with a medium density (~25%) range grass cover would reduce the probability of water infiltrating into and through the barrier system by a factor

1 mm = 1 m

1 cm = 10 m

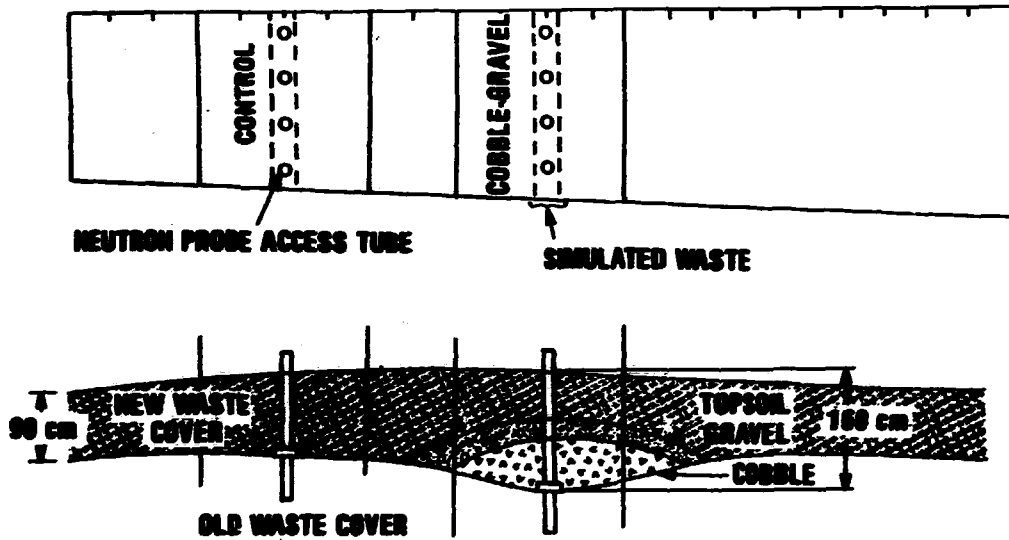


FIGURE 3. AREA B - LOW-LEVEL SITE BIOINTRUSION STUDY  
(SITE DECOMMISSIONED IN 1947)

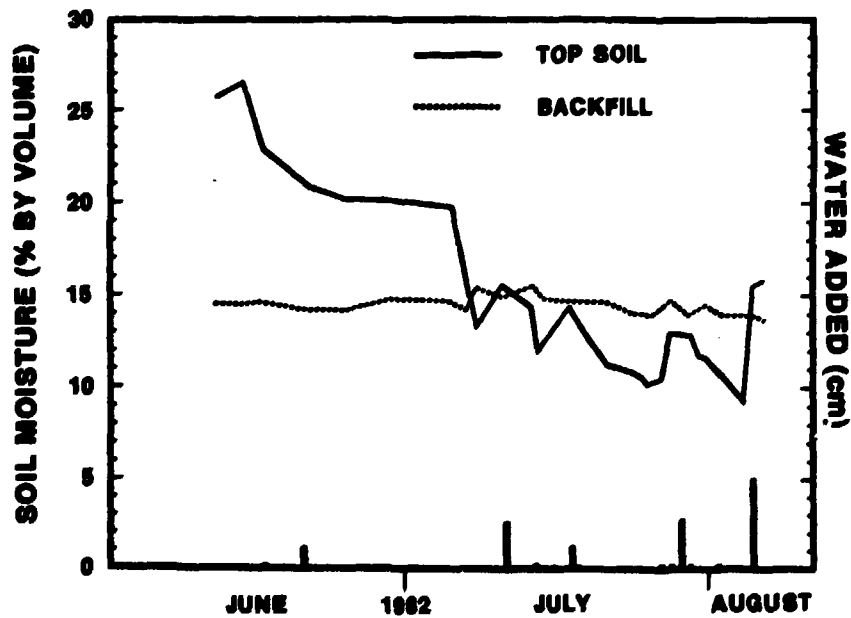


FIGURE 4. MOISTURE IN CRUSHED TUFF L. RIER  
COVER SYSTEM

of four compared with a bare soil surface. Additional topsoil, up to 90 cm, would eliminate the probability of water infiltrating through the rock barrier system and into the waste, but at additional expense and loss of waste storage volume.

Neutron moisture probe measurements have been made routinely in the cover to evaluate soil moisture status in each plot. The intermediate scale experiments in the caissons at the EETF receive both natural and supplemental precipitation while the large scale plots at Area B receive only natural precipitation. The caisson experiments were seeded to barley while the Area B plots were seeded with a mixture of native grasses.

Measurements that have been made through time include the tracer (simulated waste) content of vegetation and soil moisture versus depth in the cover and backfill.

Data on the moisture status of the topsoil and backfill above and below the crushed tuff and cobble-gravel intrusion barriers in the large caissons are plotted in Figs. 4 and 5, along with the precipitation added to each plot.

The moisture data in the backfill indicate no appreciable differences between plots despite acute inputs of as much as 5 cm (2 in.) precipitation. Backfill soil moisture has averaged about 14-15% by volume in both plots from the beginning of the experiment.

The importance of vegetation in controlling the moisture balance in the soil profile is shown by the moisture data for topsoil in Figs. 4 & 5. Initial soil moisture of about 26% by volume in both plots decreased steadily to about 10% during the first 4-6 weeks of the study. Although precipitation additions to the topsoil are reflected by increases in topsoil moisture, these additions have little lasting effect on moisture in storage. For example, the addition of 2.3 cm of water to the plots on July 29 was apparent in soil moisture for only about a 6 day period. Calculations suggest that evapotranspiration rates under these circumstances (28% ground cover of barley) are roughly  $10 \text{ L M}^{-2} \text{ D}^{-1}$  (or  $10^5 \text{ L Ha}^{-1} \text{ D}^{-1}$ ). Supplemental precipitation is now being added in increasing amounts to determine the maximum potential of the vegetated topsoil in preventing infiltration through the cover and into the simulated waste. The 5 cm water addition on August 9 represents the first of the additions. Comparisons of the response of soil moisture to the additions of water will be made with predictions based on the CREAMS water balance model (3) to further evaluate the potential use of CREAMS in designing low-level waste cover systems that minimize water related problems.

Data on the tracer content of vegetation from the beginning of the caisson experiment are not available at this time, nor are any of the data from the recently completed Area B plots.



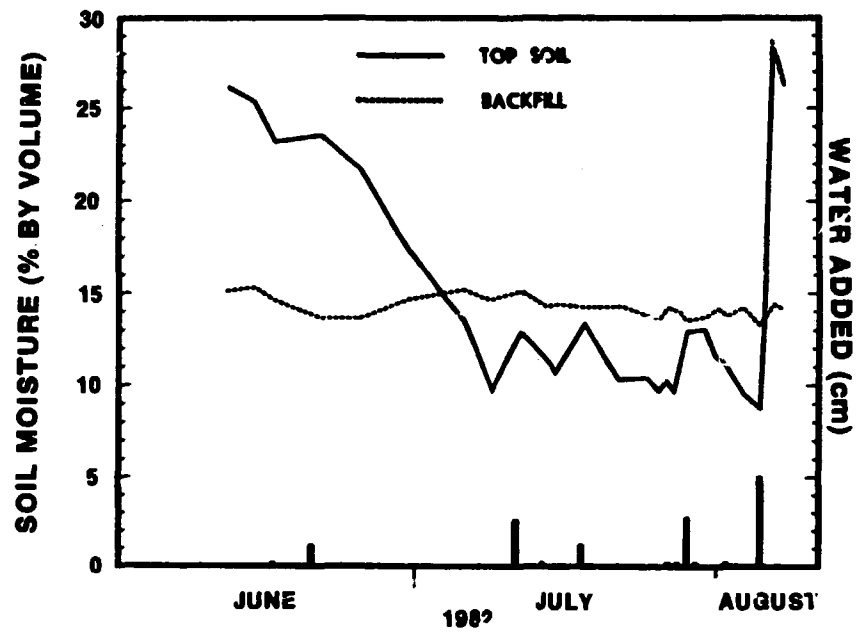


FIGURE 5. MOISTURE IN COBBLE-GRAVEL BARRIER COVER SYSTEM

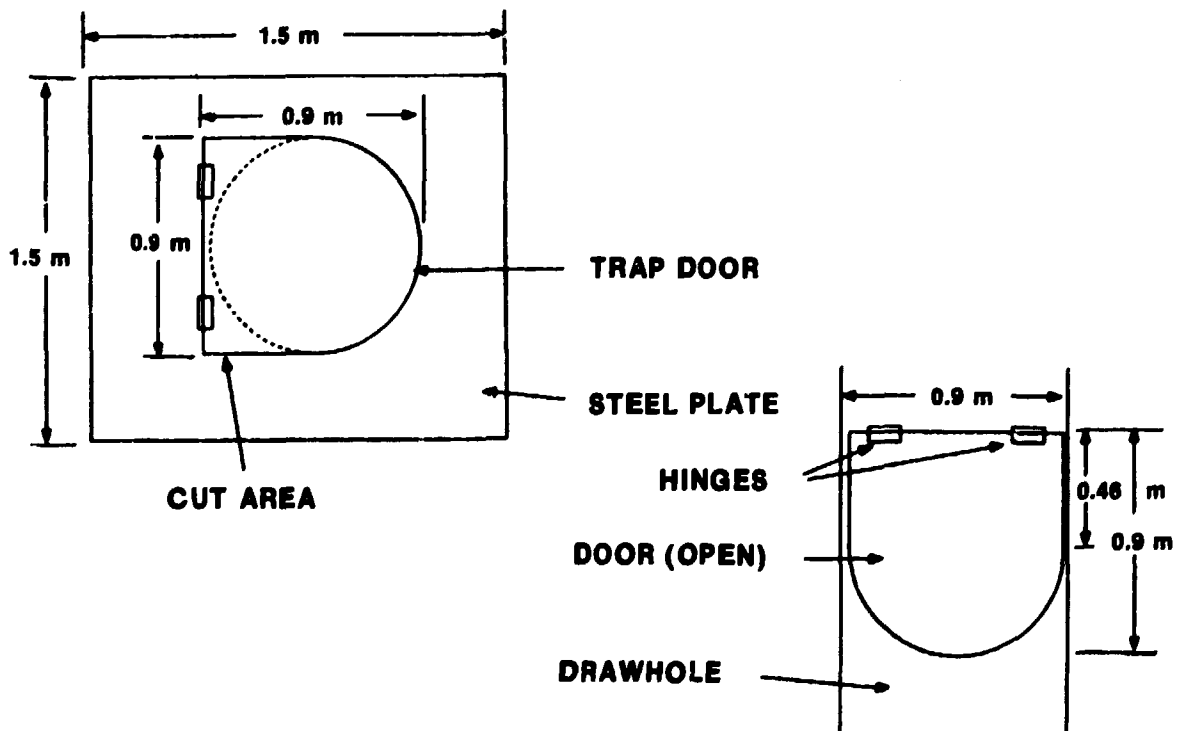


FIGURE 6. TRAPDOOR CONFIGURATION

## SUBSIDENCE EFFECTS ON BURIAL TRENCH COMPONENTS

Subsidence cavities measured on actual burial trenches vary widely in both size and shape and range from broad, shallow depressions to narrow pipes that may extend into the waste. Burial site surveys indicate that about 85% of the measured subsidence cavities are less than 2.75 m in diameter and 95% are less than 4.25 m in diameter. Subsidence effects on various components of SLB systems will be evaluated from field experiments at the EETF. The initial focus of these studies has centered on the effects of subsidence on bio-intrusion barrier system integrity. Subsidence cavities of five sizes ranging from none (control) through the maximum size observed at actual sites will be created beneath cobble-gravel cover systems as shown in Table 1. The experiments will be conducted in a trench 38 m long, 15 m wide, and 3 m deep. Beneath each plot a 0.9 m diameter hole will be augered into solid tuff to a depth necessary to equal the desired volume of the subsidence cavity (Table 1). A 1.5 m square steel plate containing a hinged trapdoor fastened by mechanical closures will be placed over each of the drawholes (Fig. 6). The drawhole configuration will allow the trapdoor to open fully. The entire trench will be backfilled to a depth of 2.2 m with screened (5 cm mesh) crushed tuff. To cause immediate subsidence, the uncompacted backfill will be used, allowing free gravity flow into the drawholes when trapdoors are released. The backfill will be covered with 0.9 m of cobble-gravel barrier material and 0.6 m of topsoil. A layer of cesium-chloride tracer will be placed at the backfill/barrier interface to indicate root penetration through the barrier under various degrees of subsidence (Fig. 7). Alfalfa will be planted on the surface of each plot.

When the drawhole closures are released, backfill will drain into the drawholes, causing subsidence at the surface. The subsided cavities will approximate right circular cones having slope angles of 35° to 40°.

Unavoidable slow subsidence of the entire trench surface will be observable throughout the duration of the experiment, resulting from continued compaction of the backfill. This secondary subsidence will be monitored by routinely surveying the elevation of the trench surface by means of marked rods positioned at the corners of the experimental plots.

Plant root penetration will be monitored by routine sampling of leaves. Cesium concentrations in the leaves will be measured as a function of time and location relative to the subsided cavities. Root penetration (if any) can be expected to occur first at the cavity rims, where maximum tensile stress will occur.

At the end of the experiments (FY84), the plots will be excavated to verify the tracer data. At the same time, both the upper and lower

TABLE 1. DIMENSIONS OF CAVITIES AND DRAWHOLES

| Desired Subsidence Cavities <sup>a</sup> |                 |                             | Required Drawhole Dimensions |              |
|------------------------------------------|-----------------|-----------------------------|------------------------------|--------------|
| Drawdown<br>(m)                          | Diameter<br>(m) | Volume<br>(m <sup>3</sup> ) | Diameter<br>(m)              | Depth<br>(m) |
| 0.75                                     | 2.4 - 2.9       | 1.5 - 2.22                  | 0.9                          | 3.3          |
| 1.0                                      | 3.6 - 4.4       | 5.3 - 76                    | 0.9                          | 11.5         |
| 1.25                                     | 1.8 - 2.1       | 0.6 - 0.9                   | 0.9                          | 1.35         |
| 1.5                                      | 3.0 - 3.6       | 2.9 - 4.2                   | 0.9                          | 6.4          |

<sup>a</sup>Slope of cavity sidewalls is 35 to 40°.

TABLE 2. MOISTURE CYCLING EXPERIMENT

The purpose is to determine whether water movement back to the surface of a low-level radioactive waste disposal pit caused by the drying of the surface in an arid environment is a possible pathway for transporting contaminants to the surface of the facility.

#### Experiment Variables

|                          |                       |
|--------------------------|-----------------------|
| Tracer depth             | 30 and 60 cm          |
| Average moisture content | 25 and 50% saturation |
|                          | 10 and 20 vol% water  |

#### Presence or absence of plants

#### Measurements

|               |                                 |
|---------------|---------------------------------|
| Water content | Neutron moisture probe          |
| Temperature   | Copper Constantan thermocouples |

#### Analysis of soil profile at the end of the experiment

#### Tracers Used

|              |                            |
|--------------|----------------------------|
| Liquid phase | Stable Cs, Sr, and Co      |
| Vapor phase  | Tritium as tritiated water |

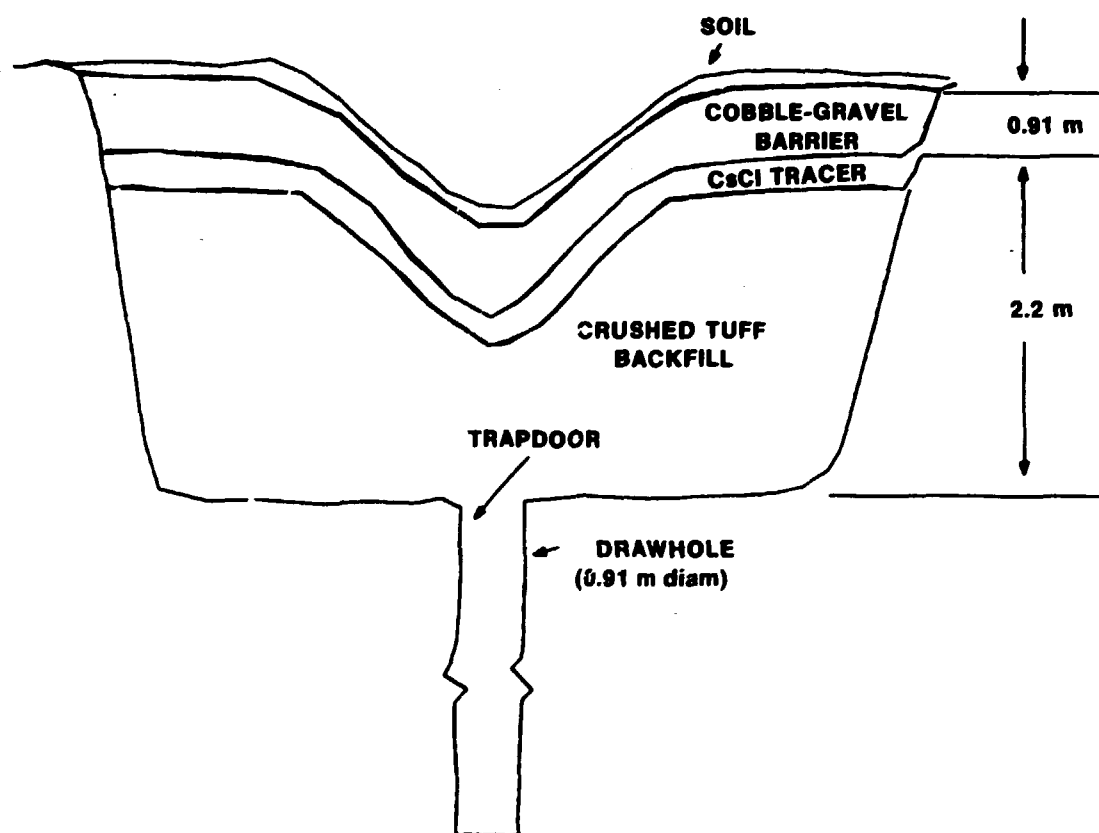


**FIGURE 7. SUBSIDENCE TESTING OF BIOLOGICAL INTRUSION BARRIERS**

surfaces of the bio-barrier will be mapped to determine the physical effects of subsidence on the barrier and to correlate with the tracer data and root measurements.

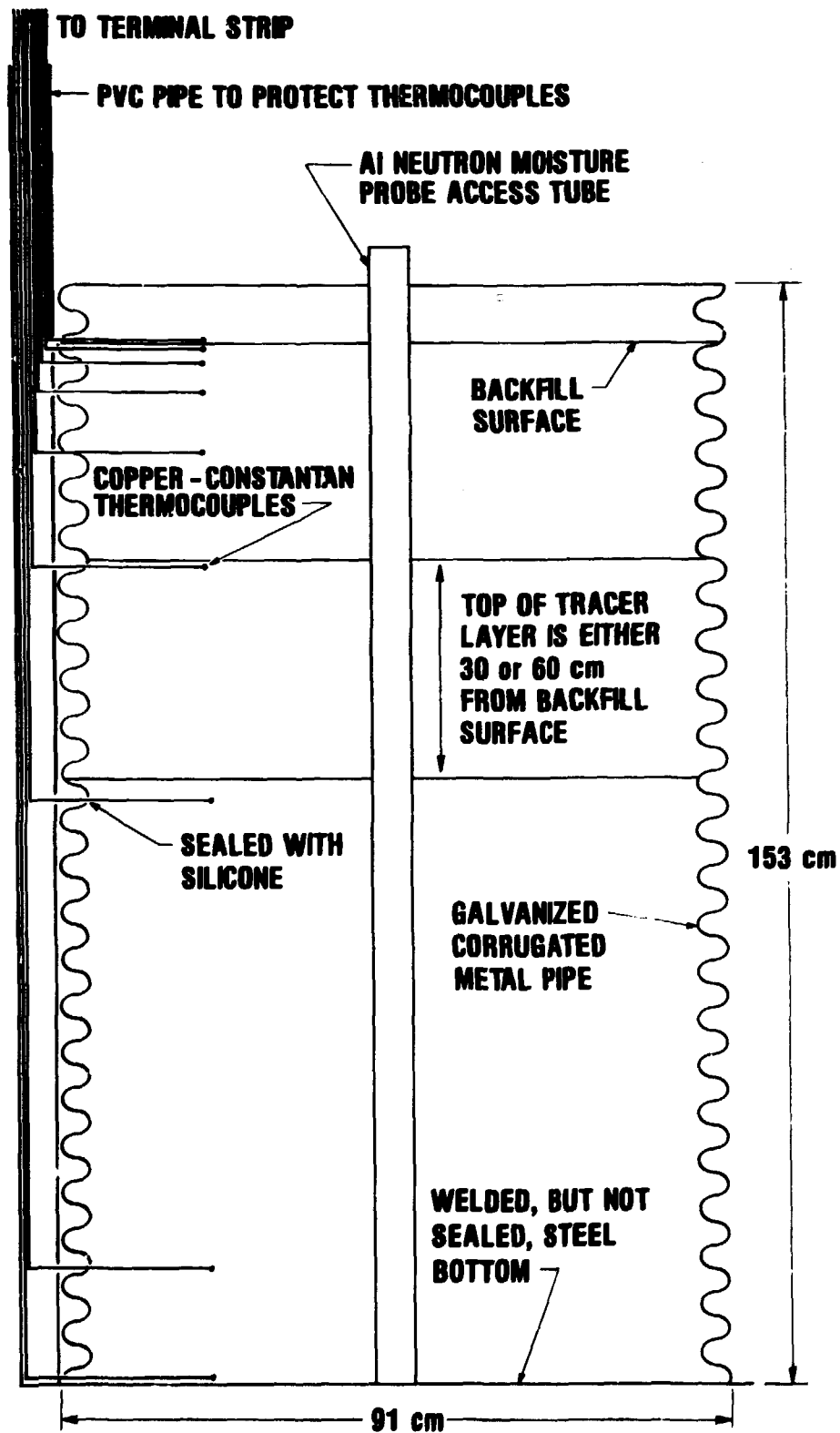
#### MOISTURE CYCLING EFFECTS ON CHEMICAL TRANSPORT

A process about which little is known is the effect of capillary forces created by evaporation of water at the soil surface in drawing soil moisture and soluble chemicals to the surface of a low-level waste site. A study was initiated to evaluate the importance of the process as a radionuclide transport mechanism and to determine the relationships of this transport to soil water status, soil temperature, and the presence or absence of vegetation on the soil surface. Information was also gathered on the relative importance of evaporation versus evapotranspiration (Fig. 1) in soil water behavior.

Sixteen soil columns (0.91 m diameter by 1.5 m deep) were constructed in metal culverts that were filled with screened (0.6 cm mesh) crushed tuff according to the experimental design shown in Fig. 8. Experimental variables were initial soil moisture status, simulated waste burial depth, and the presence or absence of plants (Table 2). Stable isotopes of cesium, strontium, and cobalt were used to monitor liquid phase transport while tritiated water was used to evaluate vapor phase transport.

Measurements of soil moisture within each profile were made through time with a neutron probe. Soil temperature measurements were routinely made at various depths in each profile (Fig. 9) with copper-constantan thermocouples. Both natural and supplemental precipitation were used in watering the soil columns. The vegetated plots were seeded to barley, alfalfa and yellow sweet clover. At the conclusion of the experiment (late FY82), horizontal soil cores will be taken at various depths in each profile to evaluate vertical transport of the tracers as a function of experimental variables. Average soil moisture in two of the 16 soil columns as a function of time are plotted in (Fig. 10), along with monthly water additions. The upper curve represents average soil moisture in an unvegetated plot; the lower curve is the corresponding data for the vegetated plot. A total of about 90 cm of water was applied to each column during the one year period represented in (Fig. 10).

Soil moisture (% by volume) as a function of time increased by about a factor of two in the non-vegetated soil column. The implication of these data is that evaporation at the soil surface (Fig. 1) does not have the capability of preventing soil moisture accumulation within the soil profile at the watering levels used in

**FIGURE 8. MOISTURE CYCLING EXPERIMENTAL UNIT**

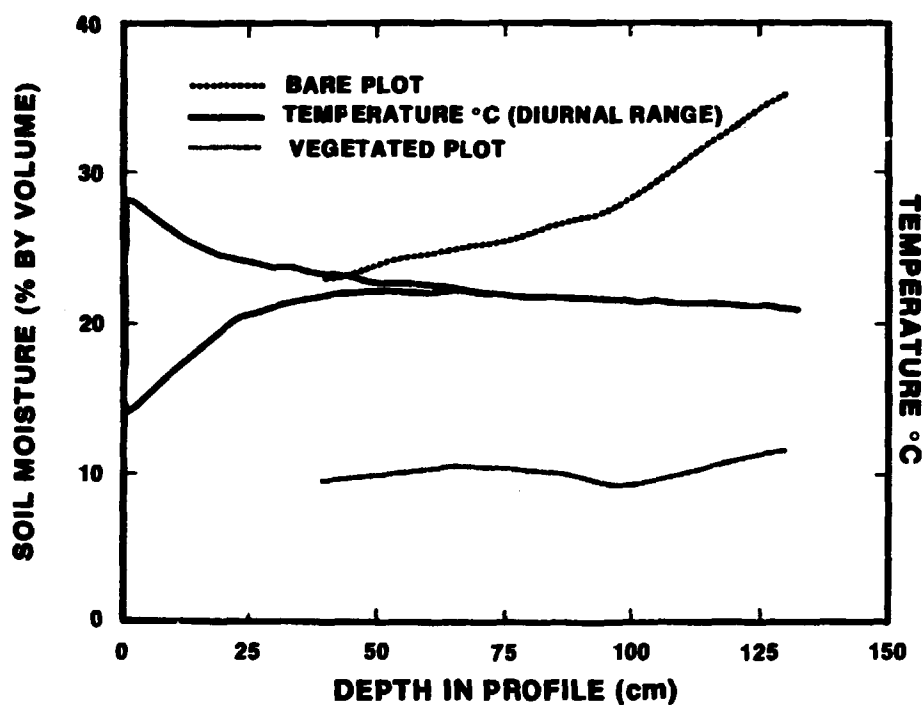


FIGURE 9. SOIL MOISTURE ON 7/20/82

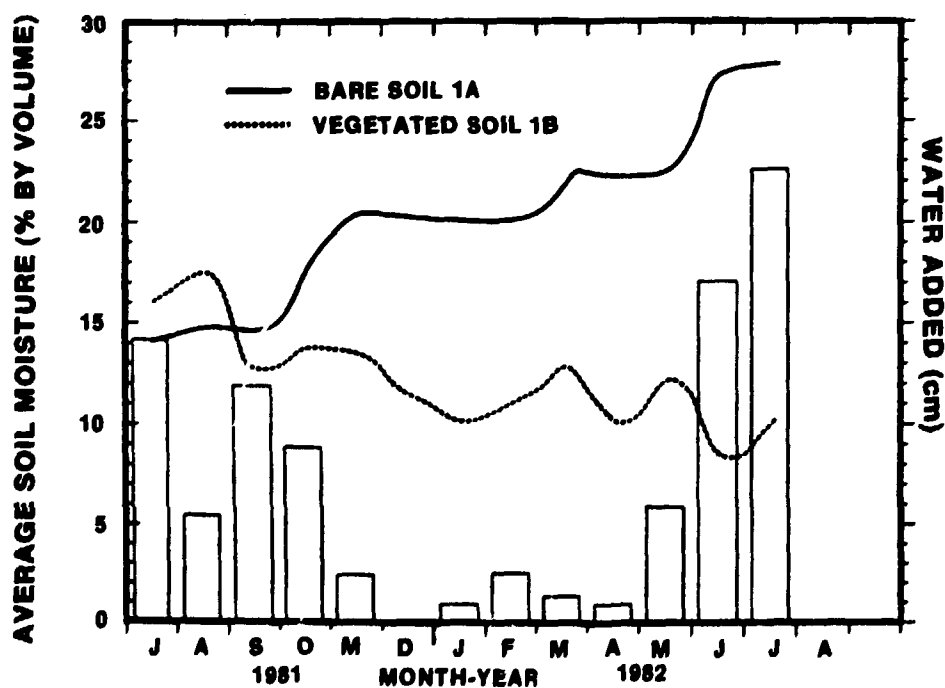


FIGURE 10. BEHAVIOR OF WATER IN BARE AND VEGETATED SOIL PROFILES

this experiment. In fact, the soil moisture data as a function of depth in the profile (Fig. 9) suggest that evaporation only affects soil moisture to a maximum depth of 40 cm. Soil temperature data support that contention (Fig. 9) in that the major variation (based on early morning and mid-afternoon measurements in June) in soil temperature occurs within 25 cm of the soil surface. Soil moisture at about 36% (by volume) as measured 130 cm deep in the soil column (Fig. 9) suggests that the lower layers of the unvegetated soil column are approaching or are at saturation.

In contrast, the vegetated surface not only has effectively transpired all of the added water (90 cm) from the soil column, but has also removed most of the initial soil water present at the start of the experiment (Fig. 10). Soil moisture as a function of depth in the vegetated profile was relatively constant and low at 10% by volume water.

The addition of water to the non-vegetated and vegetated soil columns is clearly reflected by increases in soil moisture. In the unvegetated column, addition of precipitation caused corresponding increases in soil moisture with little or no losses between subsequent water additions. In the vegetated soil column, addition of water was reflected by the soil moisture data but only for short periods of time because of the capability of transpiration to remove these additions.

The soil moisture data from this experiment clearly demonstrate the advantages of vegetation in controlling soil moisture and thus preventing accumulation of water in the soil profile. To select a vegetation cover that maximizes transpiration losses of water to the atmosphere, information is needed on the water use efficiency of native plant species that are seeded or that invade low-level waste sites.

#### SOIL EROSION CONTROL TECHNOLOGY

A particularly important aspect of erosion control methods is that they are often effective because they enhance infiltration rates and reduce surface runoff. Although this may reduce erosion, it can cause additional problems with moisture seepage or percolation through the trench cover profile. Research is needed to quantify the interaction between erosion control technologies and water balance in the soil profile.

Results of the modeling and experiments in this subtask will provide experimental data on erosion control technologies suitable for arid SLB sites, as well as information on the effect of erosion control technologies on subsurface components of the water balance. The results obtained in these experiments will permit comparison of these results with the results obtained in similar experiments by the USDA and will provide the interface to use the data obtained from agricultural systems for application to SLB. It will also extend the hydrologic model (CREAMS), to cover the unusual surface treatments



proposed to cover closed-out disposal sites.

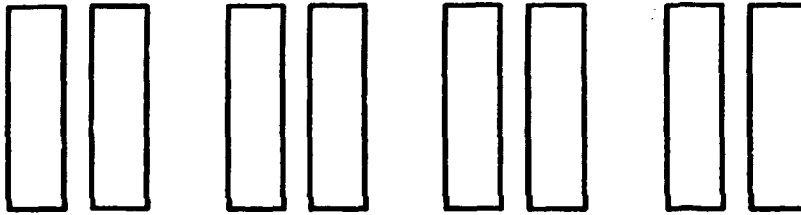
Both the CREAMS (3) modeling and field research efforts at Los Alamos will center around successfully determining values for the soil erodibility factor and the cover management factor of the Universal Soil Loss Equation (USLE), as well as determining water balance relationships for the trench cap environment. The soil erodibility factor of the USLE is an experimentally determined value for a particular soil whose value is influenced by properties such as soil texture, organic matter content, soil structure, and permeability. The cover management factor of the USLE is influenced by plant canopy, plant residue mulch and tillage, and waste management practices affecting soil porosity, roughness, compaction, and microtopography. The water balance relationships are not only important for a successful prediction of soil loss rates from the trench cap, since there is such a strong correlation between soil erosion rates and soil runoff rates, but are also used to predict percolation of water through the trench cap and surface losses of water through evaporation and plant transpiration.

Four pairs of 10 by 35 ft erosion plots have been constructed at the EETF where soil loss rates have been measured during experimental runs with a USDA rotating boom rain simulator. Soil erodibility factors will be determined on four of these plots for bare soil covers with standard Los Alamos SLB configurations (six inches of topsoil over about thirty inches of backfill) and with disked surfaces (Fig. 11). The other four plots will be used to determine the cover management factor for a 25 and 75% vegetative cover of barley.

Water balance relationships will be determined for all 8 plots using a neutron moisture probe at various soil depths in 3 locations per erosion plot. The amount of precipitation (applied at a rate of 2.5 inches/hour) was measured during each 1-hour simulator run, as well as the total amount of runoff and sediment yield. Evapotranspiration will be calculated by solving the water balance equation using experimental data to estimate all the remaining variables in the equation. Field measurements of leaf area index for the barley cover will also be made to validate CREAMS estimation of water losses caused by plant transpiration.

In FY83 4 new pairs of 10 by 35 ft erosion plots will be emplaced at the EETF and 2 of the FY82 plot pairs will be reused. The four vegetated plots that were used in FY82 will be reused to determine the cover management factor for gravel and riprap erosion control treatments (Fig. 12). The most promising biobarrier or wick system, as determined by other Los Alamos experiments, will be included in the four new erosion plots to determine a soil erodibility and cover management factor for CREAMS. The remaining 4 new plots will be used to determine the cover management factor for natural range grass cover on a trench cover to compare with similar data collected in FY82 on barley covers.

**1. Emplace 4 pairs of 35 by 10 ft erosion plots:**



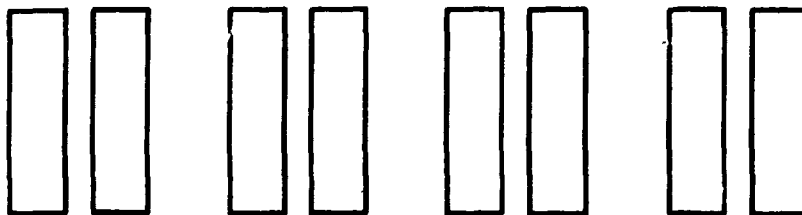
**2. Plot treatments:**

**A. 4 plots to determine soil erodibility factor for bare soil covers:**  
 2 plots: standard SLB configuration  
 2 plots: tilled surface

**B. 4 plots to determine cover management factor:**  
 2 plots: sparse barley cover  
 2 plots: dense barley cover

**FIGURE 11. FY82 EROSION FIELD EXPERIMENTS**

**1. Emplace 4 pairs of 35 by 10 ft erosion plots**



**2. Plot treatments:**

**A. 4 plots to determine soil erodibility and cover management factors for profiles containing biobarriers or wicks**

**B. 4 plots to determine the cover management factor for natural range grass cover to compare with the FY82 barley cover data**

**C. 4 plots to determine the cover management factor for FY82 plots with gravel-riprap added**

**FIGURE 12. FY83 EROSION FIELD EXPERIMENTS**

The data generated on soil erosion, water balance relationships, and CREAMS parameters at Los Alamos, Boise, Idaho and Tombstone, Arizona will be used by research groups in the USDA and within the LLWMP. This will allow the LLWMP to utilize a very broad data base from the agricultural research community, and result in a more generic solution to problems encountered in the design of shallow land repositories located throughout the arid and semiarid areas of the western U.S.

#### ACKNOWLEDGEMENTS

This work was supported by the Department of Energy under contract W-7503-ENG-3 with Los Alamos National Laboratory. The assistance of K.V. Bostick, E.M. Karlen, J.L. Martinez, W. Herrera, E. Lopez, B. Drennon, G. Trujillo, G.C. White, E. Montoya, and J.G. Steger is greatly appreciated.

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## A BRIEF OVERVIEW OF REMEDIAL ACTIONS NECESSARY AT MAXEY FLATS

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### ABSTRACT

This presentation addresses certain remedial measures which the Kentucky Natural Resources and Environmental Protection Cabinet has initiated at the Maxey Flats Nuclear Disposal Site to prepare the facility for stabilization and eventual closure.

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### INTRODUCTION

The Kentucky Natural Resources and Environmental Protection Cabinet (NREPC) has been asked to participate in the DOE's Low Level Waste Planning Symposium, and the Corrective Measures Session in particular, as a result of its activities related to the Maxey Flats Nuclear Disposal Site (MFDS). We hope to provide you with a State's perspective, experience, and some potential generic applications for remediation at a low level waste disposal facility. As you may not be aware, Kentucky is the site of a nuclear waste disposal facility which is closed to the receipt and disposal of commercial nuclear waste; however, the facility has not yet been stabilized or decommissioned. For the purposes of this presentation, a "decommissioned facility" is meant to imply a facility that requires essentially no outside intervention or maintenance in order to retain its integrity.

### GENERAL BACKGROUND INFORMATION

In 1962, the federal government halted the receipt of commercial shipments of nuclear waste at its facilities for disposal purposes due to the availability of two commercial facilities to dispose of such waste. The first commercial facility to be licensed and operational was located at Beatty, Nevada (1962). The second commercial facility was Maxey Flats, Kentucky, which was licensed in the fall of 1962. Additional disposal sites opened at West Valley, New York (1963); Richland, Washing-

ton (1965); Sheffield, Illinois (1967); the last facility to open was Barmwell, South Carolina, in 1971. Of these six commercial sites, only three remain open for the shallow land disposal of nuclear waste, i.e., Beatty, Richland and Barmwell (1). At the time of this presentation, Beatty is presently closed for an indeterminate amount of time.

The land on which the Maxey Flats facility is owned by the Commonwealth of Kentucky, is required by the federal regulations (10 CFR Part 20(302b)). The facility, located on nearly three hundred acres in Fleming County in eastern Kentucky, was originally owned by the Nuclear Engineering Company (NECO). It was operated under the auspices of the Kentucky Cabinet for Human Resources (KCHR), the agency responsible for licensing the use, possession, and disposal of radioactive material in the state. Maxey Flats was operational in the sense that commercial nuclear waste was received and buried from the spring of 1963 until December, 1977. In May, 1978, the State of Kentucky signed an agreement with NECO to purchase the leasehold estate and rights for the operation of the site. Since July, 1979, the license for facility operations has been held by the NREPC.

#### GEOGRAPHY AND GEOLOGY

The Maxey Flats Disposal Site is located in Fleming County, Kentucky, approximately 10 miles northwest of the city of Morehead. The actual facility is located on an isolated plateau in the knobs region of northeastern Kentucky, characterized by the presence of many knob-like erosional remnants of Mississippian and Late Devonian era rocks. These rocks generally consist of clay-shale, siltstone, fine grained sandstone, and fissile carbonaceous shales. Maxey Flats is located on one of the finger-like projections extending from this plateau (2).

The Commonwealth owns about 280 acres of land at Maxey Flats. About 45 acres are designated as a restricted access area (controlled by NREPC for purposes of radiation protection). Within the restricted area, about 25 acres have been used for disposal of nuclear wastes into trenches; the remainder of the restricted area primarily issued for operations and maintenance facilities.

The climate at Maxey Flats is humid continental generally characterized by warm, humid summers and cold winters. The mean annual precipitation is approximately 46 inches. For the years 1978 through 1981, the annual precipitation has ranged from 33 to 51 inches. The precipitation is not evenly distributed throughout the year. The driest months are usually during late summer and autumn, and the wettest months during spring and early summer.

## BRIEF DISPOSAL OPERATIONS HISTORY

At Maxey Flats, radioactive waste was disposed of in excavated trenches approximately 60 feet wide, 30 feet deep, and 100 to 300 feet in length, and the waste dumped into the open trench. The soils excavated from the trench were used as backfill to lower radiation exposures to acceptable levels and to fill the voids formed by this method of waste emplacement. Sometimes additional backfill was mounded over the trench area and then compacted to promote surface water runoff. Shallow rooted vegetation was then planted to inhibit erosion. Figure 1 is a general schematic of the site with recent improvements as elaborated on later in the presentation.

The first disposal of radioactive waste occurred in the spring of 1963 and the last commercial waste accepted for disposal was in December, 1977. During this interval, about 4.75 million cubic feet of nuclear waste was disposed of by the technique of shallow land burial. This volume of waste has been estimated to contain about 2.4 million curies of byproduct material, about 241,769 kg of source material, of 431 kg of special nuclear material, and in excess of 63 kg of plutonium.

Low-activity wastes disposed of at the Maxey Flats facility included such miscellaneous materials as paper, trash, clothing, protective apparel, laboratory glassware, obsolete equipment, radiopharmaceuticals, and miscellaneous rubble. Higher activity wastes included solidified liquids, shielding accessories, filters, ion-exchange resins, and activated metals. Transuranic waste can be found in glove boxes, rubber tubing, gaskets, plastics, paper and rags. These wastes arrived in packages ranging from cardboard boxes, wooden crates, and metal drums to concrete blocks. When the facility was operational, the wastes were considered to be acceptable for disposal as long as they were a solid and the packaging met U.S. Department of Transportation transport regulations. Thus, the less hazardous material tended to arrive in fiberboard boxes or wooden crates and the more hazardous material in shielded drums or casks, or concrete structures (3).

In the mid 1970's, several studies were performed on facility operations, and from these studies several recommendations were made concerning methods to improve operation of the water management program (i.e., trench leachate pumping, analysis, and volume reduction by an evaporator system constructed in the early 1970's). All these activities were carried out in order to minimize the potential for transport of radioactivity (4). In response to those recommendations, KCHR required the site operator to continue to remove water from trenches in order to minimize the potential for transport of radioactivity and to bring and maintain the trenches in a "dry status." Water collecting in the trenches appeared to principally result from surface infiltration of rainfall rather than from groundwater movement. Improvements in operations undertaken at the facility were to reduce the likelihood of water contacting the buried waste and have included grading and improving surface drainage,

recapping older trenches to reduce cap permeability, improving procedures for capping new trenches, establishing a vegetation cover over completed trenches, eliminating on-site ponds, installing additional sumps in trenches to facilitate liquid removal, routine backfilling of waste as it was placed in the trenches, and removing precipitation from trenches as the trenches were being filled. These efforts, plus the removal of several areas of surface contamination, appear to have been effective in reducing the release of radioactivity from the site, and radioactivity levels detectable in the off-site environment has decreased according to KCHR.

The facility was closed on December 27, 1977, by order of KCHR until completion of a water management program, and the completion of further studies to determine the long-term use and safety of the site.

Following negotiations with NECO, Kentucky signed an agreement with NECO in May, 1978, whereby the leasehold estate and rights to the site were purchased by the state.

In May, 1979, NREPC was given the mission by the state legislature to stabilize the facility and eventually decommission the facility in accordance with state and federal requirements. A fee was charged for disposal of radioactive waste at the site in order to provide long-term funding for perpetual care of the site after closure. Unfortunately, the Maxey Flats perpetual care and maintenance fund contains inadequate monies to pay for post-closure care, maintenance, and water management activities. This fund only contains about \$300,000. NREPC has been spending about \$1,200,000 annually to stabilize the facility for the past three years and this level of funding is expected to continue at least for the next two to four years. This money has been provided solely by the taxpayers of Kentucky.

#### CURRENT REMEDIAL ACTIVITIES

At the time the NREPC assumed control of Maxey Flats, the facility had two above ground ponds containing about 1 3/4 million gallons of slightly contaminated trench liquids. The small one was originally open to rainfall, but was provided with a plastic cover in the summer of 1979 to control its increasing volume of contaminated liquid. The pond contained about 500,000 gallons of liquid which has been used as low tritium dilution liquid in the evaporator system. The pond liquid was finally processed in June, 1981. The pond side walls were pushed in and the pond bottom brought up to present grade level. The area was then recontoured to enhance drainage and reseeded for erosion control.

The large pond originally contained about 1,250,000 gallons of liquid which has also been used as low tritium dilution liquid in the evaporator system. The large pond contents were finally processed in late spring, 1982. A fate similar to the small pond was in store for the large pond.

An old tank farm and associated open pond existed on site for about ten years. Originally, the tanks were used to hold the evaporator concentrates. These concentrates were reprocessed and consolidated in a covered tank building. After inspection of the empty tanks, it was determined that their useful life had come to a peaceful end. A decision was made to dismantle and bury the tanks. The accompanying pond liquid was finally processed and the pond filled in with clean soil left over from the small pond walls and the pond bottom brought up to grade level. The area was contoured to enhance drainage and reseeded for erosion control.

Eventually, NREPC has been able to get ahead of the overflowing ponds, the poor drainage, surface erosion, and the rapid infiltration of surface runoff into the trenches. These remedial actions allowed us the opportunity to look at the site in an objective manner, and it was estimated that an additional 600,000 gallons of surface runoff was still infiltrating into the trenches on an annual basis. Prior to the implementation of our remedial measures, it was estimated that two million gallons of surface runoff was infiltrating trenches, entering uncovered ponds, and entering open trenches on an annual basis. However, NREPC was only able to treat about 1.2 million gallons annually through our submerged combustion evaporator system used as a volume reduction device.

A decision was made to identify those trenches which have consistently been high leachate producers or possess high tritium activity. Thus, using a combination of 10 and 15 mil PVC, about 600,000 square feet of the trench surface has been covered with the plastic. These trenches are located in the middle and western section of the trench area. It is recognized that this is only a short term "solution" to the problem, but during its lifetime the cover would prevent a substantial amount of surface runoff from infiltrating into the trenches. In fact, continuous trench liquid level recorders indicate that no observable increase in trench liquid level has occurred as a result of rainfall over the previous ten (10) months. During this interim, a number of different techniques for "final" trench cap sealing could be demonstrated. It is hoped that during this interim a longer term "solution" could be demonstrated and the implementation of such begun before total deterioration of the present surface cover.

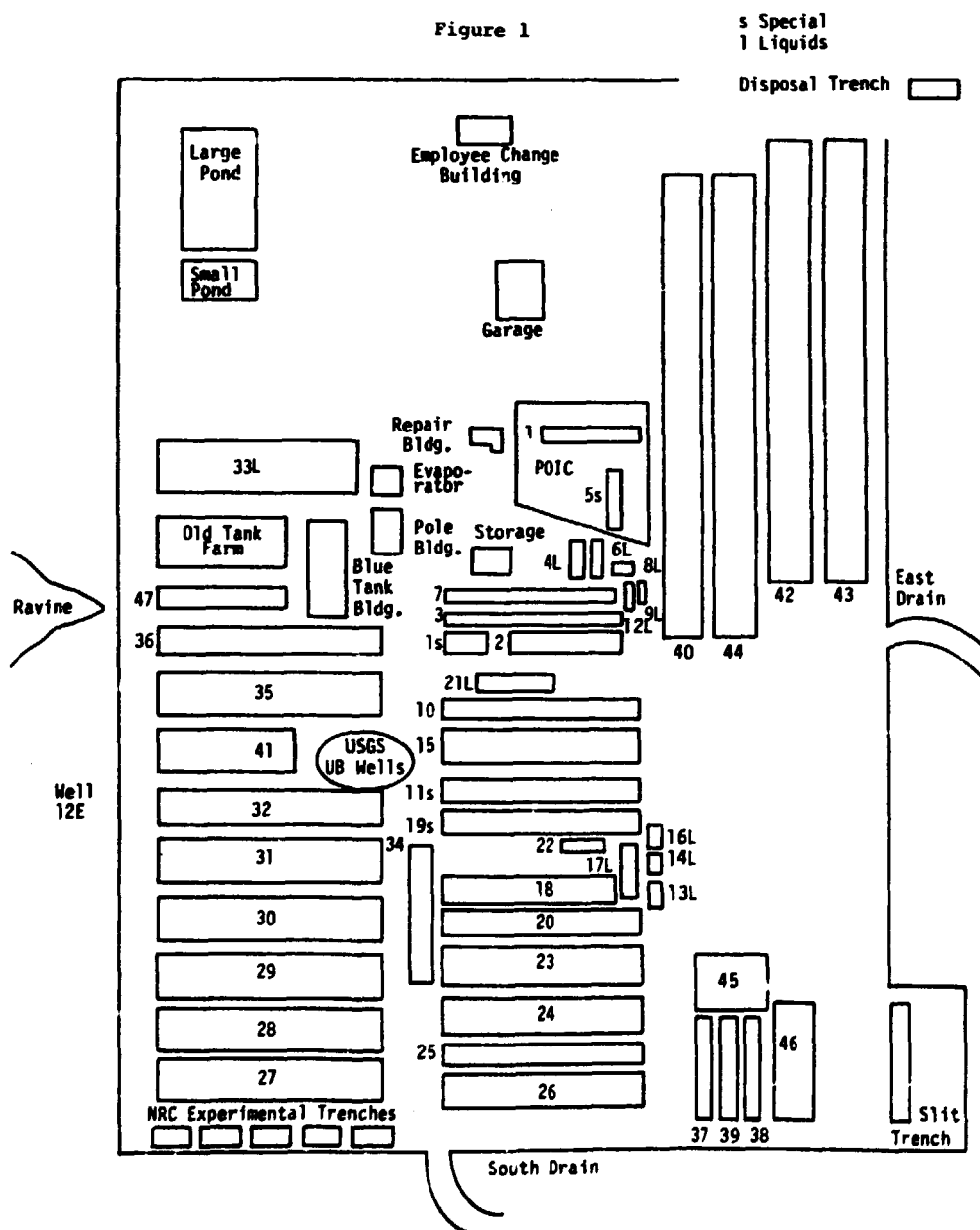
#### THE PRIORITIES

In the time which this interim solution bought, we have developed three priorities which we feel must be addressed and solved before stabilization and closure can occur at a facility such as Maxey Flats. These priorities are: trench cover rehabilitation; tritium immobilization; and leachate characterization, solidification and transport. Each of these topics will be discussed briefly in the following paragraphs.



General Schematic of the Maxey Flats Site

Figure 1



Subsidence is a problem with the trenches. The specific problem does take many forms and all will have to be addressed in a complete trench cover rehabilitation remediation program. First the older trenches exhibit general area subsidence where the entire trench simply settles with time. Without an impermeable cover over the trench and if proper drainage is not maintained, the trench cover can actually function as a pond with the water infiltrating into the trench. The newer trenches also exhibit a catastrophic subsidence where large areas may subside a foot or more in depth and become obvious sources of water ponding. Over time, most subsidence was thought to be complete in 15-20 years; however, recent information indicates this is only a short pause in the subsidence occurrence curve and the problem has a substantially longer time frame before the trenches could be considered approaching a stable configuration. A third problem is sidewall separation where the trench and its contents physically separate from the original trench wall. Separations have been observed encompassing almost the entire length of trenches. During rains, these separations provide a funnel for runoff to infiltrate into a trench. The original wasteform has obviously contributed to this phenomenon as waste was received in packages ranging from cardboard boxes, wooden crates, and metal drums to concrete cylinders.

Tritium immobilization is a high priority with Maxey Flats operations as tritiated liquids are the controlling limiting factor for evaporator operations. We have traditionally processed liquids containing 2-3 uCi/ml of tritium with a license limit of 7 uCi/ml. However, at this time there are substantial quantities of trench leachate with tritium concentrations ranging from 20 to 100 uCi/ml with very little dilution liquids left with which to process the high tritium liquids. According to shipment records, about 850,000 curies of tritium were disposed of at the site with about 640,000 curies of tritium being received in one shipment from a federal laboratory.

A large amount of work aimed at defining leachate characterization is being funded by the Nuclear Regulatory Commission (NRC). This work is being coordinated by Pacific Northwest Laboratories (PNL) and involves the University of California-Berkeley, Los Alamos National Lab, and Brookhaven National Lab in addition to numerous PNL researchers. Two NRC publications have been produced as a result of this work, and the reader is encouraged to review these documents identified as NUREG/CR-1832 and NUREG/CR-2383.

#### PROOF OF INSTALLATION CONCEPT

As a prelude to actual construction of a proposed multilayer trench cover by an outside contractor, a Proof of Installation Concept (POIC) plan was developed to simulate actual construction of a multilayer trench cover. The POIC was to be implemented by our present operations contractor, Hittman Nuclear and Development Corporation, as they are

experienced in working with radioactive material and contaminated materials. A plot of about  $\frac{1}{2}$  acre was chosen encompassing Trenches 1 and 5S as a test of POIC.

Within the top 8 inches of soil, several 55 gallon drums were exposed with the heavy construction equipment being utilized. A B-3 solidified resin cask was unearthed including the lead sheeting utilized to lower the radiation levels to acceptable limits. Several drum tops were cut upon trying to go deeper than a foot. So the initial attempt to remove 18 inches of soil from the trench covers was abandoned.

In addition to the uncovered waste, several areas of subsidence were uncovered. Both a small hole (about 1 foot in diameter) and a rather long subsidence (about 18 inches wide and 15 feet long) were unearthed early in the project. The waste revealed by these subsidence area indicated that the metallic waste was in excellent condition, with mainly the paint missing. Hence, one item to be considered for long term trench stabilization is the inevitable subsidence which will result from the degrading of the metallic waste drums. Whether this will occur in an additional 20, 50, or 100 years is undetermined at the moment. Trench 1 was closed in May, 1963, and Trench 5S was closed in April, 1964. Thus, the prevalent idea of 15-20 years of subsidence is not totally correct. There will be another major component of subsidence which will have to be taken into account in the design of a long-term multilayer trench cover.

#### TRENCH STABILIZATION

In a project effort with the assistance of Oak Ridge National Laboratory (ORNL) approved by the U.S. Department of Energy, a study of the feasibility of using an injected grout to fill void spaces in a trench will be attempted. For purposes of the initial demonstration, a 30 foot long trench section will be utilized as a test volume. The trench to be used is about 25 feet wide and about 25 feet deep. An estimate of available void volume will be made based on liquid draw down data. That amount of grout will be injected into the trench section or until the trench top cover exhibits surface rise. We plan to have the sheet pilings, injection wells, observation wells, liquid draw down sump, and void volume estimates completed before winter and the actual injection to occur in the spring of 1983. At this time, it is thought that grout permeability and grout core removal tests will provide adequate data to determine if the methodology is technically feasible and economically viable.

## FUTURE DIRECTION

As a result of the success we have had with a short-term plastic cover, the Cabinet is proceeding to prepare the necessary paperwork to install a long-lived (about 20 years) plastic membrane to cover the trench areas where possible. Due to the success which the present cover has kept rainfall out of the trenches, it is thought that this long-term cover will provide us the necessary time to evaluate a number of multi-layer wick-effect trench covers over time and proceed with final stabilization of the site. This technique should allow us the time to obtain the necessary information on how final closure will have to be accomplished, i.e., barrier walls, grouting for stabilization, subsidence control, etc.

## SUMMARY

One point we have tried to convey in this short presentation is that NREPC is actively involved in ensuring continued protection of the health and safety of the public, and in establishing the site to be stabilized, and finally placing the site into a decommissioned status. We hope that by illustrating the operational experience of NREPC at Maxey Flats, other agencies may benefit and that future disposal practices may be altered to alleviate some of the conditions which are continuing to cause problems at Maxey Flats.

All of the above described actions implemented by NREPC have permitted us to have sufficient time to evaluate a number of remedial actions and determine their effectiveness. In addition, the obtaining of federal support, direct or indirect, has allowed this Cabinet to stretch its limited budget dollars. All of these mechanisms have provided this Cabinet with the time to prepare a plan of action, remediation, and stabilization. All of these efforts leading, in time, to the closure of Maxey Flats.

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**LAND DISPOSAL ALTERNATIVES FOR LOW-LEVEL WASTE\***

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**1.0 INTRODUCTION**

The Environmental Protection Agency is faced with developing environmental standards for the disposal of large volumes of low-level nuclear waste. Commercial low-level waste generally consists of radioactive wastes (excluding mill tails), reactor wastes, wastes from universities and medical facilities, and some wastes resulting from the decommissioning of nuclear facilities.

The objective of this project is to develop data regarding the effectiveness and costs of various options for disposing of specific low-level nuclear waste streams. This will be accomplished through the following steps:

- o Characterize the properties of the commercial low-level wastes requiring disposal.
- o Evaluate the various options for disposing of this waste.
  - Characterize selected representative waste disposal sites.
  - Design storage facilities suitable for use at those sites.
- o Calculate the effects of various waste disposal options on population health risks.
- o Estimate the costs of various waste disposal options for specific sites.
- o Perform trade-off analyses of the benefits of various waste disposal options against the costs of implementing these options.

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\*This work is supported by the Environmental Protection Agency under contracts #68-01-6562 and 68-02-3174.

These steps are described in the following sections.

## 2.0 WASTE CHARACTERIZATION

Various processing options for the 36 waste streams described in the Draft Environmental Impact Statement of 10 CFR Part 61 "Licensing Requirements for Land Disposal of Radioactive Waste" (NUREG-0782) are being evaluated as to their radiological, chemical and physical properties. The radiological data consists of the Ci/m<sup>3</sup> by nuclide for each waste stream and processing option. The chemical and physical properties include container failure rates, nuclide leach rates, density, porosity, ease in handling and shipping, etc. Emphasis will be placed on defining those parameters that are required as inputs to the PRESTO risk model and to the cost/benefit analysis described in the next sections.

The processing options will be selected from the published literature. Current practice will be compared to compaction, incineration, solidification, and other processing and packaging techniques as they apply to each waste stream. Two waste processing and packaging options will be considered in detail, the current practice, and the practice recommended for future use. Each waste stream and applicable processing options will be evaluated against the appropriate waste form requirements for the eight disposal options: Shallow Land Burial (SLB), Improved Shallow Land Burial (ISLB), Sanitary Landfill (SLF), Engineered Surface Storage (ESS), Intermediate Depth Burial (IDB), Deep Geologic Burial (DGB), Deep Well Injection (DWI), and Hydrofracture (HF). The waste streams most compatible with each disposal option will be further evaluated in the risk and cost/benefit analyses.

## 3.0 DISPOSAL MODEL DESIGNS AND COSTS

Generic designs (models) are being developed for the low-level waste disposal methods shown in Figure 1. The key features of each disposal option are summarized in Table 1. The generic models represent the current state of thinking regarding size, construction and operation of such facilities. Each model will specify dimensions and materials for construction. In all cases sufficient details will be provided to allow accurate costing of the facility.

Each model will be accompanied by a description of normal disposal operations. The description of operations will identify any environmental risks during operations that may be peculiar to the disposal method and will allow accurate estimation of costs of operation.

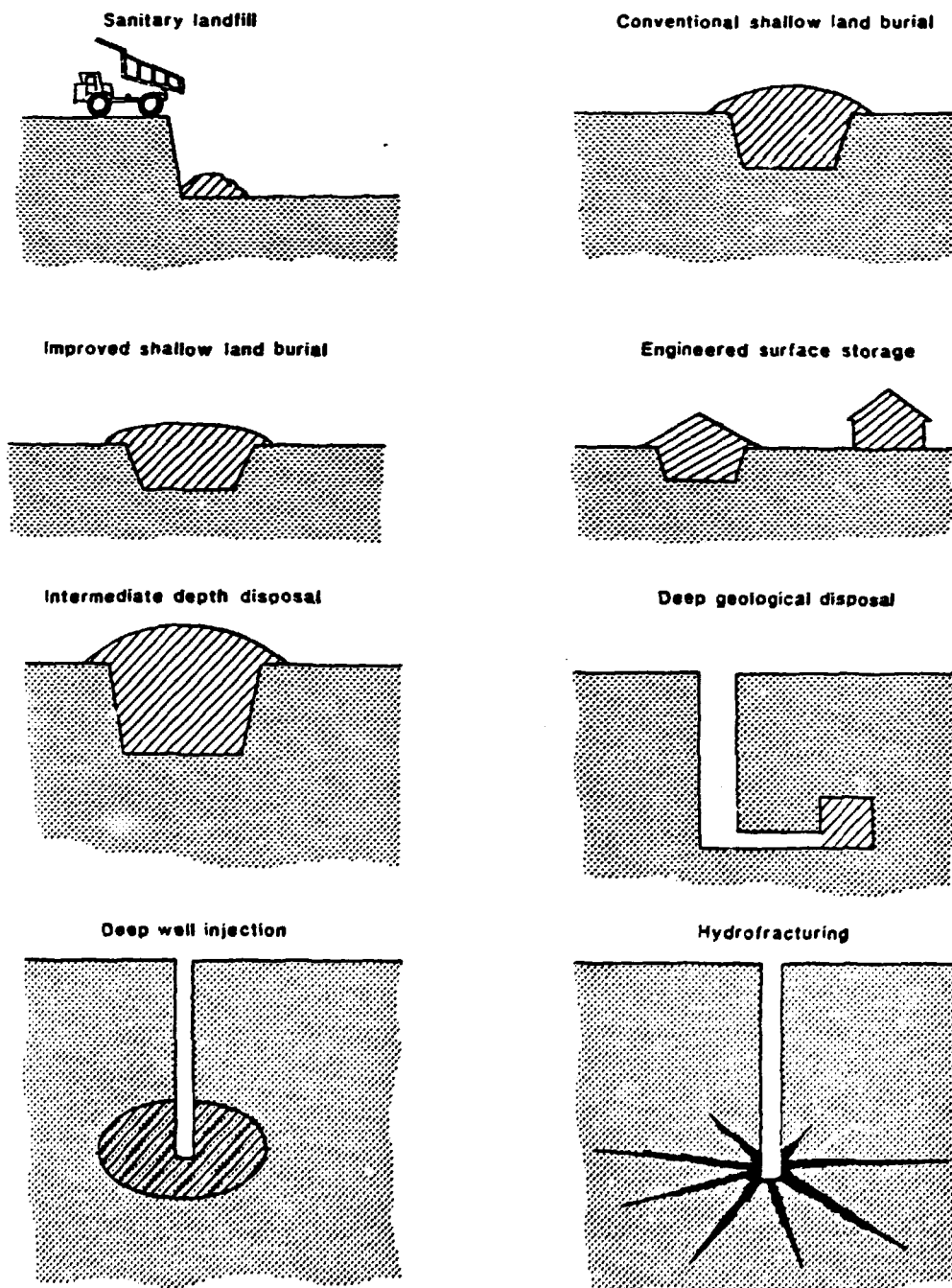


Figure 1. Low Level Waste Disposal Options



Table 1. Features of Various Low Level Waste Disposal Options

| Disposal Method                    | Distinguishing Features                                                                                                                                                                                                                                                                                          | Disposal Method                          | Distinguishing Features                                                                                                                                                                                                                                                                         |
|------------------------------------|------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------|------------------------------------------|-------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------|
| Sanitary Landfill                  | <ul style="list-style-type: none"> <li>• Waste may not be packaged</li> <li>• Waste laid on fill surface and covered with layers of dirt</li> <li>• No segregation from other biological or chemical waste forms</li> <li>• Optional site security*</li> </ul>                                                   | Intermediate Depth Disposal              | <ul style="list-style-type: none"> <li>• Similar to conventional or improved shallow disposal</li> <li>• Excavation is deeper</li> <li>• Backfill cover is thicker</li> <li>• Site secured*</li> </ul>                                                                                          |
| Conventional Shallow Land Disposal | <ul style="list-style-type: none"> <li>• Waste placed in excavations</li> <li>• Excavations backfilled with material excavated</li> <li>• Some segregation of wastes</li> <li>• Site secured*</li> </ul>                                                                                                         | Deep Geological Disposal in Mined Cavity | <ul style="list-style-type: none"> <li>• Deep excavated room entered through shaft or tunnel, may use an existing mine or cavity</li> <li>• Storage room and access backfilled following waste emplacement</li> <li>• Storage area may lie below an aquifer</li> <li>• Site secured*</li> </ul> |
| Improved Shallow Land Disposal     | <ul style="list-style-type: none"> <li>• Similar to conventional shallow disposal</li> <li>• Improved cover over excavation</li> <li>• Engineered features used to augment containment</li> <li>• Improved drainage</li> <li>• Some segregation of wastes</li> <li>• Site secured*</li> </ul>                    | Deep Well Injection                      | <ul style="list-style-type: none"> <li>• Waste injected into deep porous or permeable medium or into an unuseable aquifer</li> <li>• Injection rate may be low</li> <li>• Site secured*</li> </ul>                                                                                              |
| Engineered Surface Storage         | <ul style="list-style-type: none"> <li>• Waste storage totally enclosed with man made materials</li> <li>• Facility can be designed to accommodate high level short lived activity</li> <li>• Property is secured for limited time period</li> <li>• Waste subsequently disposed in sanitary landfill</li> </ul> | Hydrofracturing                          | <ul style="list-style-type: none"> <li>• Requires slurry waste form</li> <li>• Injection under pressure into suitable deep formation</li> <li>• Waste locked into formation by solidification</li> <li>• Site secured*</li> </ul>                                                               |

\* Administrative control is limited to a few hundred years.

#### 4.0 SITE CHARACTERIZATION

Hydrogeologic and climatic settings are being specified from the literature for the seven different disposal sites listed in Table 2. The settings include the parameters needed by the risk assessment model PRESTO. Typical parameters include depth to the top of the aquifer, aquifer productivity and velocity, distance to aquifer discharge, wind roses, annual rainfall and evaporation, soil permeability, sorption in soil and underground formations, and formation mechanical properties.

Deep disposal methods will require analysis of the regional and local hydrologic systems encompassing the facility area. Topics to be addressed are depth to and thickness of aquifers and confining layers, permeability and porosity of aquifers, location of recharge and discharge areas, and analyses of the hydrogeochemistry of ground water likely to be encountered by materials leached from the disposal area. PH, Eh, specific conductance, alkalinity, and chemical analyses of environmental waters will aid in predicting the path of released nuclides and whether or not natural attenuation or increased mobility can be expected.

#### 5.0 RISK ANALYSIS

Oak Ridge National Laboratory under contract to the EPA has developed a computer model for evaluating the risks associated with disposing of LLW in shallow land burial sites. This model, called PRESTO (Prediction of Radiation Effects from Shallow Trench Operations), simulates radionuclide migration from shallow trenches to an individual or population along the following pathways:

- o Groundwater transport
- o Overland flow
- o Direct exposure of wastes due to erosion
- o Resuspension
- o Atmospheric dispersion
- o Deposition onto crops and surfaces
- o Inhalation
- o Ingestion of beef, milk, crops and water.

This model is being adapted by TRW/Rogers and Associates to provide a similar analysis for other land disposal options. The radionuclide migration pathways from each of these additional disposal options are being defined in mathematical algorithms which will be incorporated into the PRESTO code. The individual and population health effects will be using calculated existing PRESTO subroutines. Other existing subroutines will be used where applicable.

The risk analysis will utilize the waste radiological, chemical, and physical properties developed by the waste characterization study for each of the 36 waste streams, the design parameters for each

Table 2. Disposal Settings to be Considered

|                                | Humid    |             | Arid   | Humid                              |                                     | Semi Arid |                         |
|--------------------------------|----------|-------------|--------|------------------------------------|-------------------------------------|-----------|-------------------------|
|                                | Sarmwell | West Valley | Beatty | Generic Low Permeability Formation | Generic High Permeability Formation | WIPP Salt | Generic Shale or Basalt |
| Engineered Surface Storage†    | /        | /           | /      |                                    |                                     |           |                         |
| Sanitary Landfill††            | /        | /           | /      |                                    |                                     |           |                         |
| Convention Shallow Land Burial | ②        | /           | /      |                                    |                                     |           |                         |
| Improved Shallow Land Burial   | /        | /           | /      |                                    |                                     |           |                         |
| Intermediate Depth Disposal    | *        | /           | /      |                                    |                                     |           |                         |
| Deep Geologic Disposal         |          |             |        |                                    |                                     | /         | /                       |
| Deep Well Injection††          |          |             |        | /                                  | /                                   |           |                         |
| Hydrofracturing                |          | /           |        |                                    |                                     |           |                         |

② Base case for each waste group. Includes input data sets for 2 additional computer runs to establish data spread between high risk and low risk for one waste group.

\* Not considered due to high water table.

† A warehouse type structure is to be used as a prototype.

†† With and without site security.

disposal option developed under the disposal model designs study, and the hydrogeologic and climatic data for seven settings developed by the site characterization study as inputs into the modified PRESTO model. The result will be the health effects associated with disposing of each waste stream using each applicable disposal option for the settings defined in Table 2.

## 6.0 COST BENEFIT ANALYSIS

The cost of disposing of various types of low-level waste will be evaluated for the appropriate disposal options at the sites listed in Table 2. Appropriate sets of cost categories will be selected for each of the eight waste disposal methods. These categories will include complete coverage of all of the facility owner's eventual costs in all phases of facility life including site selection, licensing, construction, operation and decommissioning. These cost category sets will be specifically structured to highlight the principal cost driving parameters of each disposal method in each phase. Sufficient detail will be included in cost categories to evaluate variations due to differences in disposal method and setting without overburdening the results with details that would remain essentially fixed.

Variations of risks and benefits associated with the different low-level waste disposal methods will be analyzed. A baseline disposal method will be selected for each waste group. Changes in risk and cost with respect to the baseline method will be calculated for all of the other methods appropriate to that waste group. The facility costs and cost to dispose of a unit of waste figures developed earlier in this project and the health risk estimates calculated using PRESTO will be used to estimate changes in benefits divided by changes in costs. Methods which provide greater benefit (reduced risk) for less cost than the baseline methods will be examined.

Figure 2 illustrates the range of possibilities in the cost/risk space. Generally three regions exist as defined in the figure. In the shaded region both cost and risk increase with respect to the baseline. It is intuitively evident that no method falling in that area should be chosen in preference to the baseline case. In the area marked BEST SELECTION both cost and risk reductions are realized; any method falling in that area should be used instead of the baseline case. Finally, the areas marked TRADEOFF REQUIRED allow cost reductions at increases in risk and vice-versa. These provide options from which alternatives to the baseline method must be selected. A horizontal line running across the figure indicates there may be other technical factors in the choice of method, such as an upper bound on acceptable risk.

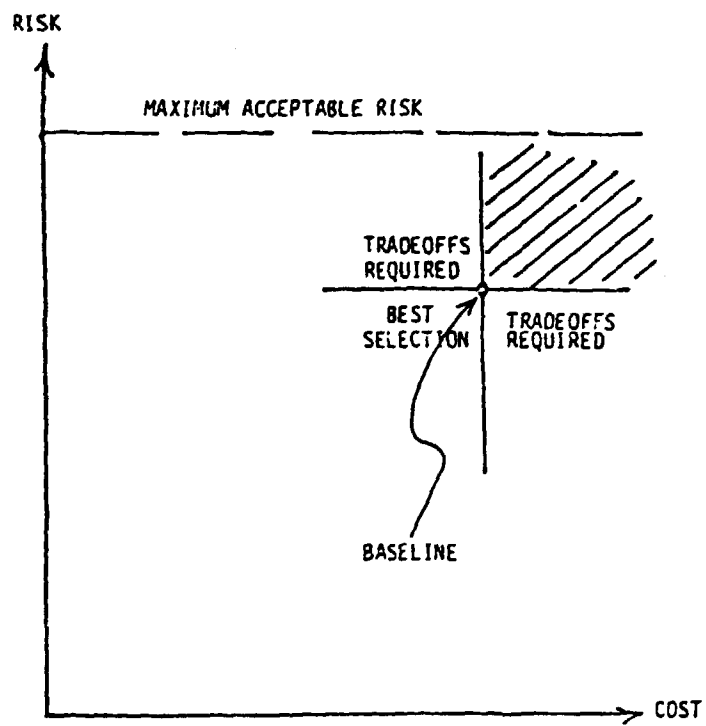


Figure 2. Possible Benefit/Cost Variations Around the Baseline

## 7.0 CONCLUSION

The analyses described in this paper will permit a rational evaluation of the relative merits of disposing of various types of low level nuclear wastes at different sites using any one of eight disposal techniques. The evaluation will be based on the cost of the disposal approach and on the health risk posed to the populace.



## ALTERNATIVES TO SHALLOW LAND BURIAL

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### ABSTRACT

During FY79 and FY80 the Los Alamos National Laboratory and its contractors performed a preliminary assesement of several alternatives to shallow land burial of low-level waste, including deeper burial, mined cavities, specially engineered storage buildings, well injection of liquid waste, and seabed disposal. Only deeper burial and mined cavities seem acceptable as near-term alternatives. A waste management program using a combination of disposal alternatives is recommended. Research needed to implement the deeper burial and mined cavity options is identified.

## INTRODUCTION

Since the early 1940s, solid low-level radioactive wastes (LLW) have been disposed to shallow trenches and shafts. With the exception of limited use of absorption beds and injection wells, liquid wastes have been solidified and buried. Until the early 1970s, some wastes were dumped into the ocean. Shallow land burial (SLB), however, has historically been the primary means of radioactive waste disposal in the United States.

Initially, SLB was intended to provide better containment than conventional landfill practices of the time. The advent of commercial nuclear power production and the increasing use of radioactivity for medical and industrial purposes have resulted in ever-increasing volumes and varieties of waste types and waste forms. Many high-activity and/or mobile wastes are not suited to SLB. Repeated releases from disposal trenches resulting in the closing of commercial burial facilities strongly suggest that SLB, as currently practiced, is not adequate as a disposal method for LLW.

Several options exist as means to alleviate this situation and to reduce the likelihood and consequences of exposure to the public. One is to develop new techniques to improve the containment capability of SLB. Many research projects are directed toward this end. Another option is to develop alternative disposal methods for LLW. During FY79 and FY80 the Los Alamos National Laboratory and its subcontractors at the University of Arizona, the University of Texas, Austin, and JRB Associates, Inc. studied the advantages of several alternatives to SLB, including deeper burial, mined cavities (geologic repositories for LLW) engineered structures, subsurface injection of liquid wastes, and seabed disposal. On the basis of technical, economic, and political considerations only two near-term alternatives seem realistic: deeper burial and mined cavities. In this paper I can only briefly outline the alternatives. More complete discussions can be found in the cited references.

## DEEPER BURIAL

This alternative is similar to SLB except that the waste is buried deeper. While trench size may vary, depending on site characteristics, usually trenches about 20 m deep, 10-20 m wide, and a few hundred meters long should be sufficient. Engineered caps 5-10 m thick are an integral part of the disposal system, as are liners, sumps, and drain systems (1,2). Trench depth will depend on local geology, hydrology, and the type of equipment used. In general, the deeper the trench, the larger the volume of waste that can be buried for a given surface area. But side slope stability may limit depth.



Ideally, the waste is above the local water table and infiltration of precipitation is reduced by carefully designed trench caps and wick systems - concentric layers of natural materials, such as sand and gravel, that "wick" moisture away from the waste (3,4,5). The major advantages of increased depth of burial are that the effects of plant and animal intrusion and of surface erosion are significantly reduced, and that most shallow excavation activities will not penetrate deeply enough to expose the waste (1,6).

Deeper burial does, however, place the waste closer to the water table. For many potential sites, especially in the western United States, the local water table is sufficiently deep that ground water will not be an important factor in trench design (7). In areas of high precipitation the waste should be placed above or entirely below the zone of water fluctuation (8) and in rock units having higher permeabilities than the trench/cap system (1,9). If the waste must be below the water table, wick systems, which are effective only under unsaturated conditions, are out of the question. In this instance, more reliance must be placed on trench liners and radionuclide migration barriers, and on waste packages.

All the technology necessary for site construction, waste loading, and operation is currently available.

#### MINED CAVITIES

The mined cavity option involves placing the waste in excavated underground rooms at depths of 100-300 m (10). These are essentially geologic repositories for LLW. While considerably more expensive than SLB or deeper burial, this alternative is especially attractive for the disposal of some high-activity wastes, such as reactor resins and irradiated components.

Probably the most important aspect of mined cavity disposal is site selection in the proper host rock (8,11). Site selection requires geologic, hydrologic, and rock mechanic studies before a disposal site can be established (12). In general, favorable rock types are plutonic rocks (such as granite), basalt, limestone, and tuff (7,10). Shale and salt seem less favorable but, in some areas, may be acceptable. These rocks are more mechanically safe, environmentally stable, and impermeable than sandstones and metamorphics. However, if the disposal site is located above the water table, sandstones may be suitable. We do not recommend using existing mines (operational or abandoned) for waste disposal, as rock properties may not be optimum and there is always the danger of abandoned mines or stopes being reopened as a result of economic pressures. In addition, the ground may be unstable in previously mined areas, ventilation may be more difficult, and access routes may be long and impracticable (10).

Much of the required research in this area is already being conducted under the high-level waste (HLW) geologic repository programs and is immediately applicable to mined cavities for LLW (7,13). In addition, because of the lower thermal loading and shorter half-lives of LLW, the requirements for a mined cavity are less restrictive than for the geologic repository (11). Some candidate sites rejected by the HLW programs may be suitable for LLW. Many suitable sites are available in all regions of the country. Marginal localities can be discarded because a sufficient number of good sites exist (10).

Major disadvantages of mined cavities are the higher costs of site selection, construction, operation, and site closure (10).

### ENGINEERED STRUCTURES

Above-ground or slightly below-ground engineered structures are probably not acceptable as a disposal method but may be useful for interim storage of some LLW, such as irradiated components. Unless otherwise noted, the following is referenced to work performed by the University of Arizona for the Los Alamos National Laboratory (14,15).

The waste can be more easily monitored and, if necessary, cooled for short time periods. Because storage buildings can be completely engineered for specific functions, site requirements are not as strict as for SLB or any of the alternative disposal methods (8). However, geologic and hydrologic investigations will be required to define design criteria, site monitoring programs, and accident response procedures. If long-lived radionuclides are present, the waste may require moving, after some reasonable time period, to less accessible disposal areas (a deeper burial trench or mined cavity, for example).

While in storage, the waste will be available for recovery of any desired materials. In future relocation of the waste, the latest state-of-the-art techniques can be used. However, deterioration of the waste matrix and containers may make removal more hazardous than the initial emplacement. Relocation will also increase final disposal costs.

As the storage buildings will be designed to withstand any expected earthquakes, tornadoes, and so forth, catastrophic releases are not anticipated. However, slow loss of structural integrity through time may result in slow releases of radioactivity directly into the biosphere. Another problem is that the waste will be accessible to unauthorized intrusion by man. Thus the site can never be decommissioned and strict site control measures must be maintained throughout the hazardous lifetime of the waste.

Several foreign countries are using engineered structures for temporary storage and aging of wastes before final disposal (16,17,18,19).

## WELL INJECTION

Many wastes are liquids, and solidification may increase their volume, as well as require off-site disposal. This alternative is a method for on-site disposal of liquid wastes. Three techniques are currently in use. Fluid injection involves injecting liquid waste through a disposal well into a permeable host rock, often at depths of a kilometer or more (8). Injection may be by pumping or by gravity flow (9,20). The host rock must have high porosity and permeability to avoid fracturing and be bounded above and below by impermeable strata (21,22,23).

The hydrofracturing technique is similar, but the waste is mixed with a grout, which eventually hardens. The host rock is usually a well-bedded shale. Horizontal fractures are initiated with a small slug of water at a pressure exceeding in situ rock strength, followed by injection of the waste/grout mixture, which extends the fractures, resulting in a thin grout sheet a few hundred meters across (8). After several injections, the well is cemented off and a higher level is developed (24, 25, 26). Liquids that are formed as the grout cures can be pumped back to the surface for reuse.

Liquid waste/grout mixtures can also be injected into specially constructed caverns. The caverns can be excavated by standard mining methods. The waste is prepared in a fashion similar to that for hydrofracturing, then slurried down an access shaft (9,20).

The extreme mobility of liquid wastes requires that strict site selection criteria be established and waste emplacement procedures be carefully followed. Monitoring wells must be drilled to determine the limits of fluid migration. Retrievability is, of course, impossible.

The Soviet Union uses well injection regularly (27) and Oak Ridge National Laboratory has extensive experience in hydrofracturing (26). Cavern injection is under investigation in the Federal Republic of Germany (28). Considering the current trend in this country toward solidification and burial of liquid wastes, this alternative does not seem particularly attractive at this time.

## SEABED DISPOSAL

Seabed disposal of LLW involves waste treatment and packaging, followed by dumping the waste containers over a specified deep sea disposal area. The containers free-fall to the ocean floor.

Little is known about the seabed as a disposal medium and extensive investigations on the sorptive properties of pelagic sediments, ocean currents, and food chains are required before a

suitable site can be selected (29). Expensive port facilities and specially equipped ships will be needed. Thus seabed disposal is not a near-term option.

Several foreign countries having limited land area use the seabed as a waste disposal site (18,30,31). Between 1946 and 1962, when land disposal came into general use, the United States dumped much of its waste into the ocean; then the practice of ocean dumping decreased from 1962 to 1970 (32). Ocean disposal is currently under jurisdiction of the Environmental Protection Agency, which requires demonstration that no alternative disposal methods exist before a license can be issued (33). Many forms of land disposal, including those discussed in this paper, do exist.

A considerable amount of research on seabed disposal is being conducted in foreign countries (16,18,30,31,34) and subseabed disposal of HLW is under investigation in this country (7,9). While research for the U.S. HLW program is somewhat extravagant for LLW, all foreign and domestic research in this area is directly applicable. We therefore suggest that no new research for seabed disposal of LLW is necessary.

#### RESEARCH NEEDS

We have identified two disposal methods that seem reasonable as near-term alternatives to SLB. These are deeper burial and mined cavities. However, before these options can be implemented, further research in several areas is still required. These research needs are outlined below.

##### Research Needed To Implement the Deeper Burial Alternative

1. Identify monitoring techniques to measure water movement through inhomogeneous media (jointed, fractured, bedded rocks).

Several site-specific studies along these lines are being conducted by national laboratories (in tuff at Los Alamos and in shale at Oak Ridge) in support of SLB. Analogous research for the HLW programs is being performed by universities for the Nuclear Regulatory Commission (NRC) and by the Nevada Nuclear Waste Storage Investigation (NNWSI) Project at the Nevada Test Site (NTS). This work must be applied to deeper burial.

2. Identify possible chemical reactions that may occur under various oxidation/reduction and pH conditions in a deeper trench.

Generic research in this area is being conducted by Brookhaven National Laboratory (BNL). New research along these lines does not seem warranted at this time.

3. Determine possible migration rates and controlling pathways for radioactive gases, such as tritium, radioactive carbon dioxide and methane, and probably krypton and iodine. Potential releases must be related to pressure gradients, thermal gradients (for gases dissolved in trench water), and soil permeability.

Again, BNL is already involved in this kind of work. Results of work performed under the Remedial Action Program for uranium mill tailings is immediately applicable, but these studies must be applied to deeper burial.

4. Study special backfill materials to maintain desired chemical conditions in the trench region.

NRC is supporting research in this area at the SLB site at Maxey Flats, Kentucky. Also, similar studies are underway in the Federal Republic of Germany. These activities should be applicable to deeper burial.

5. Test the use of reinforced fabrics in trench caps as to effectiveness and life expectancy.

The best materials to use in trench caps seem to be natural materials, but fabrics may be useful to provide short-term stability until the sand, clay, and gravel have achieved sufficient consolidation. Life expectancy estimates can be readily obtained from the manufacturer. Research to test various fabrics is currently underway at Maxey Flats and at the University of Arizona.

6. Test different slope stabilization techniques to determine which ones will allow for maximum slope of trench walls.

This information is site-specific, and slope stability will depend on the host rock, trench depth, and prevailing ground conditions. There is a considerable amount of experience at current SLB facilities. Any additional needed information should be readily available from mining and construction companies.

7. Determine sorptive capacity for radionuclides in soil/waste solution systems.

Investigations to address this question are being conducted at Maxey Flats by the U.S. Geological Survey and at West Valley, New York

7. Determine sorptive capacity for radionuclides in soil/waste solution systems.

Investigations to address this question are being conducted at Maxey Flats by the U.S. Geological Survey and at West Valley, New York by the New York State Geological Survey for the NRC. Sorptive studies using liquid waste from Los Alamos and soils from Maxey Flats, Sheffield, Illinois, Beatty, Nevada, and Los Alamos are being performed by the Los Alamos National Laboratory.

8. Field testing to verify computer models for unsaturated flow.

Very little work is being done in this area. As part of a program to study advanced SLB technology at arid sites, Los Alamos is verifying an unsaturated flow model (WAFE). Small programs in Canada and at Hanford are investigating the problem, but more work is clearly needed.

9. Obtain experimental data on hydraulic conductivity as a function of saturation and porosity.

These studies will require considerable time in drilling and performing in situ hydraulic conductivity experiments, such as slug and packer tests.

10. Field test wick systems (unsaturated conditions).

Los Alamos is currently investigating various wick system designs as part of a program to develop advanced SLB technology.

11. Test various waste matrix materials for leach rates.

Leach rates of waste matrices are being extensively studied by the HLW program in this country and by LLW and intermediate-level waste programs in Europe. Results from these investigations will be applicable.

#### RESEARCH NEEDED TO IMPLEMENT THE MINED CAVITY ALTERNATIVE

1. Modeling of fluid flow into storage rooms or tunnels under unsaturated conditions for various input parameters, such as water content, unsaturated hydraulic conductivity, infiltration rates, and source terms.

Saturated flow into mines has been studied extensively, but little has been done to address the problem of unsaturated flow. The only obvious analogous studies in the geologic repository program are related to the NNWSI project at NTS. This work is quite generic at this time.

2. Determine the economic and cost/benefit factors for mined cavities as a function of size, host rock, mine layout, and type of equipment used.

Several design studies have been, and are being, performed in relation to the geologic repository for HLW. The repository will be much deeper than a mined cavity for LLW, and design criteria are much more stringent. Thus costs will have to be scaled down before being applied to a mined cavity.

3. Modeling of migration pathways for various radionuclides most likely to cause exposure to man.

This work has essentially already been done by the Office of Nuclear Waste Isolation (ONWI), the national laboratories, and many universities. The information is scattered throughout the technical literature and must be pulled together and evaluated.

4. Characterize variations in horizontal and vertical permeabilities as a function of depth in candidate host rocks (granite, basalt, tuff, limestone, shale) and identify possible relationships between bed or flow thickness and permeability.

We have been unable to find any previous or current research in this area. The amount of information required is quite extensive and there is a very limited data base to draw from.

5. Investigate changes in permeabilities of surrounding rocks caused by drilling and blasting and by rig drilling.

This question is being addressed by the University of Arizona for NRC.

6. Measure permeabilities of large selections of rock.

Lawrence Berkeley National Laboratory is conducting these investigations.

7. Evaluate various potential entry seal materials and configurations with reference to long-term stability, behavior under field conditions, physical and chemical properties, and the effect of placement and seal geometry on surrounding rocks.

This work is already in progress. The initial work, as applied to geologic repositories, is being performed under ONWI. Confirmatory research supported by NRC is also underway. The results of this research are applicable to mined cavities for LLW.

8. Evaluate changes in rock chemistry and possible long-term effects of mining operations (for example, oxidation processes and the effects of bacteria and algae introduced into the mine during operation).

This issue is being addressed by BNL and no new work seems required at this time.

9. Determine in situ rock characteristics without extensive drilling.

The development of improved geophysical techniques for use in determining rock quality and rock mass classification is being done by the University of Arizona for NRC.

#### SUMMARY

During FY79 and FY80 the Los Alamos National Laboratory and its contractors at the University of Arizona, the University of Texas, Austin, and JRB Associates, Inc. performed a preliminary evaluation of several alternatives to shallow land burial. These alternatives include deeper burial, mined cavities, engineered structures, well injection of liquid waste, and seabed disposal. We identified two alternatives as reasonable near-term disposal options: deeper burial and mined cavities.

The Low-Level Waste Management Program would do well to consider a combination of alternatives rather than disposing of all LLW in the same manner. For example, most LLW could be buried in deeper trenches, while high-activity wastes and those containing longer-lived radionuclides could be disposed of in mined cavities for LLW. Waste having potential resource value and some other wastes could be temporarily stored in engineered structures until final disposition has been decided.

Research needed to implement the deeper burial and mined cavity alternatives has been identified. Some of this work is currently in progress, some is being conducted for other programs and must be applied to LLW, and some is new.



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## REVIEW AND SUMMARY OF THE LLNL WASTE MANAGEMENT HAZARD ASSESSMENT PROJECT

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### ABSTRACT

The Waste Management Hazard Assessment project at the Lawrence Livermore National Laboratory has been concentrated on study of hazards resulting from toxic material (either man-made or natural) buried underground. Its objective has been the development of rationale to support and assist in waste management programs. The project began in June, 1979 and is scheduled for completion in September, 1982. This paper presents a review and summary of highlights and accomplishments of the project. These include: development of the geotoxicity hazard index, analog modeling methodology, and various perspectives on the nature of risks related to radioactive waste management.

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### INTRODUCTION

Over the past two decades, the preponderance of research on radioactive waste management was devoted to development of methods and techniques for its safe disposal. Very little study had gone toward determination of technical rationale to support and guide this effort. In 1979, the Waste Management Hazard Assessment (WMHA) project was begun at the Lawrence Livermore National Laboratory (LLNL) under sponsorship of the DOE Low-Level Waste Management Program (LLWMP). This project was primarily devoted to defining the nature of the problem and assessing rationale for its proper solution. Among its objectives were:

- o Development of risk measures for waste management
- o Defining the hazard of underground toxic material, either man-made or natural.
- o Development of perspectives on waste management hazards, and
- o Assessment of methodology for predicting consequences of underground disposal of waste.

The project is scheduled for completion at the end of FY82. During this final year of the project, LLNL has been supported by subcontract with Science Applications, Inc., for the orderly completion and termination of ongoing tasks. Much of the work of the WMHA project, particularly in the areas of hazard measures, indices, and waste classification has been assumed by EG&G - Idaho and is continuing under their direction.

The purpose of this paper is to present a review and summary of highlights and accomplishments of the WMHA conducted at LLNL.

## PROJECT REVIEW

Perhaps the most effective way to review the work produced by the WMHA project is to present brief abstracts and summaries of papers, reports, and publications. The purpose is to provide the reader an acquaintance with the subject matter and to stimulate interest with the approach taken. Of course, more detailed insights can be gained by reviewing the original published material. For clarity and convenience, report references are presented at the end of each abstract review. These reviews will be presented in chronological order:

## WASTE MANAGEMENT RISK ASSESSMENT ISSUES

A review was performed of risk assessments, policy statements and general perspectives on radioactive waste management. The review included technical literature, public testimony and popular articles. An interesting insight is that widely divergent views can be drawn from the same basic data. These views are reviewed and assessed.

Reconciliation of divergent viewpoints presents a difficult challenge. It is doubtful that issues having an emotional or philosophical basis can be resolved through technical efforts. It is suggested that improved public education may provide the key.

Smith, C.F. and J.J. Cohen, "Perspectives and Reconciliation of Viewpoints on Risk Assessment Issues", Waste Management '80, University of Arizona, Tucson, AZ., March 1980.

## NATURAL BACKGROUND RADIATION

To assess the potential for cancer induction by low level radiation exposure, a statistical evaluation of cancer incidence vs. background radiation levels in the U.S.A. was performed. A significant negative correlation was found between background radiation exposure and cancer incidence. Based on the analysis it was found that the very high cancer

induction rates ( $>10^{-3}$  cancers/man-rem) that have been suggested by some authors, would be highly unlikely.

Cohen, J.J., "Natural Background as an Indicator of Radiation-Induced Cancer", Proc. 5th Int'l. Cong. of the IRPA, Jerusalem, Israel, March 1980.

#### GEOTOXICITY HAZARD INDEX

To adequately define the problem of waste management, quantitative measures of hazard must be used. This study reviews past work in the area of hazard indices and proposes a geotoxicity hazard index for use in characterizing the hazard of toxic material buried underground. Factors included in this index are:

- o An intrinsic toxicity factor, formulated as the volume of water required for dilution to public drinking-water levels.
- o A persistence factor to characterize the longevity of the material, ranging from unity for stable materials to smaller values for shorter-lived materials.
- o An availability factor that relates the transport potential for the particular material to a reference value for its naturally occurring analog.
- o A correction factor to accommodate the buildup of decay progeny, resulting in increased toxicity.

Smith, C.F., J.J. Cohen and T.E. McKone, "A Hazard Index for Underground Toxic Material", UCRL 52889, June 1980.

#### LEAD ENCAPSULATION

Lead encapsulation of high level nuclear waste has been proposed as a means for enhancing its long term stability. In this application, lead is used for metal matrix formation or as a canister material. This study shows that the lead itself could pose a more severe long term health problem than the nuclear waste. For example, for the KBS package design it is calculated that after 400 years of decay, the lead shield has a greater intrinsic toxicity than does the waste it surrounds, since lead has a greater leachability in groundwater than does borosilicate glass. Finally, the biological effects of lead are similar to those due to radiation in many respects including carcinogenicity.

Cohen, J.J., D.G. Coles, and L.D. Ramspott, "Lead Encapsulation of Nuclear Waste: A Risk Assessment", Trans. American Nuclear Society, Las Vegas, NV., June 1980.

## TRANSMUTATION RATIONALE

An assessment has been performed on rationale behind the concept that transmutation of nuclear waste could lead to an elimination or reduction of hazard. This concept is largely based on the myth that long half-life is a primary contributor to the hazard of a radionuclide. From this assessment it is concluded that transmutation is neither necessary, economical, or safer than other disposal options. Additionally, more radwaste could actually be produced than destroyed in the process. This paper presents a detailed critique of rationale for transmutation and a discussion of the perspective on resultant risk.

Smith, C.F., and J.J. Cohen, "A Critique of Rationale for Transmutation of Nuclear Waste", Proc. Int'l. Conf. on Nuclear Waste Transmutation, University of Texas, Austin, TX., July 1980.

## HAZARD OF LONG-LIVED RADIONUCLIDES

There is a tendency to equate the potential hazard of radionuclides to their half-lives. This tendency is somewhat paradoxical since, carried to its extreme, the toxic stable elements should cause the greatest concern. Also, the inverse relationship between specific activity and half-life should be a primary consideration.

As a case in point, the hazard of iodine-129 exposure is assessed. It is determined, for example, that if the iodine atoms normally in the human body were entirely I-129, one would still be below maximum body burden levels.

Considering the laws of solubility, certain long-lived radionuclides could never reach MPC levels in water. For example, converted to mass units, the MPC<sub>w</sub> concentrations for In-115 and Nd-144 are respectively 14.5 and 59.2 kg/liter. Further, the MPC level for U-238 has been based on its chemical toxicity rather than its radiotoxicity.

Considering these and other examples, it would appear that long half-life per se, is not a cause for concern in radiological protection, although it is a factor which can be taken into consideration.

Cohen, J.J., C.F. Smith and A.J. Toy, "The Hazard of Long Lived Radionuclides", Health Physics Society meeting, Seattle, WA., July 1980.

## TOXICITY AND CARCINOGENICITY OF MINERAL DEPOSITS

The question of the hazard to humans and the environment is paramount to disposal of low-level radioactive wastes. The toxic hazard of heavy metals can serve as a useful reference point for comparison to low-level wastes. In this survey, certain aspects of heavy metals and their toxic properties were reviewed for eventual comparison to the low-

level waste problem. Five elements known to cause toxic effects to humans were considered: arsenic, selenium, cadmium, lead and nickel.

Furst, A. and I. Harding-Barlow, "A Survey of Toxicity and Carcinogenicity of Mineral Deposits", University of San Francisco report to LLNL, October 1980.

#### ASSESSMENT OF ACTUAL VS. PERCEIVED RISK

Actual risks (those which can be objectively evaluated) are usually different from what is perceived by the public. Many factors contribute to the disparity between actual and perceived risks. These factors are discussed and the nuclear waste problem is assessed as a case in point.

It is determined that when technological fixes are applied to reduce unfounded public fears (perceived risks), the approach may not only be ineffective but could be counterproductive. It is suggested that waste management procedures which are out of line with actual risks may prove detrimental toward the goal of gaining public confidence.

Cohen, J.J. and C.F. Smith, "Assessment of Actual and Perceived Risks of Energy Development", Trans. American Nuclear Society, Washington, D.C., November 1980.

#### RISK ASSESSMENT AND RADIOACTIVE WASTE MANAGEMENT

This paper reviews the development of the Geotoxicity Hazard Index and the continuing study to improve estimation of the availability term. The objective has been to achieve more accurate application of analog methodology. The analog approach is empirical and relates the relative abundance of geotoxic materials to their rate of biological intake. The main advantage over deterministic modeling as a means of effects prediction is that it circumvents or eliminates the need for input parameters which are uncertain, unknown or unknowable. Incorporation of new or improved information will provide increased confidence in the results.

Cohen, J.J. and C.F. Smith, "Risk Assessment and Radioactive Waste Management", Trans. of the Interagency Workshop on Modeling and Low-Level Waste Management, Denver, CO., December 1980.

#### WASTE MANAGEMENT HAZARD ASSESSMENT

A summary and overview of the WMHA project as of January 1981 is presented. The topics covered include: Waste Management Hazard Indices, Risk Assessment Review, Geotoxicity Data Base, LLW Monitoring, LLW Alternatives Evaluation, LLW Modeling Methods and Technical Assistance. A



detailed discussion on the hazards of long-lived radionuclides is also presented.

Cohen, J.J., "Waste Management Hazard Assessment", Proc. 2nd Ann. NLLWMP, Informational Meeting, ORNL/NFW-81/08, San Diego, CA., January 1981.

#### MODELS AND CRITERIA

An objective of predictive modeling is the determination that health and safety objectives can be met by waste management programs. Computational limitations are discussed in detail. The study provides the following conclusions: (1) Definitive Criteria are required. (Vague criteria are useless), (2) Models should not require unknowable inputs, (3) Model complexity does not necessarily enhance credibility and, (4) Model output is no better than its input. The results of even the most sophisticated models should be viewed with great caution.

Smith, C.F. and J.J. Cohen, "Models and Criteria for Waste Repository Performance", UCRL-85402, Proc. Symp. on Uncertainties Associated with Geologic Disposal of Radioactive Waste. Gatlinburg, TN., March 1981.

#### ANALOG METHODS FOR MODEL VALIDATION

Incorporation of toxic or hazardous material in the earth's crust is a phenomenon not unique to radioactive waste burial. Useful insights on the environmental transport and effects of underground toxic or radioactive material can be derived from comparative analysis against natural (mineral) analogs.

This paper includes a discussion of the background and rationale for the analog approach, a description of several variations of the approach, and some sample applications to illustrate the concept, focusing on Radium-226 and Iodine-129 as specific case studies.

It is concluded that natural analogs can provide useful insight, not only for nuclear, but also for non-nuclear hazardous material underground disposal. The approach can also be a tool for model verification. Finally, the approach provides a focus for studying and understanding geotoxic phenomena and the role of natural minerals in human health and disease.

Cohen, J.J. and C.F. Smith, "Validation of Predictive Models for Geologic Disposal of Radioiodine Waste Via Natural Analogs", UCRL-85355 Proc. Symp. on Uncertainties Associated with Geologic Disposal of Radioactive Waste, Gatlinburg, TN., March 1981.

## WASTE FORM RATIONALE

Significant effort has been devoted toward development of improved waste forms suitable for the long term isolation of high-level nuclear waste. This study reviews and assesses the rationale motivating this effort. In the absence of definitive waste form criteria, it is not possible to determine how good is good enough. A review of model calculation results indicates that overall repository performance is relatively insensitive to waste form characteristics. From a cost/benefit or ALARA standpoint it may be concluded that the already developed borosilicate waste form is adequate for purposes of waste isolation in geologic media. Accordingly, there may not be sufficient justification for continued effort toward waste form improvement.

Musgrave, B.C., J.J. Cohen, and C.F. Smith, "Assessment of Rationale for Development of Improved Waste Forms", Trans. American Nuclear Society, Miami, FL., June 1981.

## BENEFICIAL RADIATION EFFECTS

A review is made of recent assessments indicating the possibility that certain low levels of exposure to ionizing radiation, may have a net beneficial affect. Previous analysis of a negative correlation between background radiation and cancer incidence, as well as evidence on radiation "hormesis" by T.D. Luckey and statistical analyses by R.J. Hickey are reviewed. This review indicates there is a very real possibility that, as with other harmful stresses, there may be a net beneficial effect of exposure to low levels of radiation.

Cohen, J.J., "Possible Beneficial Effects of Low-Level Radiation", Trans. American Nuclear Society, San Francisco, CA., November 1981.

## GEOTOXIC MATERIALS IN THE SURFACE ENVIRONMENT

Knowledge of the natural distribution of toxic substances in the surficial geologic environment provides a useful baseline for the assessment and eventual use of geologic media for the storage and/or disposal of hazardous wastes. This study developed data sources and provided insights into the existence and behavior of geotoxic substances in the environment. In addition, geotoxic deposits were related to the general problem of hazardous and radioactive waste disposal.

Basic types of geotoxic deposits identified were: ore bodies, evaporities, geothermal deposits and volcanic materials. Specific elements examined for their environmental and biologic behavior were As, Cr, Cd, Ni, Pb, Se, U, and V.

The properties of these toxic elements were summarized and presented in a toxicity matrix, which is used as a first approximation for ranking

their geotoxicity. This ranking showed that arsenic and cadmium has the highest toxicity, while uranium has the lowest.

Baseline data provided by this report could be used for 1) perspectives study on hazardous waste problems, 2) a better understanding of toxic material transport, 3) an overview on waste repository siting in high geotoxicity areas, and 4) an improved understanding of geochemistry and health relationships.

Koranda, J.J., J.J. Cohen, C.F. Smith, and F.J. Ciminesi, "Geotoxic Materials in the Surface Environment", Lawrence Livermore National Laboratory, UCRL-53215, December 1981.

#### AVAILABILITY OF GEOTOXIC MATERIALS

"Availability" refers to the potential for release, migration, and human assimilation of toxic materials buried in the earth's crust (geotoxic materials). This report provided an approach to derive availability factors which describe the physical and chemical characteristics of buried toxic material, and its behavior in the geochemical environment.

Natural analogs of buried toxic materials that were studied for this report are the Oklo natural fission reactors, uranium ore deposits, and hydrothermal ores.

Important parameters affecting the mobilization or immobilization of a selected group of toxic elements were defined from the geochemical literature and consisted of: Eh, pH, mineralogy, presence of anions or cations, and temperature. The m-factors (modification factors to availability) were generally defined but not numerically quantified for site-specific conditions.

Wachter, B.G. and P.L. Kresan, "Availability of Geotoxic Material", University of Arizona report to LLNL, January 1982.

#### OCEAN DISPOSAL

A risk assessment was performed to determine potential consequences of oceanic disposal of low-level radioactive waste. Past practices were reviewed and effects were predicted based on analog methodology. It is determined that radiological consequences would be very small relative to those due to natural oceanic radium which, in turn, are a trivial fraction of man's overall background radiation dose. It is unfortunate that oceanic disposal has an extremely adverse political image since, from a public health standpoint, it could possibly provide the safest alternative for LLW disposal.

Cohen, J.J. and C.F. Smith, "Risk Assessment of Oceanic Disposal of Low-Level Radioactive Waste", Waste Management '82 University of Arizona, Tucson, AZ., March 1982.

## GEOTOXIC ANALOGS

Incorporation of toxic material into the geosphere is not a phenomenon unique to underground burial of radioactive or hazardous chemical waste. Several useful analogies can be drawn from observations derived from effects of natural toxic materials. These are reviewed and discussed. Toxicity indices were reassessed to determine the effect of incorporating data from ICRP-30 as opposed to the 10 CFR 20 values. The analog approach can be particularly useful for verification of effects predicted by calculational modeling.

Cohen, J.J., C.F. Smith, and D.A. Waite, "Geotoxic Considerations for Underground Burial of Hazardous Materials", Waste Management '82, University of Arizona, Tuscon, AZ., March 1982.

## TOXIC HAZARDS OF UNDERGROUND EXCAVATION

A frequently stated concern regarding underground disposal of radioactive waste is that, at some future time, people may intrude into the repository and suffer the consequences of exposure to the radioactive materials. A systematic survey of the overall toxic hazards of random intrusion by underground excavation was conducted. Instances where random intrusion resulted in hazardous conditions are documented and a perspective on the "intruder scenario" is developed.

Chitnis, V. et al., "Survey of Toxic Hazards of Underground Excavation", University of Michigan report to LLNL, April 1982 (draft), final report to be published.

## NATURALLY POISONED AREAS

Certain areas of the earth are characterized by such high concentrations of toxic elements that their use by plants or animals is precluded. The dry salt basins of Nevada and western Utah represent unique geochemical environments for the entrapment and concentration of mobile toxic elements such as arsenic, antimony and mercury. In this survey, naturally poisoned areas were identified and described, and factors which influence natural toxic material availability were compiled and analyzed. These areas were assessed for potential use as low-level radioactive waste disposal sites.

Brookins, D.G., H.A. Vogler, and J.J. Cohen, "Low-Level Radwaste Disposal in Poisoned Lands/Inland Seas", Trans. of the American Nuclear Society, Los Angeles, CA., June 1982.

## WASTE MANAGEMENT COST EFFECTIVENESS

A rationale and methodology for application of cost/benefit analysis to radioactive waste management is presented and discussed. Determination of cost effectiveness requires the establishment of definitive goals and objectives against which the effectiveness of any program or method may be determined. It also requires establishing a cost effectiveness guideline defining a point beyond which marginal cost per unit of hazard reduction may be considered excessive. Where the goal of a program is the reduction of public concern, technical methods may not generally be cost-effective.

Cohen, J.J., "Cost Effectiveness in Nuclear Waste Management", Trans. American Nuclear Society, Los Angeles, CA., June 1982.

## SUMMARY AND DISCUSSION

The closure and restriction of low-level radioactive waste disposal facilities in the U.S.A. reflects a widespread public concern toward such facilities. The work of the WMHA project indicates that this concern is not founded on a rational technological basis. Attempts to predict the effects of underground disposal of radioactive waste have been largely based on application of deterministic modeling using "worst case" assumptions. The calculated effects are therefore highly biased toward conservatism (pessimism). Despite this approach, the results of these modeling efforts generally predict that the consequences are minimal, however due to uncertainties in the models and input parameters, the results have not provided a sound basis for decision making. Neither have these results had much of an effect on mitigating public fears.

The analog approach provides a method of characterizing risks and hazards of underground burial of wastes utilizing observations of natural analogs to the radionuclides of concern. Although the approach is largely empirical, it has the advantage of reflecting actual data and observation as opposed to theoretical assumptions. Further development of analog methodology will be required to gain confidence of the public and the technical community. However, this approach has already found application in the areas of perspective development, hazard characterization and in cost benefit analysis which requires average collective dose inputs rather than worst case estimates of effects. Perhaps continued development, particularly in quantifying biological availability phenomena, will lead to attainment of greater confidence in the methodology.

ADDITIONS TO THE PRESTO COMPUTER CODE TO FACILITATE  
COMPARISON OF RISKS FROM LOW-LEVEL WASTE DISPOSAL OPTIONS\*

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PRESTO OVERVIEW

PRESTO (Prediction of Radiation Effects from Shallow Trench Operations) is a computer code developed by the Oak Ridge National Laboratory under EPA sponsorship. (1) PRESTO calculates health effects from disposal of low-level radioactive waste in shallow trenches for up to 1,000 years after burial. It models groundwater transport, overland flow, erosion, resuspension, atmospheric dispersion, inhalation, deposition on crops and surfaces and ingestion of contaminated beef, milk, crops and water. Normal performance of the disposal facility and human intrusion can be accounted for and the code calculates cancer risks to individuals and populations.

Under EPA contract, TRW and Rogers and Associates are engaged in developing cost/benefit ratios for the following alternatives for low-level waste disposal: engineered surface storage (ESS), sanitary landfills (SLF), shallow land burial (SLB), improved shallow land burial (ISLB), intermediate depth burial (IDB), deep geologic disposal (DGD), hydrofracture (HF), and deep well injection (DWI). The basis of the risk analyses will be the PRESTO code, but it must be supplemented in order to calculate risks from the broad range of disposal options that are the subject of the current effort. Generally, the additions take two forms: they represent characteristics of the disposal options, such as ESS, DGD, HF and DWI, that are clearly different from shallow land burial; or, they model details that permit distinction between SLB and other near surface burial options such as ISLB, IDB, and SLF. This paper describes the motivation and the essence of the additions to PRESTO presently under development.

The approach to expanding the capabilities of PRESTO is, whenever possible, to modify existing submodels for use in modeling the alternative disposal methods. Modifications of submodels and subroutines will be accomplished in a manner such that, when disposal

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\*This work is supported by the Environmental Protection Agency under contract #68-02-3174.

options other than shallow land burial are not being modeled, PRESTO will execute as though the code had not been modified. In this manner, the modified code can be used as an unmodified version of PRESTO when desired.

Engineered surface storage will be modeled as follows: container failure, either as the result of simple mechanical failure or gas generation will be represented as releases to the atmosphere that depend on user supplied expected container failure rates. Thereafter, the releases will be treated by the PRESTO atmospheric models. Because the principal benefit of ESS occurs when the majority of the radioactivity of the waste comes from short-lived nuclides, the set of isotopes currently treated by PRESTO will be expanded to include a number of nuclides not usually considered in analyses of low-level waste burial. Direct exposure to crews handling the wastes within the ESS structure will also be modeled. Following the completion of the ESS phase, it is assumed that the waste will be transferred to a SLF. In the risk analysis the SLF models will be applied to calculate risks after that transfer.

PRESTO will be enhanced to treat mixtures of a fraction of the waste with cover soil in order to realistically model sanitary landfill. For SLF's not under administrative control, the loading rate and length of the loading period will be taken into account to take credit for decay of the shorter-lived isotopes while the facility is being controlled.

Improved shallow land burial will be modeled in the same manner as SLB except that use of special materials such as montmorillonite clay as backfill between the waste packages will be incorporated in the modeling. Only minor modifications to PRESTO are required, to permit the backfill to be modeled as having a different permeability than the cap.

In order to distinguish intermediate depth burial from other near-surface burial options (such as SLB) two expansions of PRESTO will be implemented. One will treat plant root penetration and plant uptake as a function of cover thickness. The other will account for changes in water infiltration into the waste due to consolidation of the cover with time. Each of these expansions permits the migration model to account for the thicker cover over the waste that is the key feature of IDB.

The major addition to PRESTO that will improve its ability to model deep disposal options such as deep geologic disposal, hydrofracture and deep well injection will be a one-dimensional convective/dispersive flow model, based on analytic solutions for infinite pipes. This model will be constructed in a manner that treats the interactions between vertical flow of water to an aquifer and flow through the aquifer. This will more accurately represent the potentially low flow rates at depth. Once

the isotopes leave the aquifer the PRESTO routines will be used to model movement in the biosphere and exposure to man.

The additions being made to PRESTO will permit calculation of risks from all eight low-level waste disposal options under study in a manner which is consistent. The approach will make maximum efficient use of the existing PRESTO, while allowing the distinguishing features of the options to be modeled.

#### REFERENCES

1. Little, Craig A., D.E. Fields, D.J. Emerson, and G. Hiromoto, "PRESTO: A Low-Level Waste Environmental Transport and Risk ASsessment Code--Methodology and User's Manual," Oak Ridge National Laboratory, Draft, August 1, 1982.



NEVADA TEST SITE  
GREATER CONFINEMENT DISPOSAL FACILITY\*

August 1982

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\*Work supported by the U. S. Department of Energy Nevada Operations Office Health Physics Division under Contract No. DE-AC08-76NV00410.

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ABSTRACT

The Greater Confinement Disposal Facility (GCDF) Project at the Nevada Test Site (NTS) is presented in four phases. Phase one discusses why the NTS chose to study greater confinement disposal (GCD) and why an augered shaft was selected from among alternatives. The FY-82 accomplishments of the project are also highlighted. Phase two represents the actual implementation of the project by explaining the six steps required to construct the GCDF. Phase three provides an overview of projected FY-83 activities. Phase four compares confinement and costing for various methods of waste disposal to illustrate why greater confinement is effective. The shallow test plot/GCDF relationship is also introduced.

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INTRODUCTION

To bring greater confinement into an accurate perspective, some of the history and alternatives behind the concept will be presented first. Historically, back in the days of NUREG-0456, intermediate depth burial (IDB) was contrasted to shallow land burial (SLB) as a means for reducing the number and severity of potential pathways of human exposure to radiation.

In Nevada, it is recognized that IDB would be suitable to the Nevada Test Site (NTS) as well as an ideal method to provide greater confinement for leaking tritium waste and for high-specific activity (HSA) low-level waste (LLW). This waste, because of the large quantity of radioactivity (with a half-life of <100 years), continues to present a hazard even after institutional controls on a LLW site have ceased.

Early in 1981 (February 3, 1981), the lead office for the LLW Management Program recommended a name change (IDB to Greater

Confinement Disposal [GCD]). Greater confinement was based on performance guidelines and design considerations. The degree of confinement of GCDF provides compliance with these performance requirements.

Prior to the name change, other alternatives were considered:

- A deep trench — which was suitable to large quantities of low specific activity (LSA) waste. But the NTS SLB was adequate for LSA waste.
- A large borehole — which was suggested by B. Church, DOE/NV. This concept would be ideal for the HSA-LLW and perhaps suitable for transuranics.

Meanwhile Ford, Bacon and Davis Utah (FBDU) had been grinding out theoretical radionuclide migration rates, which tend to support GCDF containment of limited quantities of transuranics.

The next effort was to begin the project, construct the shaft, and instrument the surrounding environment to detect migration.

WHY DID NTS DECIDE TO STUDY GREATER CONFINEMENT DISPOSAL?

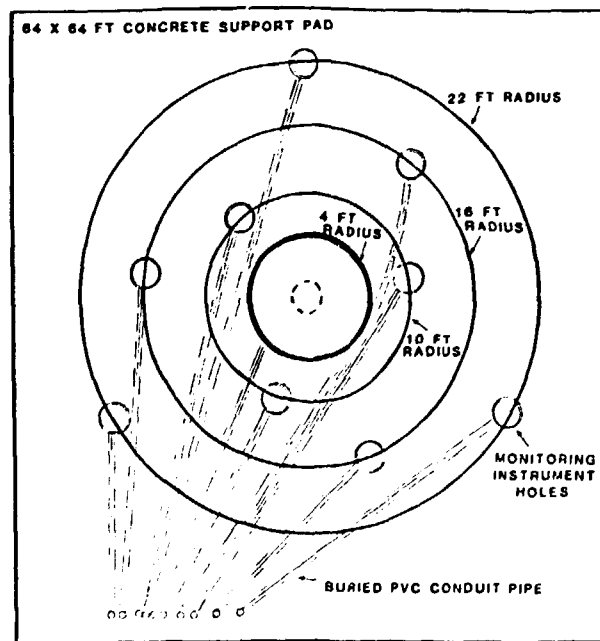
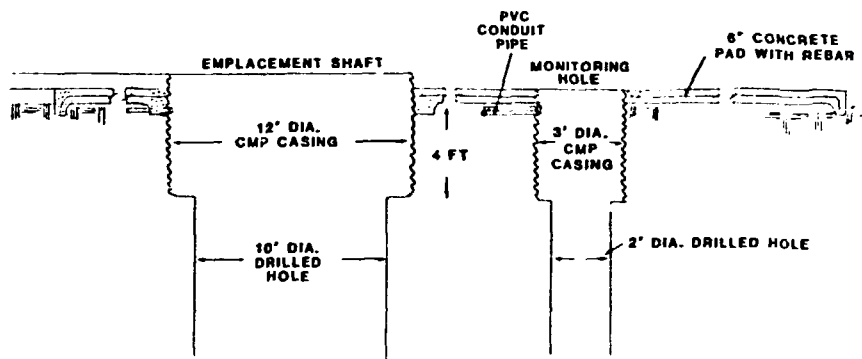
- Operational concerns over leakage from high concentration tritium wastes
  - Need to find alternative methods for handling and disposing of non-contact handled wastes
- 

WHY USE AN AUGERED SHAFT FOR GCDF?

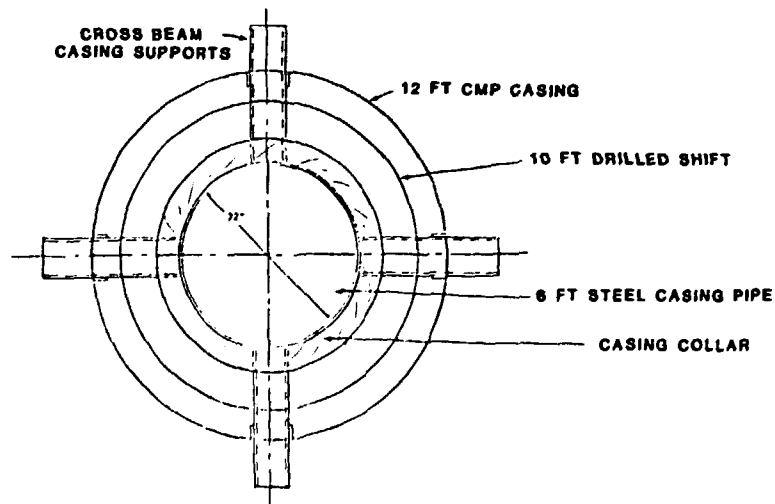
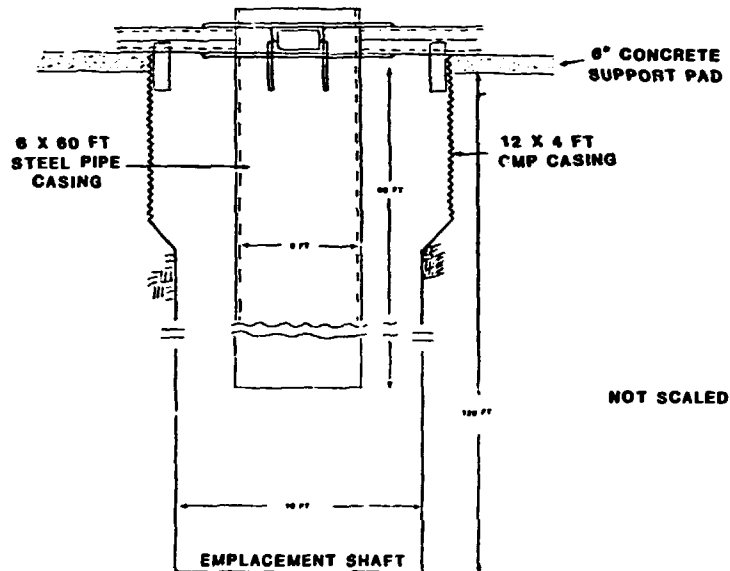
- Most cost effective design for obtaining 120 feet disposal depth for demonstration
- Radiological safety in handling hot sources
- Simplified design of instrument monitoring system

## GCDF FY-82 ACCOMPLISHMENTS

- Completed design and majority of construction of GCDF
- Augered and instrumented emplacement shaft and one monitoring hole (four more will be drilled this year)
- Prepared specifications and bid waste handling system
- Completed safety assessment documents



DESIGN OF GCDF SUPPORT PAD AND DRILLED HOLES



DESIGN OF EMPLACEMENT SHAFT CASING

## GCDF CONSTRUCTION PROJECT AT THE NTS, FY-82

## ● PREPARATION OF THE GCDF SUPPORT PAD

- The area was surveyed to locate monitoring and emplacement holes.
- Conduit pipes were installed to house instrumentation.
- Four-foot long corrugated metal pipes were installed on all holes to prevent surface sluffing.
- The pad was formed, then 3/4 inch rebar on 6-inch centers was laid inside.
- Six inches of concrete were poured into the 64- by 64-foot support pad.

## ● DRILLING OF THE BOREHOLES

- A 10-foot diameter, 120-foot deep emplacement shaft was auger drilled.
- One monitoring hole, 26 inches wide by 120 feet deep, was drilled. (Four more will be drilled this year, leaving four to be drilled next year.)
- A 20-foot deep, 10-foot diameter shallow test plot was located nearby.

## ● ASSEMBLY OF THREE INSTRUMENT STRINGS

- Instrumentation was attached manually to the 120-foot strings.
- Sixteen sampling units were prepared per string. Each unit contained 2 thermocouples, 2 thermocouple psychrometers and a soil atmosphere sampler.
- Packages were taped together, then insulated for 50°C.
- Horse hair deflectors above each delicate instrument package prevented breakage during backfilling operations.



- INSTRUMENTATION EMPLACEMENT

- Instrument lines were attached to the legs of an instrumented tripod, then raised by the crane hook.
- Guide ropes were used to prevent strings from rotating or tangling.
- The tripod and three strings were lowered downhole.

- EMPLACEMENT OF THE STEEL CASING WITH CROSS-BEAM SUPPORTS

- A 60-foot long, 6-foot diameter pipe of 3/4 inch steel was used.
- The casing was raised to a vertical position, manually centered over the hole, then carefully lowered downhole.

- BACKFILLING OF THE GCDF

- A hopper was used to control the slow release of soil downhole to prevent damage to delicate instrumentation.
- Backfill material was alluvial fines.
- The stemming rate was about a foot per hour (16 feet filled in 12 hours).

#### GCDF FY-83 ACTIVITIES

- Complete drilling and instrumentation of monitoring holes
- Install waste handling system and emplace wastes
- Complete construction of facilities
- Begin experiments

CONFINEMENT FACTORS

WASTE CHEMISTRY x

WASTE FORM x

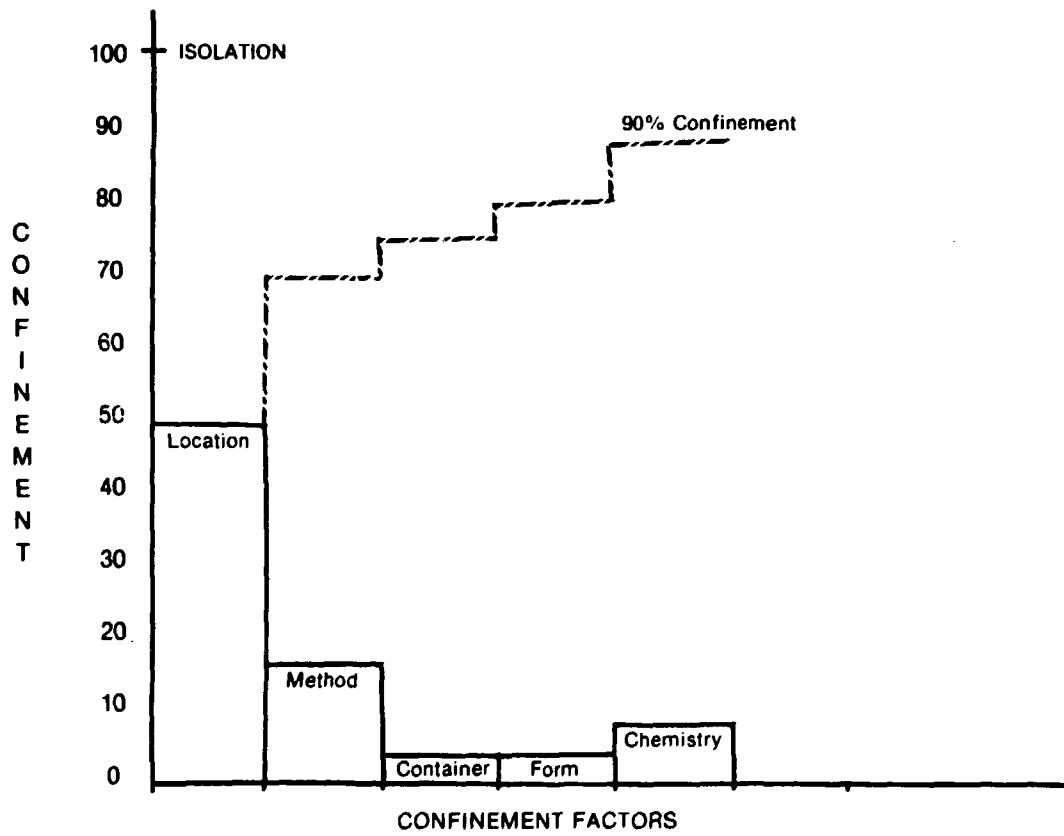
WASTE CONTAINER x

DISPOSAL METHOD x

DISPOSAL LOCATION =

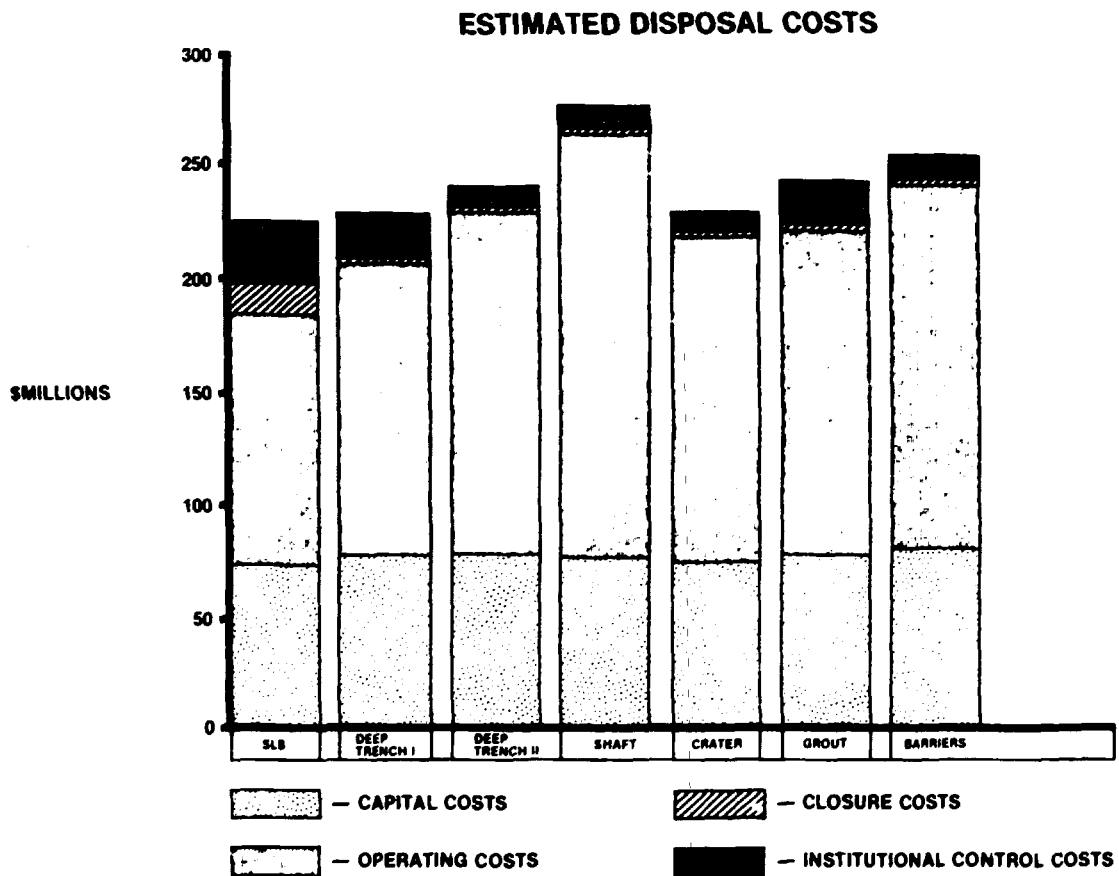
CONFINEMENT

### HYPOTHETICAL CONFINEMENT CHART



## COST ESTIMATE FACTORS

- CAPITAL COSTS — siting, design, and initial construction
- OPERATIONAL COSTS — receipt and disposal of waste and construction of disposal trenches/cells
- CLOSURE COSTS — decommissioning and stabilization
- INSTITUTIONAL CONTROL COST — long term care and monitoring



## DISPOSAL COST ESTIMATE SUMMARY

|                       | <u>\$/ft<sup>3</sup></u> | <u>SLB<br/>Δ %</u> |
|-----------------------|--------------------------|--------------------|
| ● Shallow Land Burial | 6.31                     | 0                  |
| ● Deep Trench — I     | 6.36                     | >0.1               |
| ● Deep Trench — II    | 6.66                     | 6                  |
| ● Augered Shaft       | 7.80                     | 24                 |
| ● Deep Crater*        | 6.37                     | >0.1               |
| ● Grouted Trench      | 6.88                     | 9                  |
| ● Engineered Barriers | 7.18                     | 14                 |
| ● Mined Cavity        | 23.80                    | 377                |

\*Major costs occur during final backfilling which are not included in estimate.

# WHY GREATER CONFINEMENT?

- Reduces potential for human and biological intrusion
- Increased disposal limits
- Cost effective alternative to waste processing/form for most LLW
- Cost savings for long term care and closure

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## NTS/LANL COOPERATIVE SHALLOW TEST PLOT EXPERIMENTS

- Simulate GCDF experiments at shallow (20 feet) depth
- Calibrate instruments and tracer experiments
- Refine computer model analyses



*Aug*

## TECHNICAL SUPPORT NON-SLB(GCDF)

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## ABSTRACT

The Los Alamos National Laboratory is providing technical support for the GCDF at the Nevada Test Site (NTS). This technical support consists of computer modeling of the Greater Confinement Demonstration Facility (GCDF), design and emplacement of a Shallow Test Plot at NTS, and instrument testing at Los Alamos. Results to date on the computer modeling and the Shallow Test Plot are described.

## INTRODUCTION

The overall objectives of this task are to provide technical support, based on the extensive and comprehensive research program at the Los Alamos National Laboratory on the science and engineering of Shallow Land Burial(SLB), and to help develop the technology and documentation needed to open a site providing greater confinement than SLB. The technical support provided will be to (1) assist in the design and modeling of the Greater Confinement Demonstration Facility (GCDF) at the Nevada Test Site (NTS), (2) design experiments to evaluate the instrumentation proposed for the GCDF, (3) assist in testing and evaluation of the instrumentation, (4) provide input into the program based on experience with SLB, (5) perform extensive modeling efforts based on work done in other parts of the Low-Level Waste Management Program (LLWMP), (6) document SLB technology and procedures applicable to greater confinement, and (7) in later fiscal years assess the engineering, cost, and benefits of technologies available for greater confinement. During FY-1982 this technical support has been concentrated in three subtasks:

(1) Computer modeling to aid in the design of, and the analysis of results from, the GCDF,

(2) Design and joint emplacement and operation with REECO of a Shallow Test Plot (STP) at the NTS, and

(3) Design of an instrument test and calibration facility on one of the caissons in the Experiment Clusters in the Los Alamos Experimental Engineered Test Facility.

The Los Alamos role is to provide technical support only. The GCDF design and concept is due to REECO and its subcontractors. This paper will describe the results obtained and the work performed for subtasks 1 and 2 during FY-1982.

## FISCAL YEAR 1982 ACCOMPLISHMENTS

## COMPUTER MODELING OF THE GCDF

The computer modeling was done by Dr. Brian J. Travis of Group ESS-5 of the Los Alamos National Laboratory using the WAFE code. This same code is being used for the subsurface calculations for the SLB Technology-Arid and the Remedial Action Technology-Arid tasks being performed for the LLWMP by the Los Alamos National Laboratory. A brief description of the WAFE code is given in Table 1.

TABLE 1

## WAFE

- MASS AND HEAT TRANSPORT IN POROUS MEDIA
- MULTI-PHASE, MULTICOMPONENT FLOW
- TIME PLUS TWO DIMENSIONS
- NOT RESTRICTED TO DARCY FLOW
- VARIABLE MATERIAL PROPERTIES
- TIME DEPENDENT SOURCES, SINKS, BOUNDARY CONDITIONS
- BUOYANCY
- NUMERICALLY IMPLICIT

TABLE 2

## MATERIAL PROPERTIES USED IN WAFE CALCULATION

|                                  | Alluvium           | Backfill           | Bulk Waste        | Tritium Waste     | Source            |
|----------------------------------|--------------------|--------------------|-------------------|-------------------|-------------------|
| Permeability<br>(Darcys)         | 10                 | 2                  | 0.001             | 0.001             | 0                 |
| Porosity                         | 0.35               | 0.38               | 0.05              | 0.05              | 0.75              |
| Gr. Density (g/cm <sup>3</sup> ) | 2.50               | 2.50               | 1.50              | 2.00              | 8.90              |
| Particle Size (cm)               | 0.1                | 0.1                | —                 | —                 | —                 |
| Thermal Cond.<br>erg/cm/°C/s)    | $7.5 \times 10^4$  | $7.5 \times 10$    | $1.0 \times 10^4$ | $2.5 \times 10^4$ | $3.6 \times 10^7$ |
| Specific Heat<br>(erg/g/°C)      | $8.56 \times 10^6$ | $8.56 \times 10^6$ | $1.0 \times 10^7$ | $1.2 \times 10^7$ | $4.0 \times 10^6$ |
| Initial Conditions               |                    |                    |                   |                   |                   |
| Temperature (°C)                 | 20                 | 20                 | 20                | 20                | 20                |
| Saturation (%)                   | 30                 | 10                 | 0                 | 0                 | 0                 |
| Pore Pressure (kPa)              | 90                 | 90                 | 90                | 90                | 90                |
| Heating Rate (watts)             | 0                  | 0                  | 0                 | 0                 | 1000              |

Because of the proposed waste content for the GCDF (heat generating capsules, tritium wastes, and other bulk wastes) and the proposed use of gaseous tracers, the temperature effects of the waste loading and the effects of the temperatures resulting from the waste on the movement of water and on the proposed instrumentation must be determined. To do these calculations, the waste forms, content, bulk properties, and loading configuration must be accurately known. In addition, the properties of the backfill material and of the host material must be known. Several sets of calculations have been performed to help specify the waste loading configuration. In this paper only the results from the last iteration will be shown.

Figure 1 shows the GCDF experiment geometry for which calculations will be reported. A heat source, assumed to be 1000 watts is placed at the -105 foot level. The heat source is assumed to be placed in a 55-gallon drum packed with copper turnings to dissipate the heat evenly over the entire surface area of the drum. This configuration for the heat source was chosen because previous calculations had indicated that the interface temperature between a 1000-watt capsule and the backfill material would be extremely high. Five feet of backfill is assumed over the heat source and then a 10 foot layer of bulk wastes is assumed to be emplaced. Five feet of backfill is placed over the bulk wastes and then a 10 foot layer of tritium waste is assumed to be placed in the facility, which is then backfilled to the surface. The material properties used in the calculations are shown in Table 2. The change in permeabilities with saturation for liquid and gaseous water is shown in Figure 2.

With the configuration, loading, and heat sources shown in Figure 1, after 100 months, the temperature at the drum-backfill interface rises to a maximum of about 360°C. One foot inside the backfill-alluvium interface the maximum temperature reached is about 115°C, while 1 foot into the alluvium from the backfill-alluvium interface the maximum temperature reached is about 80°C. These maximum temperatures will occur at the same depth as the heat source. After 100 months the temperature at the drum-backfill interface has practically stopped rising while the temperatures just inside the backfill and the alluvium are still rising. These calculations assume that the drum containing the sources reaches a uniform temperature and that the material properties are as indicated in Table 2.

The WAFE code also follows the movement of the water in the system both in the liquid and vapor phases. Figures 3 through 9 show the results of the calculations at approximately 12-month intervals for times up to 103 months. Each figure is a contour plot of saturation and temperature at the time indicated on the figures. The key to the contours is contained in Table 3. Note that with time the heat from the source is gradually spreading out and away from the source and after 103 months has reached the ground surface of the GCDF. As mentioned above, after 103 months the temperature away from

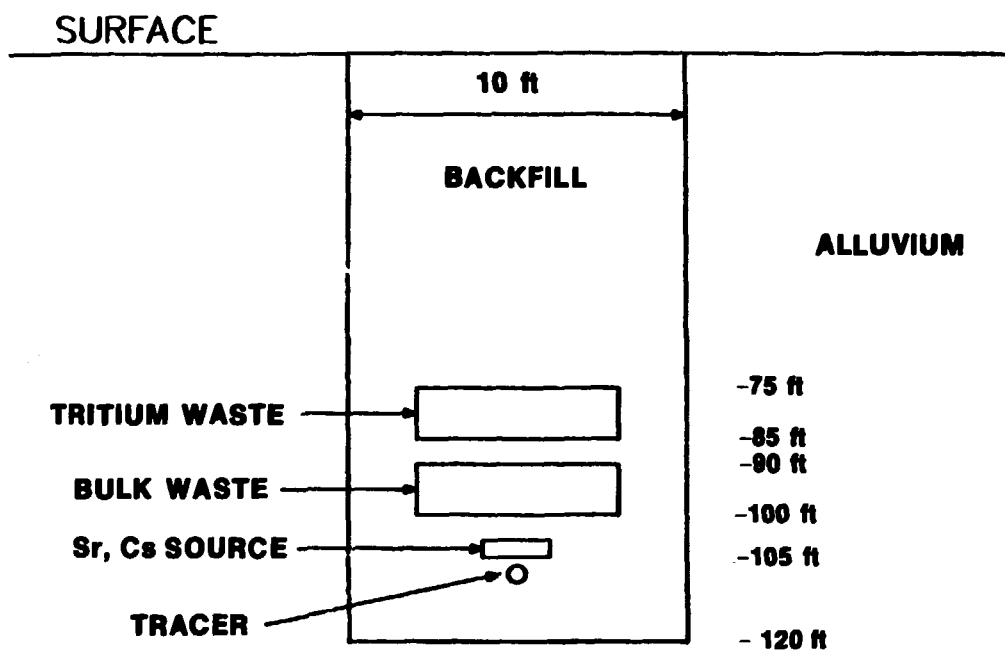


**FIGURE 1. GCDF EXPERIMENT GEOMETRY**

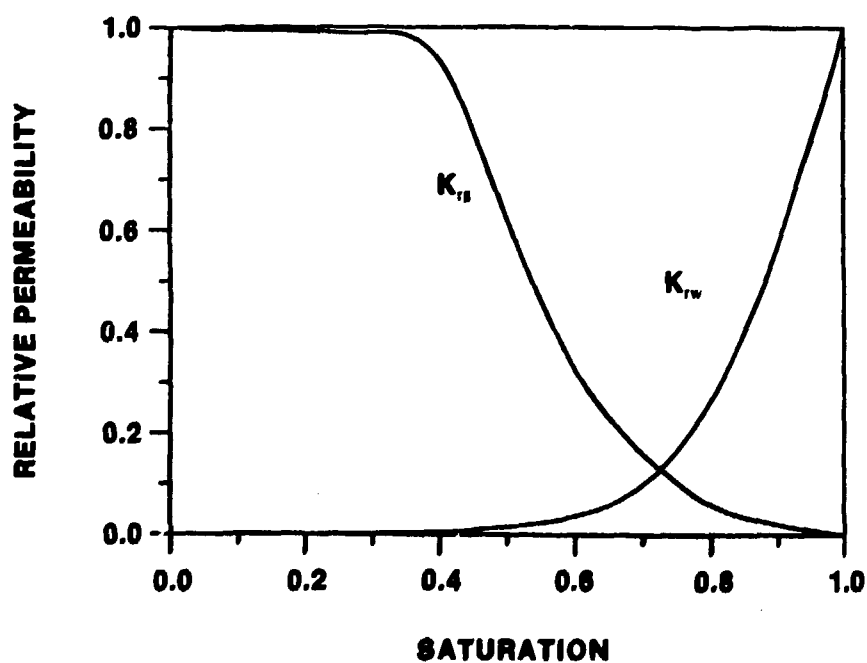


**FIGURE 2. CHANGE IN PERMEABILITIES WITH SATURATION**

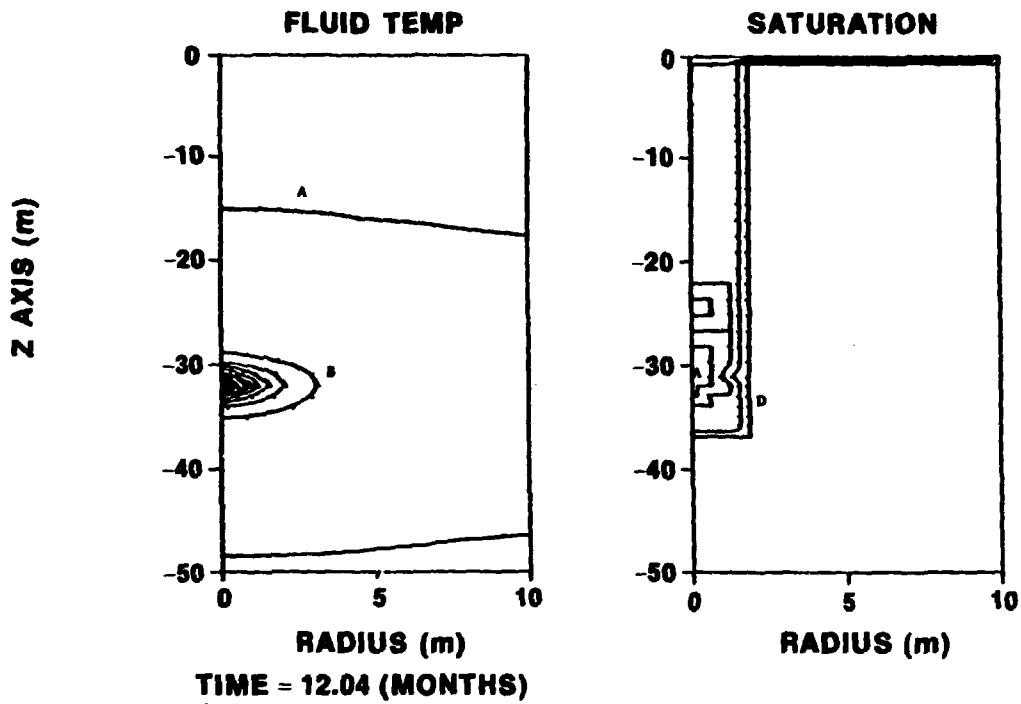


FIGURE 3. TEMPERATURE AND SATURATION AFTER 1 YEAR

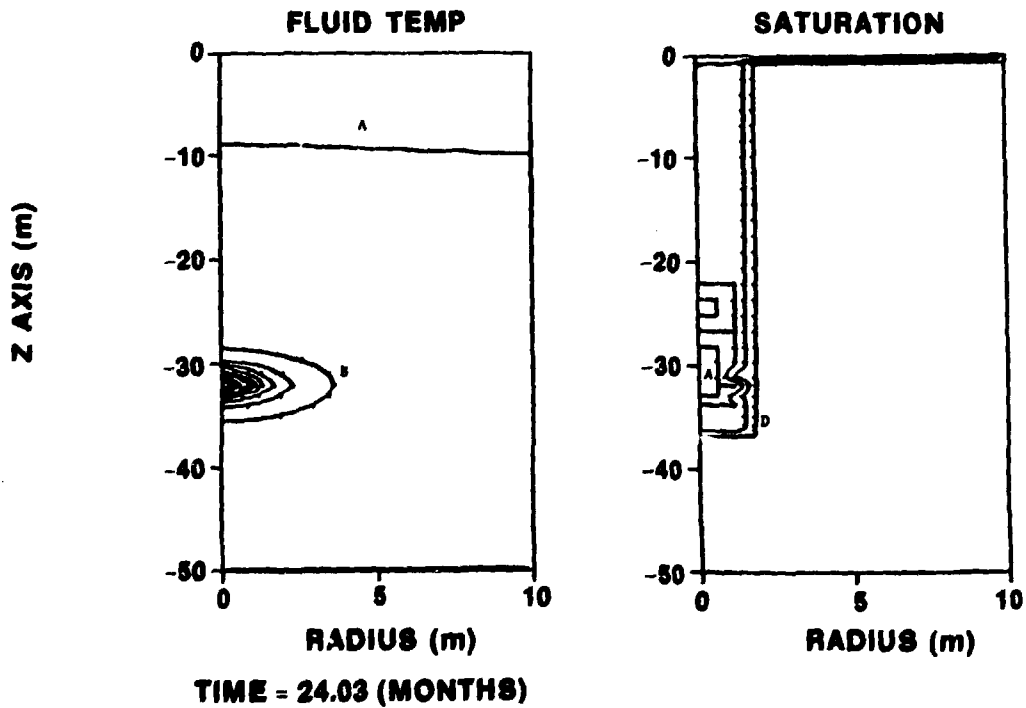


FIGURE 4. TEMPERATURE AND SATURATION AFTER 2 YEARS

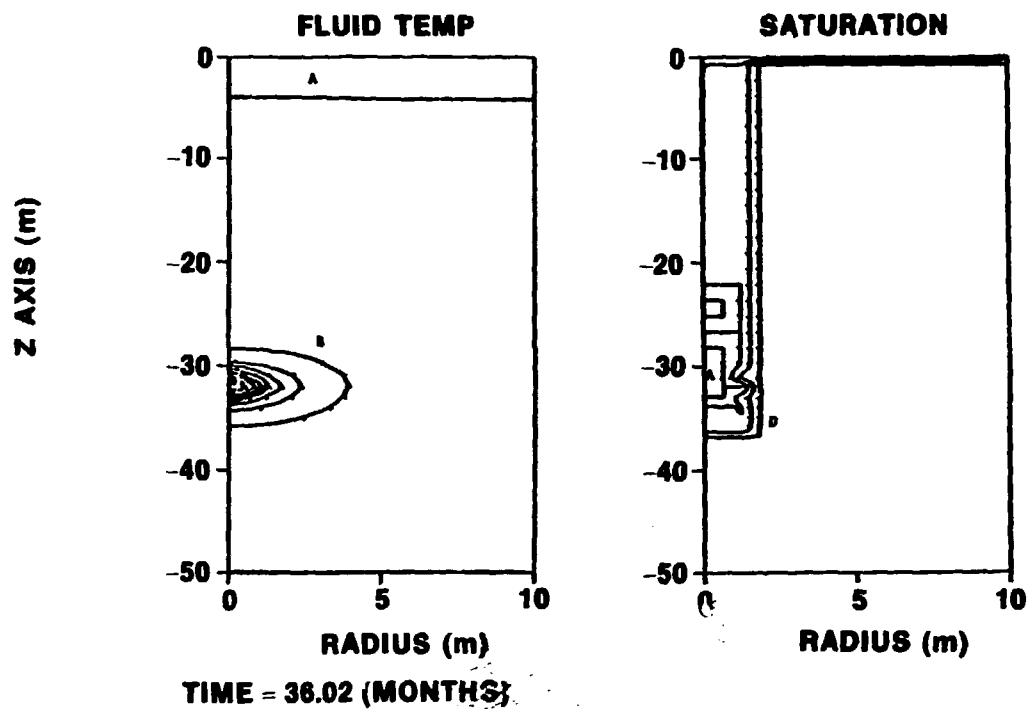


FIGURE 5. TEMPERATURE AND SATURATION AFTER 3 YEARS

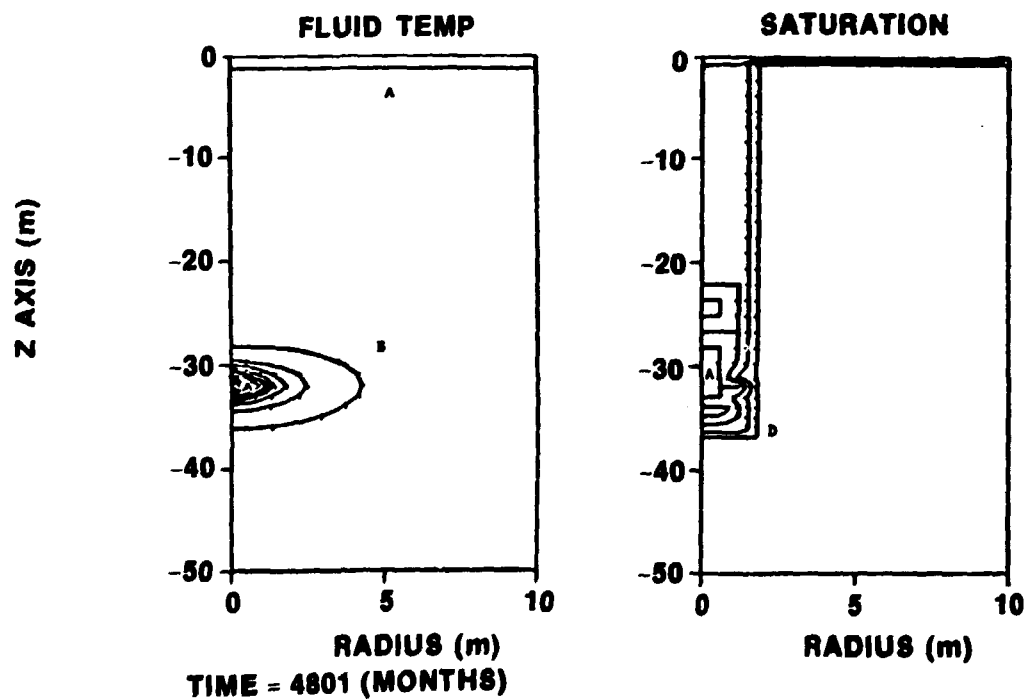


FIGURE 6. TEMPERATURE AND SATURATION AFTER 4 YEARS

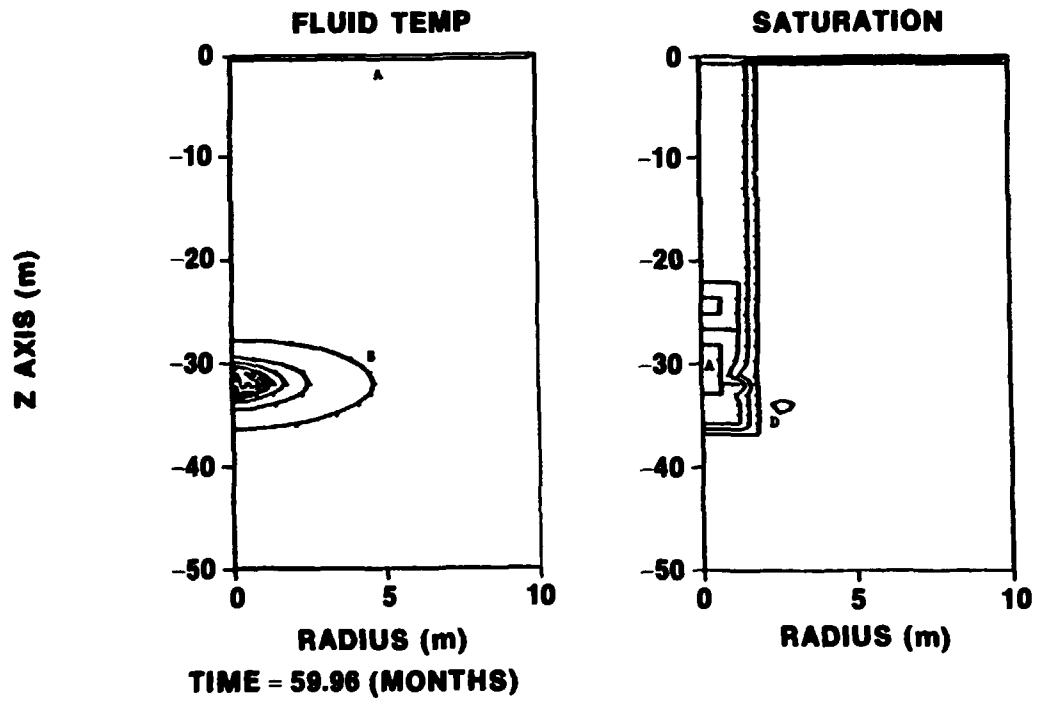


FIGURE 7. TEMPERATURE AND SATURATION AFTER 5 YEARS

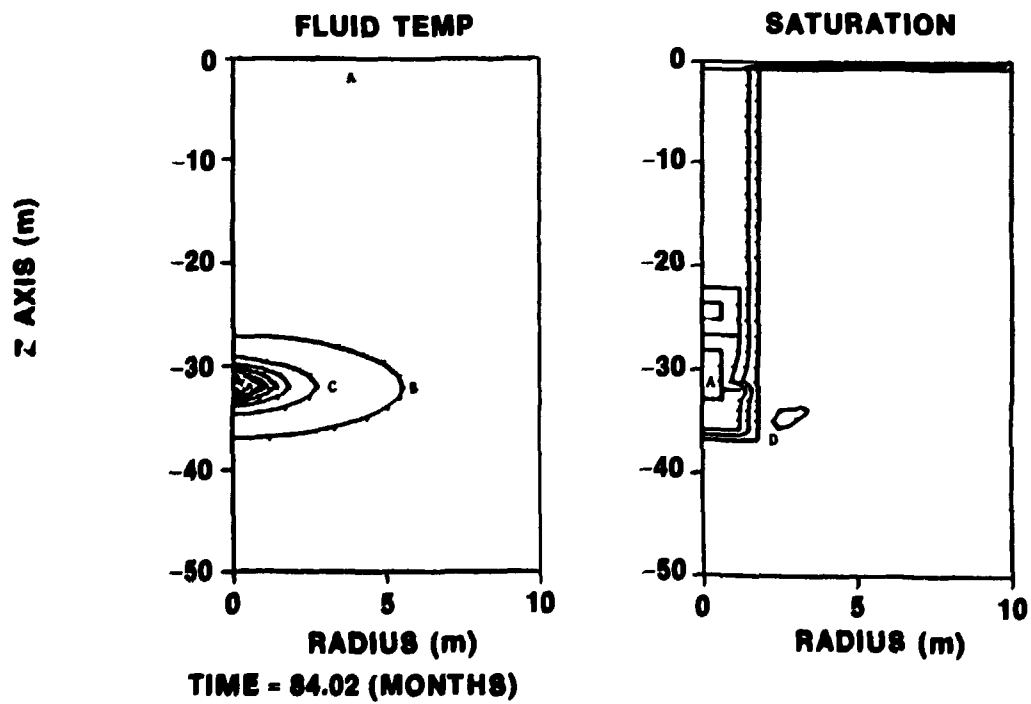


FIGURE 8. TEMPERATURE AND SATURATION AFTER 7 YEARS



**TABLE 3**  
**KEY TO FIGURES 3 - 9**

| <u>Contour ID</u> | <u>Temperature (°C)</u> |
|-------------------|-------------------------|
| A                 | 20                      |
| B                 | 40                      |
| C                 | 60                      |
| D                 | 80                      |
| E                 | 100                     |
| F                 | 125                     |
| G                 | 150                     |
| H                 | 200                     |
| I                 | 250                     |
| J                 | 300                     |
| K                 | 400                     |
| L                 | 500                     |

**SATURATION VALUES FOR SATURATION CONTOUR PLOTS**

|   |      |
|---|------|
| A | 0.00 |
| B | 0.10 |
| C | 0.20 |
| D | 0.30 |

the source-backfill interface is still rising. From Figure 9, which is for 103 months, the effects of the temperature on the moisture can be most readily seen. Note that the moisture is driven away from the vicinity of the source into the alluvium; the moisture is not only driven from the backfill around the emplaced wastes but is driven into the alluvium surrounding the GCDF. This is not surprising because the temperature near the backfill-alluvium interface reaches about 115°C after 103 months.

It must be emphasized here that these calculations were based on the material properties listed in Table 2 and on the relative permeability curves shown in Figure 2. When more reliable data are obtained, either from laboratory work or from the Shallow Test Plot at NTS, these calculations can be redone to more accurately reflect the materials used in the GCDF. The complexity of the problem is such that a code like the WAFE code is probably required for calculations relative to the GCDF and possible contaminant transport out of this type of a facility.

#### INSTRUMENT TESTING AT NTS (SHALLOW TEST PLOT)

The purpose of the GCDF is to demonstrate disposal of waste types not suitable for SLB in such a manner as to provide greater confinement than SLB. To document this greater confinement a reference set of data on SLB for similar wastes must be obtained. Although these wastes will not be disposed of in this Shallow Test Plot, the temperature effects resulting from the sources can be simulated. Also, tracers to simulate the possibility of waste components escaping from these waste types can be injected near the heat source and their movement followed in a manner similar to that proposed for the GCDF. In addition, the effects of the temperatures on the proposed instrumentation can be determined in a Shallow Test Plot. Instrument calibration under field conditions will be performed under more closely controlled conditions in subtask 3 at Los Alamos. Also, experiments in a Shallow Test Plot can be used to validate the model used in subtask 1.

Figure 10 is a drawing of the Shallow Test Plot emplaced at NTS. It is the same diameter as the GCDF but is only 20 feet deep. It is located within 150 feet of the GCDF. The corrugated metal pipe used in the drilling operation was left in place and extends about 3 feet down into the hole. Instrument strings were as shown on the figure. Instrument clusters are in 4 places starting at the bottom of the hole at 5-foot increments, in addition to being placed in the center of the hole directly at the bottom. Each instrument location consists of a thermocouple, a thermocouple psychrometer, and a soil atmosphere sampler. The instruments used in this STP are identical to those used in the GCDF.

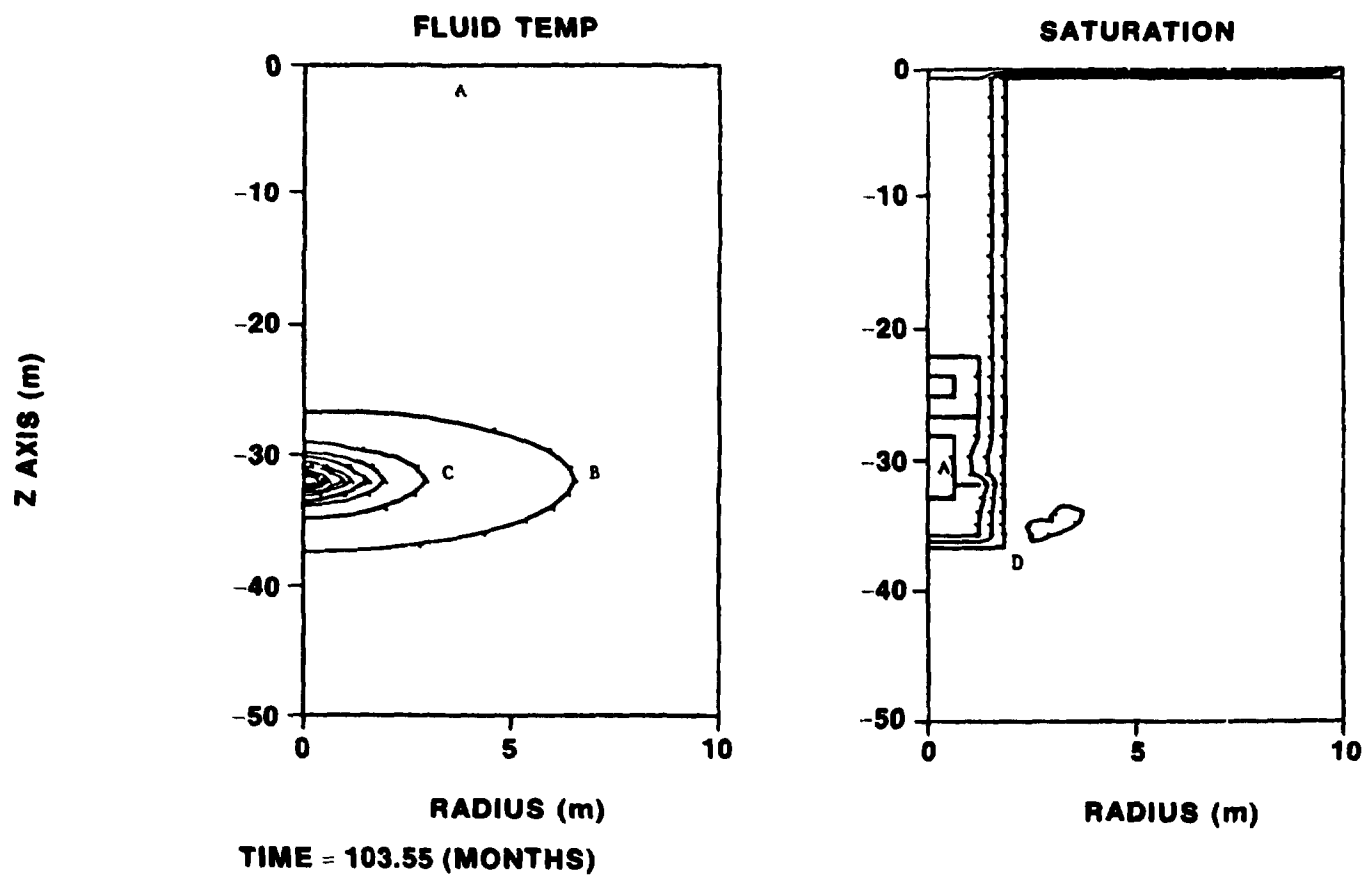


**FIGURE 9. TEMPERATURE AND SATURATION AFTER 103 MONTHS**

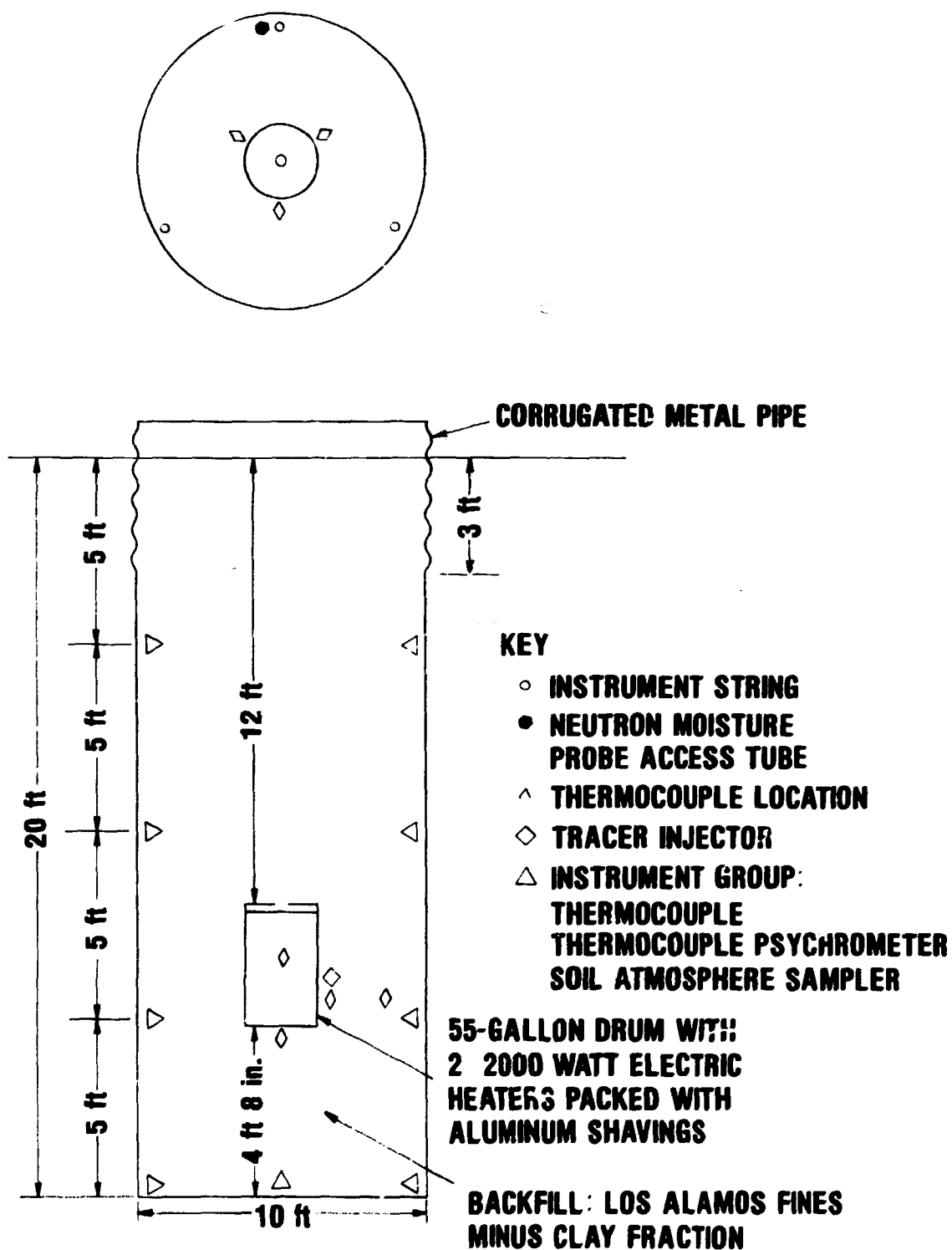


**FIGURE 10. SHALLOW TEST PLOT AT NT8  
JOINT LOS ALAMOS AND REECO EXPERIMENT**

The heat source is simulated in this experiment as closely as possible to that being considered for the GCDF. Two 2000-watt electric heaters were placed inside the drum with the electrical connections placed on the top of the drum. The drum is filled with aluminum shavings to distribute the heat load over the entire surface of the drum, in a manner similar to the copper shavings that will be used in the GCDF.

Calibrated thermocouples were equally spaced on the sides of the drum in three places about half way between the top and bottom of the drum. One additional thermocouple was placed on the bottom of the drum. Asbestos-covered wire was attached to the electrical leads on the top of the drum and combined with the thermocouple wires into a harness that goes to the surface of the experiment. Two heaters were placed in the drum for redundancy; if one heater fails or if a lead wire is broken, there will be a back-up heater. The electrical connections to the heaters were protected by welding another lid from a 55-gallon drum about 4 inches above the top of the drum and bringing all the wires out through a hole in the upper lid (Fig. 10).

To determine the temperature gradient in the backfill, two thermocouples were placed between the drum and the side of the hole (Fig. 10). In addition, at this elevation a soil atmosphere sampler was installed to be used in a reverse fashion as a tracer injector port.

This Shallow Test Plot was emplaced at NTS during the first week of June of 1982 by personnel from Los Alamos and from REECO. After emplacement all instruments were checked for continuity and performance and the continuity of the heaters was checked. Except for one thermocouple psychrometer, which was bumped by a ladder during the installation of the drum, all instruments survived the emplacement procedure even though protective cones were not placed over the instruments on the instrument strings. Base line measurements with the thermocouple psychrometers and thermocouples are being made by REECO personnel.

Emplacement of the Shallow Test Plot at NTS completes all Milestones in this task for FY-1982.

#### FUTURE WORK

Future work will be in all of the subtasks listed above and in some additional areas.

#### COMPUTER MODELING OF THE GCDF

As more and better data become available on material properties, waste forms, waste form properties, and waste loading in the GCDF and as data are obtained from the Shallow Test Plot, more computer

simulations of the GCDF will be performed. The WAFE code will be used to model the subsurface portion of the GCDF; CREAMS will be used to model the surface component of the GCDF, and the two models will be combined to completely model the GCDF.

#### INSTRUMENT TESTING AT NTS (SHALLOW TEST PLOT)

A test plan for tracer injection and monitoring, heating of the source, and adding water to the STP is in preparation. When the instrumentation is completed at NTS, these experiments will be performed to provide the data required on the temperature effects on the instruments and the moisture and tracer movement. These data will be used to validate both WAFE and CREAMS.

#### INSTRUMENT TESTING AT LOS ALAMOS

A carefully constructed and instrumented experiment will be constructed in one of the caissons in the Experiment Clusters at Los Alamos. The instruments will be calibrated and tested for performance, reliability, and longevity at temperature and soil moisture ranges expected in the GCDF. This experiment will also provide data for model validation.

#### ACKNOWLEDGMENTS

Technical discussions with J.G. Steger and R.K. Schulz are gratefully acknowledged. The support and help of E.W. Kendall, P.T. Dickman, E. Williams, and J.J. Brown of REECO during the emplacement of the Shallow Test Plot is gratefully acknowledged.

## GREATER CONFINEMENT DISPOSAL

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## ABSTRACT

Progress has been made in three areas related to Greater Confinement Disposal. A large lysimeter has been constructed to study clay caps and liners, SRP low level waste is being characterized based on activity level, and potential sites for GCD on the Savannah River Plant are being evaluated.

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INTRODUCTION

The long-term consequences of past and current low level waste burial practices are being studied by SRL. These studies show that employment of improved practices for burial of certain wastes is desirable to achieve greater protection against vegetative uptake and erosion, and to avoid the necessity for future remedial action to decommission the burial site. Estimates indicate that the current SRP burial ground will be filled by about 1990. Development and adoption of improved practices in the next-generation burial ground is planned. Current development effort is focused on clay caps and liners for inhibiting infiltration of rainwater, characterization of SRP wastes based on activity level to facilitate greater confinement for a portion of the wastes, and selection of a proposed site (or sites) for the new burial ground.

Humid Site GCD Test: Greater confinement at humid sites will generally require low permeability materials to minimize water intrusion into wastes, and placement of the waste at an intermediate depth below the root zone and above the water table. This concept is being tested in a landfill lysimeter which is a 1/10 linear scale model of one grid of a proposed landfill system for disposing of low level radioactive salts. The landfill will meet RCRA requirements for impermeable barriers and the intent of 10CFR61 for intruder protection. These salts are a side-stream from a process planned at the Savannah River Plant for solidification of high level waste in glass. The salts will be

incorporated in Portland cement and buried in an engineered landfill. In the landfill the solidified waste product (called "saltcrete") will be buried in such a manner as to inhibit leaching and migration of chemical salts (primarily nitrate) and the trace levels of radionuclides which are still present. The average chemical composition of the solution to be mixed with cement is shown in Table 1 and the radionuclide composition is shown in Table 2.

A lysimeter has been designed and is nearly complete (Figure 1) for testing a low permeability cap and liner system for inhibiting leaching. The lysimeter is lined with Hypalon. Eight saltcrete blocks (2' x 2' x 25") formed by blending simulated (i.e., non-radioactive) waste solution with cement were poured in the lysimeter. Sumps are provided for sampling leachate collected by both the clay liner and by the underlying Hypalon liner.

Soil moisture samplers are placed in the soil between saltcrete blocks and conductivity probes are used to indicate the presence of any salt. Thermocouples were used during pouring of the saltcrete to determine the maximum temperature caused by hydration of the cement.

Figures 2, 3, and 4 show three stages of construction of the lysimeter. Placement of the clay cap and soil overburden and planting of grass remain to be done. At this point collection of data will begin. This technique of using clay to inhibit infiltration of rainwater may also be selected for use with SRP waste forms other than "saltcrete". Studies of the characteristics of other waste forms are underway and are discussed below.

Characteristics of SRP Wastes: Currently, we are investigating how to provide Greater Confinement Disposal (GCD) for the most significant (i.e., containing most of the radionuclides) fraction of SRP wastes. The GCD fraction must be identified and characterized in order to deal with it properly. This section describes briefly a method which has been employed to identify potential GCD waste. Our next step is to characterize the GCD waste physically and then develop an improved burial concept based on greater confinement.

SRP waste burials are recorded in a computer system called COBRA (Computerized Burial Records Analysis). The COBRA records provide a basis for defining the waste. Each waste receipt is accompanied at the burial ground with a burial slip in which the volume and curies (or grams) of radionuclides is given, along with whether it is combustible or non-combustible. Also specified is the "variety" of waste that is buried. The "variety" description identifies the type of radionuclide present. There are ten "varieties" of waste currently being buried at SRP (see Table 3). A radioisotope of interest has been identified for each "variety". In some cases the isotope of interest is the same as the name of the "variety", but in others it is different.

The curies of the isotopes of interest were determined by recalling curies for the "variety" in question from the computer files, and then



TABLE 1

CHEMICAL COMPOSITION OF DECONTAMINATED AGED SALT SOLUTION

| <u>Component</u>                                 | <u>Decontaminated<br/>Salt Solution, wt %</u> |
|--------------------------------------------------|-----------------------------------------------|
| H <sub>2</sub> O                                 | 68                                            |
| NaNO <sub>3</sub>                                | 15.9                                          |
| NaNO <sub>2</sub>                                | 3.9                                           |
| NaOH                                             | 4.3                                           |
| Na <sub>2</sub> CO <sub>3</sub>                  | 1.7                                           |
| NaAl(OH) <sub>4</sub>                            | 3.6                                           |
| Na <sub>2</sub> SO <sub>4</sub>                  | 1.9                                           |
| NaF                                              | 0.06                                          |
| NaCl                                             | 0.12                                          |
| Na <sub>2</sub> SiO <sub>3</sub>                 | 0.04                                          |
| Na <sub>2</sub> CrO <sub>4</sub>                 | 0.05                                          |
| NaHgO(OH)                                        | $1.7 \times 10^{-6}$                          |
| NaAg(OH) <sub>2</sub>                            | $1.4 \times 10^{-7}$                          |
| Na <sub>2</sub> MoO <sub>4</sub>                 | 0.009                                         |
| KNO <sub>3</sub>                                 | $8.6 \times 10^{-6}$                          |
| CaSO <sub>4</sub>                                | $2.5 \times 10^{-4}$                          |
| Na <sub>2</sub> C <sub>2</sub> O <sub>4</sub>    | 0.10                                          |
| Na <sub>2</sub> PO <sub>4</sub>                  | 0.13                                          |
| NH <sub>4</sub> NO <sub>3</sub>                  | $6.8 \times 10^{-6}$                          |
| NaB(C <sub>6</sub> H <sub>5</sub> ) <sub>4</sub> | 0.06                                          |
| Other salts                                      | 0.20                                          |

TABLE 2

RADIONUCLIDE COMPOSITION OF DECONTAMINATED SALT SOLUTION

| <u>Radionuclide (t1/2, yrs)</u>               | <u>Supernate, nCi/g<br/>(15-yr aged)</u> |
|-----------------------------------------------|------------------------------------------|
| <sup>3</sup> H (12.33)                        | 13                                       |
| <sup>14</sup> C (5730)                        | 0.009                                    |
| <sup>60</sup> Co (5.27)                       | 0.2                                      |
| <sup>59</sup> Ni (80,000)                     | 0.0002                                   |
| <sup>63</sup> Ni (163)                        | 0.02                                     |
| <sup>79</sup> Se (approx. $6.5 \times 10^4$ ) | 0.3                                      |
| <sup>90</sup> Sr (29)                         | 1                                        |
| <sup>90</sup> Y (3.1 hr) <sup>a</sup>         | 1                                        |
| <sup>99</sup> Tc ( $2.1 \times 10^5$ )        | 30 (3) <sup>b</sup>                      |
| <sup>106</sup> Ru (1.0)                       | 30                                       |
| <sup>106</sup> Rh (2.18 hr) <sup>a</sup>      | 30                                       |
| <sup>125</sup> Sb (2.73)                      | 11                                       |
| <sup>125m</sup> Te (58 da) <sup>a</sup>       | 0.2                                      |
| <sup>126</sup> Sn (approx. $10^5$ )           | 0.2                                      |
| <sup>126</sup> Sb (12.5 da) <sup>a</sup>      | 0.02                                     |
| <sup>126m</sup> Sb (19 min) <sup>c</sup>      | 0.2                                      |
| <sup>129</sup> I ( $1.7 \times 10^7$ )        | 0.2                                      |
| <sup>137</sup> Cs (30.2)                      | 22                                       |
| <sup>137m</sup> Ba (2.5 min) <sup>a</sup>     | 19                                       |
| <sup>147</sup> Pm (2.62)                      | 4                                        |
| <sup>151</sup> Sm (93)                        | 3                                        |
| <sup>154</sup> Eu (8.2)                       | 1                                        |
| <sup>155</sup> Eu (4.76)                      | 0.3                                      |
| <sup>238</sup> Pu (87.7)                      | 0.07                                     |
| <sup>239</sup> Pu (24,000)                    | 0.0007                                   |
| All TRU elements                              | 0.2                                      |

(a) Daughter of preceeding isotope

(b) Concentration in parenthesis applies if <sup>99</sup>Tc removal process is used.(c) Daughter of <sup>126</sup>Sn.

FIGURE 1

## SALTCRETE LANDFILL LYSIMETER

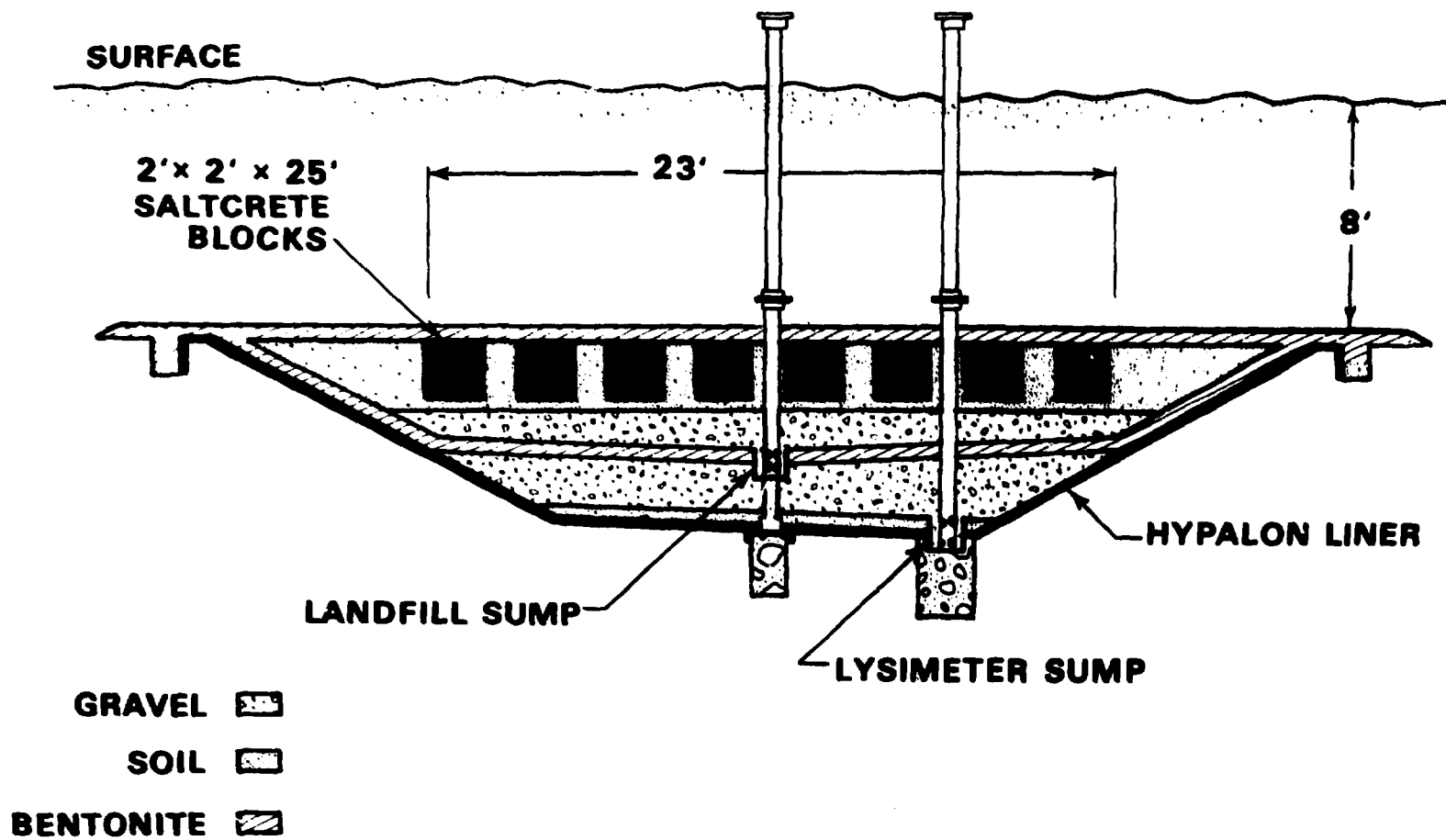




Figure 2 - Saltcrete Lysimeter Excavation

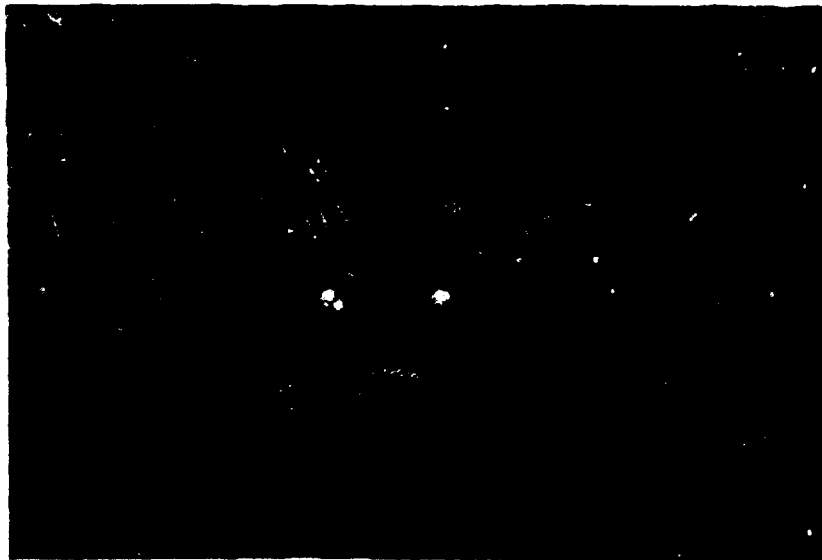


Figure 3 - Saltcrete Lysimeter Liner Installation, Typar® layer (white) covers and protects Hypalon® (black) from damage by gravel bed. Sand layer and gravel being placed.

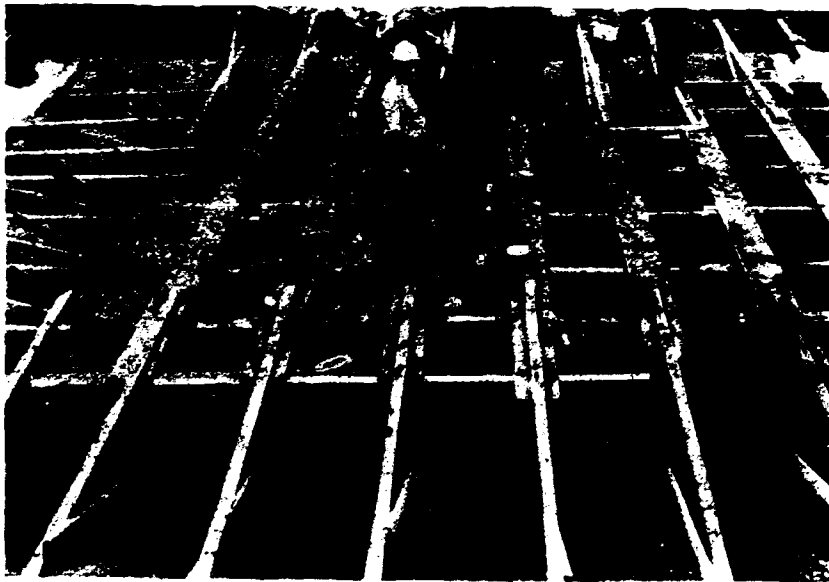


Figure 4 - Saltcrete Monoliths (2' x 2' x 25') being poured in forms.

Table 3. SRP Waste Categories

| "Variety"        | Symbol<br>Used In<br>COBRA | Isotope of Interest and Method<br>for Determining Quantity                                                                                                |
|------------------|----------------------------|-----------------------------------------------------------------------------------------------------------------------------------------------------------|
| Cobalt-60        | Co                         | $^{60}\text{Co}$ . Taken directly from COBRA records as "variety" 1 or "variety" 2.                                                                       |
| Cesium-137       |                            | $^{137}\text{Cs}$ . Taken directly from COBRA records as "variety" 1 or "variety" 2.                                                                      |
| Fission Products | FP                         | $^{90}\text{Sr}$ . Assumed to be 2% of total fission product curies, based on estimate of average content of this waste type.                             |
| Induced Activity | IA                         | $^{63}\text{Ni}$ . IA curies is assumed to be $^{60}\text{Co}$ . $^{63}\text{Ni}$ is assumed to be $0.15 \times ^{60}\text{Co}$ curies, based on DP-1519. |
| Strontium        | Sr                         | $^{90}\text{Sr}$ . Taken directly from COBRA records as "variety" 1 or "variety" 2.                                                                       |
| Tritium          | 87                         | $^3\text{H}$ . Taken directly from COBRA records as "variety" 1 or "variety" 2.                                                                           |
| Enriched Uranium | 20                         | Total Alpha. Curies based on isotopic split of 95% $^{235}\text{U}$ , 4.3% $^{238}\text{U}$ and 0.71% $^{234}\text{U}$ .                                  |
| Natural Uranium  | 81                         | Total Alpha. Curies based on isotopic split of 99.28% $^{238}\text{U}$ , 0.72% $^{235}\text{U}$ , and 0.0054% $^{234}\text{U}$ .                          |
| Plutonium-238    | 83                         | $^{238}\text{Pu}$ . Taken directly from COBRA records as "variety" 1 or "variety" 2.                                                                      |
| Plutonium-239    | 50                         | $^{239}\text{Pu}$ . Taken directly from COBRA records as "variety" 1 or "variety" 2.                                                                      |

calculating the curies of the isotope of interest (if not the same). The calculations were completed for each waste receipt by accessing the computer files. The concentration (microcuries/cc) of the radionuclide of interest was calculated and the data were then sorted on the basis of concentration. The methods used for determining the curies of the "variety" (i.e., the input to COBRA) and the methods used for determining the curies of the isotopes of interest (see Table 3) were not exact, but do serve as a reasonable basis for estimating the effect of classifying waste for GCD.

The study was based on a three year period, 1979 through 1981. This period was chosen because the data are the most recent available and it is expected that future waste generation will be similar. Results from this study indicate that most of the activity in the wastes is contained in a small volume. Limits were assigned for the concentrations of radionuclides above which waste would be placed in GCD and below which waste would be placed in Shallow Land Burial (SLB). These limits were based on our initial judgment of how to isolate the largest amount of activity while constraining the volume to be stored in GCD. Based on our estimates of the quantity of significant radionuclides present, the split between GCD and SLB assigns at least as much of the waste to GCD as the proposed NRC 10CFR61 would. Only 4% of the volume of waste would be assigned to GCD and this would contain 80% of the radioactivity; however, these figures could change as the techniques are refined.

With the use of the proposed SRP low level waste incinerator, the SLB fraction of the waste could be reduced from 1,868,000 to 668,000 cu. ft., containing 44,000 curies. The bulk of the volume is due to the combustible fraction of the waste, whereas, the bulk of the activity is in the non-combustible fraction. Incinerator ash will be placed in either SLB or GCD based on the accepted criteria. A volume reduction factor of 30 was used in the calculations for the incinerator. Table 4 is a summary of the volume and curie splits, including effects of incineration, for the wastes buried between 1979 and 1981.

The COBRA files have also been used to provide information on the source (generator) and type of waste included in the GCD fraction. This information is being used in discussions with the generators, to study the possibility of segregating waste at the source and improving packaging for stabilization and containment.

Site Selection. The 18 sites on the SRP reported earlier\* as potential sites for Greater Confinement Disposal (GCD) have been further evaluated and narrowed down to 12. These twelve are shown on Figure 5. A drilling contract has been let and additional data are presently being acquired on some of these sites so that further comparative evaluations can be made. Parameters were identified that impact significantly on

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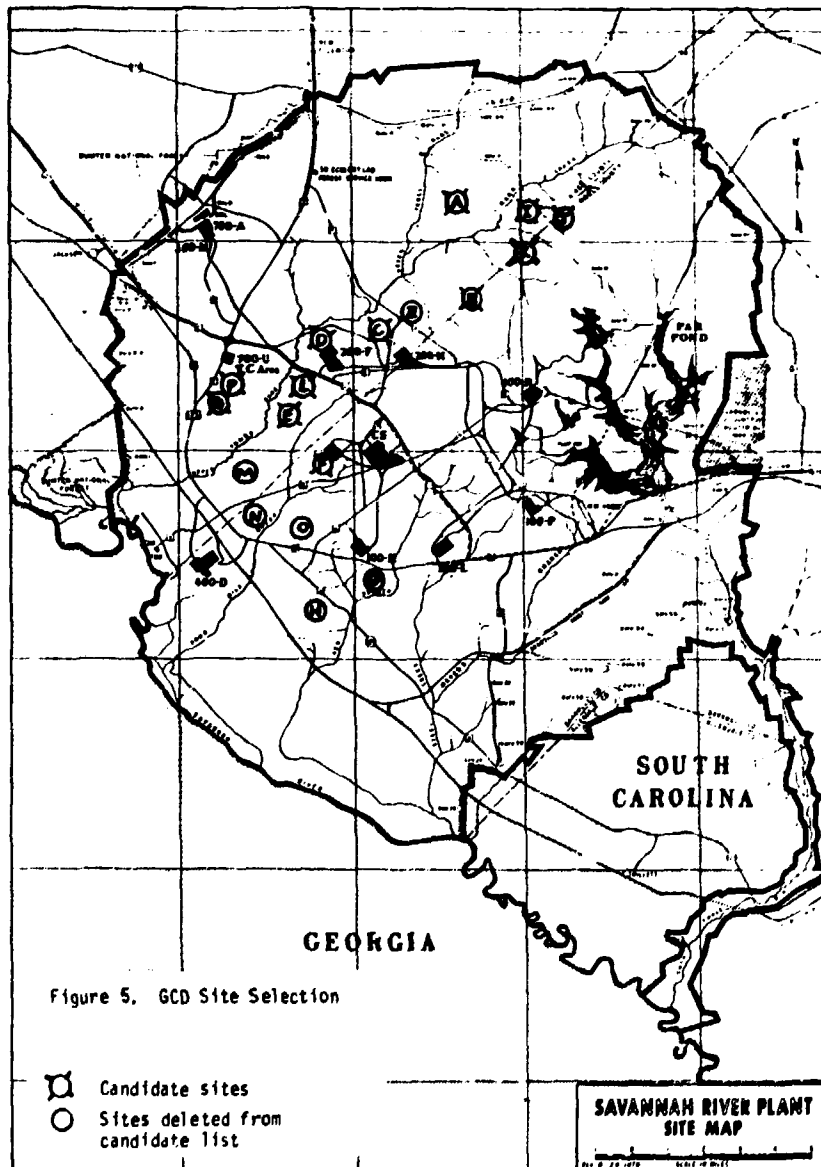
\* Proceedings of the Third Annual Information Meeting, DOE Low Level Waste Management Program, November 4-6, 1981, New Orleans, La., ORNL/NFW-81/34.



Table 4. Consignment of SRP Wastes to GCD or SLB  
Based on Activity Concentration (1979-1981)

|                               |     | <u>Combustible</u><br><u>w/o incin.</u> | <u>Fraction</u><br><u>w/incin.</u> | <u>Non-Comb.</u><br><u>Fraction</u> | <u>Totals</u>     |                 |
|-------------------------------|-----|-----------------------------------------|------------------------------------|-------------------------------------|-------------------|-----------------|
|                               |     |                                         |                                    |                                     | <u>w/o incin.</u> | <u>w/incin.</u> |
| <u>Volume, ft<sup>3</sup></u> |     |                                         |                                    |                                     |                   |                 |
|                               | SLB | 1,241,000                               | 41,000                             | 627,000                             | 1,868,000         | 668,000         |
|                               | GCD | 22,000                                  | 22,000 <sup>1</sup>                | 54,000                              | 76,000            | 76,000          |
|                               |     | <u>1,263,000</u>                        | <u>63,000</u>                      | <u>681,000</u>                      | <u>1,944,000</u>  | <u>743,000</u>  |
| <u>Curies</u>                 |     |                                         |                                    |                                     |                   |                 |
|                               | SLB | 1,120                                   | 1,120                              | 42,800                              | 44,000            | 44,000          |
|                               | GCD | 44,000                                  | 44,000                             | 128,000                             | 172,000           | 172,000         |
|                               |     | <u>45,120</u>                           | <u>45,120</u>                      | <u>170,800</u>                      | <u>216,000</u>    | <u>216,000</u>  |

1 No incineration assumed due to high radioactivity levels.



site selection. These are:

- depth to water table
- distance to waste generator
- distance to streams
- surface topography
- area
- distance to public zone

Table 5 shows the rating factors which were used in narrowing down the number of prospective sites. Depth to water table was not known for all sites, and thus this parameter was not used in this exercise. However, it is estimated that the water table depth is >40 ft. at all of the prospective sites. After the sites were rated based on the listed measurable parameters, a weighting factor was then applied as a multiplier. The weighting factors chosen are listed below:

|                       | <u>Wt. Factor</u> |
|-----------------------|-------------------|
| distance to generator | 6                 |
| distance to stream    | 5                 |
| area                  | 4                 |
| surface slope         | 2                 |
| distance to public    | 6                 |

This method of rating sites is considered to be under development and other factors will almost certainly be introduced (e.g., depth to water table and geologic factors) as additional data become available. It is our intention to narrow the sites down to 2-3 and then do a detailed hydrogeologic investigation on these. Ultimately we may select one site for GCD and another for SLB, or perhaps a single site may be found suitable for both modes of disposal.

Table 5. Site Selection Parameter Ratings

| <u>Rating</u> | <u>Dist.<br/>to<br/>Gen.<br/>(mi)</u> | <u>Dist.<br/>to<br/>Streams<br/>(ft)</u> | <u>Area<br/>(Ac)</u> | <u>Slope<br/>(%)</u> | <u>Dist.<br/>to<br/>Public<br/>(mi)</u> |
|---------------|---------------------------------------|------------------------------------------|----------------------|----------------------|-----------------------------------------|
| 4             | 0-2                                   | >2000                                    | >200                 | 0                    | >4                                      |
| 3             | 2-4                                   | 1500-2000                                | 100-200              | 1.2-2.5              | 3-4                                     |
| 2             | 4-6                                   | 1000-1500                                | 50-100               | 2.5-3.7              | 2-3                                     |
| 1             | 6-8                                   | 500-1000                                 | 25-50                | 3.7-5.0              | 1-2                                     |
| 0             | >8                                    | <500                                     | <25                  | >5                   | 0-1                                     |

SUMMARY REPORTS, CLOSING REMARKS, AND LIST OF ATTENDEES

SUMMARY REPORT  
SESSION A-1: WASTE TREATMENT

R.R. Jaeger, Chairman  
Monsanto Research Corporation, Mound Facility

Has DOE Adequately Defined the Technical Problems?

For the area of Waste Treatment covered in this session, there was general agreement that the problem was fairly well defined. There were a few specific areas, such as uranium chips from machining operations and volume reduction of tritiated water streams, that were identified as not being addressed.

There was a general consensus that the program was heavily geared toward disposal. It was suggested that other areas such as using low-level waste (metals) as containers for high-level waste be given more consideration.

Considerable discussion was centered on volume reduction. This subject was addressed from both the technical and economic issues. The commercial application of volume reduction technology is solely driven by the economic issue. As long as paybacks based on one or at most three years are used by industry, not many of the current technologies will be implemented. However, the economics are subject to rapid change as the availability, or lack thereof, of burial ground space changes. It is also environmentally prudent to maximize the utilization of existing and planned disposal sites.

Is the Program Designed to Answer the Questions?

There was a general concern that defense waste problems are not being addressed unless there is a direct commercial spill over. Since commercial application and decisions are almost solely economically motivated, more cost information should be included in reports on respective technologies.

This request for cost information led to a broad discussion on how to provide this data and on how to define a common cost basis. The only conclusion reached was that a common basis was hard to define. The contractor doing the work should provide as much information as possible with clear definition of the basis used. Thus, anyone considering implementing a given technology should conduct their own engineering estimate working whenever possible in direct communication with the individuals that generated the data.

Is the Program Being Conducted Properly?

There was general agreement that the program was being conducted properly. During discussion of this question a few administrative concerns surfaced. As might be expected, several contractors and field

offices discussed problems they have had with the dual lead lab system. These problems, primarily of the communication nature, seemed more acute at contractors further removed from the field offices and area offices. Also, the federal budget process seemed to be at the heart of several of the problems experienced.

As a result of the general discussions, it was felt that the communication process could be improved if copies of the program monthly reports normally sent to headquarters with copies to the field offices also be sent directly to the contractors.

#### Who Needs the Technology? How to Transfer Technology

For waste treatment projects there was agreement that these projects are very well scrutinized for their potential application. The potential users of the technology are addressed in the Field Task Proposal/Agreements and reviewed at least annually.

Discussions on how to transfer technology led to the conclusions that transfer of technology, solely through the open literature, is relatively poor and essentially leads to duplication of effort. Transfer of technology through one-to-one interaction is far superior.

Based on the above, two recommendations were presented to the Program. 1) Projects should not just end with final reports. If the technology is not implemented immediately at the developing contractor's site, the program must allow for at least minimum support during the technology transfer period. 2) The Program should publish and widely distribute a listing and brief description of the Program's projects to potential users of the technologies.

SUMMARY REPORT  
SESSION A-2 PACKAGING AND WASTE FORM

J.H. Kittel, Chairman  
Argonne National Laboratory

Summary of Session A-2: Packaging and Waste Form

Session A-2 considered Waste Acceptance Criteria, Waste Form Criteria, Waste Form Development, Radionuclide Assays, and Technology Transfer.

A. Waste Acceptance Criteria

1. Status

The Department of Energy's proposed Low-Level Waste Acceptance Criteria are presently under review by the Field Offices. The criteria are ultimately to be implemented in a flexible manner by the Field Office as part of Department of Energy Order 5820.

2. Issues and Problems

(a) The Department of Energy indicated that it does not necessarily intend to follow the Department of Transportation and other requirements that non-Department of Energy groups are obliged to observe.

(b) Multiple safety factors may have to be introduced into the criteria that would actually reduce safety.

(c) The bases for the 500-year site performance criterion are not stated.

3. Recommendations

(a) To avoid loss of credibility, the Department of Energy should follow the same regulations as those applied to non-Department of Energy groups, unless this is clearly impracticable because of the nature of the waste.

(b) The waste acceptance criteria should be examined for the bases of the original source terms, to determine if undue layering of source terms has occurred.

(c) The bases for the 500-year site performance limit should be defined.



**B. Waste Form Criteria****1. Status**

This Nuclear Regulatory Commission program is intended to support 20 CFR 61.

**2. Issues and Problems**

(a) High-integrity containers are an expensive means of minimizing trench subsidence.

(b) There are no clearly-stated waste form performance objectives.

**3. Recommendations**

(a) Bulk waste disposal and compaction in the trench is much less expensive than high-integrity containers as a method of minimizing trench subsidence.

(b) Waste form performance objectives should be defined.

**C. Waste Form Development****1. Status**

Emphasis is on developing acceptable waste forms for new and problem waste streams, such as oils, solvents, ion exchange resins, concentrated evaporator bottoms, etc.

**2. Issues and Problems**

(a) There is a lack of standardized testing of waste form and container materials.

(b) It is difficult to correlate short-term immersion tests with long-term lysimeter tests.

**3. Recommendations**

(a) A standardized testing protocol should be established, possibly analogous to the series of tests developed at the Materials Characterization Center for high-level waste forms.

(b) Tests in a lysimeter with inert materials, or long-term laboratory tests in representative soils, should be made.

#### **D. Radionuclide Assays**

##### **1. Status**

This Nuclear Regulatory Commission program is intended to determine typical concentrations of radionuclides in nuclear power plant wastes.

##### **2. Issues and Problems**

(a) Sampling uncertainties cause large errors in reported analyses.

(b) Some radionuclides are difficult to measure because of lack of energetic gamma emission.

(c) Modelers using curie contents in waste may be ignoring large uncertainties in source terms.

##### **3. Recommendations**

(a) Reactor operators should be responsible for improving sampling accuracy.

(b) Improved analytical techniques should be developed for weak gamma emitters, possibly using correlations with strong gamma emitters.

(c) Modelers should factor uncertainties in source terms into their projections.

#### **E. Technology Transfer Recommendations**

(a) The Department of Energy should consider designating one of its disposal sites as a model facility. Visits by state and local authorities, commercial operators, etc., should be encouraged.

(b) The Department of Energy should sponsor joint activities such as site selection, characterization, and monitoring.

(c) The Department of Energy should sponsor training courses in such activities as waste treatment, packaging, on-site storage, transport, disposal, and site closure.

SUMMARY REPORT  
SESSION B-1 ENVIRONMENTAL ASPECTS

J.R. Wiley, Chairman  
Savannah River Laboratory

The Session Chairmen were charged with guiding the group discussions along six objectives. This was essentially done in Session B-1 although we did not address each of the questions individually.

The first three objectives were:

- 1) To determine if the program has comprehensively identified problems, gaps, or issues associated with management of low-level waste;
- 2) To determine if the program as structured addresses these issues;
- 3) To determine if the tasks being performed to address the issues are technically sound.

In our discussions these objectives became lumped together in the sense that we identified a set of research or technology areas that, while they may be presently addressed to some extent within the low-level waste program, deserve greater emphasis.

o Use of vegetation to enhance evapotranspiration

This would be a possible water control measure since even in a humid site a large fraction of the rainfall is lost by this process.

o Identities and properties of chemical species that could enhance migration

This is a fundamental of organic complexes or inorganic ions that could mobilize radionuclides. There is some preliminary study in progress, but this work needs more emphasis.

o Disposal of decontamination waste generated in clean-up or volume reduction activities

Decontamination materials are designed to mobilize and collect radionuclides; radionuclide mobility is exactly what you don't want in a low-level waste disposal site. The question of where or how one disposes of the decontamination waste is largely unresolved.

- o Modeling of heterogeneity

There is general agreement that models are pretty good for homogeneous media, but in an actual site there are heterogeneities like sand lenses in soil or fractures that increase water flow. This is a specific area of modeling that needs emphasis.

- o Integration of lab data, field data, and theory in model development

There was general concern that these items were not being adequately brought together in model development.

- o Variance analysis

This item arose because it should be made clear that a model is a tool. It can be used properly, or misused. Limits of a model must be well understood in order to use it properly.

The fourth meeting objective was to identify information on technology that is appropriate to transfer to potential users.

- o Monitoring techniques and experience

The maturity of this technology is recognized by the program as evidenced by the Argonne Monitoring Handbook.

- o Models are generally good enough to be transferred to potential users as long as the limitations and appropriate use of the model as a guide is understood by the user.

- o Professional expertise and judgement

Low-Level Waste Management Program participants have accumulated a wealth of experience in shallow land disposal. This experience is ripe for transfer.

The fifth objective, to identify potential users, was generally addressed on Tuesday and was not pursued in Session B-1. However, the sixth objective, to identify appropriate mechanisms to transfer the technology, was addressed.

- o Handbooks such as the one on site monitoring, volume reduction, and other programmatic products were considered necessary but not sufficient to transfer technology.

- o The experience and judgement I mentioned as ripe for transfer could be made available by getting the experts in a specific area together to develop consensus positions. This is essentially what is done in developing standards by the American Society of Testing Materials, and I believe is essentially what happened in the recent meeting to raise the transuranic 10 nCi/g limit to 100 nCi/g. Consensus positions among the low-level waste community would instill public confidence.
- o We need to work harder to publish and disseminate our results, even intermediate results, promptly.
- o A central clearing house, or information resource center, could be designated as custodian of all low-level waste information. Factual information, reports, technical directories and other information could be made directly available to interested persons. Such an approach has been used in other programs.

SUMMARY REPORT  
SESSION B-2 IMPROVED SHALLOW LAND BURIAL

J.G. Steger, Chairman  
Los Alamos National Laboratory

This session consisted of from 30 to 40 participants. Three Department of Energy and five Nuclear Regulatory Commission papers were presented. After the papers were presented, the group was asked to provide a consensus on four questions.

I. Has the program comprehensively identified the problems, gaps and issues?

This question is a bit tricky to answer due to the interactive nature of waste, containers, and environment. With this limitation in mind, the group answered a qualified "yes" to this question.

For the purpose of evaluating the program scope, it was assumed that the Department of Energy will adopt some sort of a waste classification system similar to the Nuclear Regulatory Commission's 10 CFR 61. If this is so, then different levels of technology will be required for different classes of waste.

Class A Waste. Present technology is adequate. Since the waste is essentially non-hazardous to start with, the most important consideration is to prevent increasing the hazard (bioaccumulation). Existing engineering technology is adequate.

Class B and C Waste. A firm conclusion must be reserved pending an overall evaluation of the relative contribution that the different points of the improved technology make in meeting performance objectives under different environmental conditions. At this point in time, as one would expect in a mature program, the key aspects of the problem seem to be addressed. A qualified "yes" was arrived at for these waste types.

There were several areas brought up and discussed which the group had some concern about. These were:

Monitoring. Technology adequacy will be proven in application through the data acquired by the monitoring systems installed at operating sites. This is an essential feedback loop to the developers for them to determine if the technology is performing as planned or if some modifications are necessary. This is a critical information link for "mid-course corrections."

Subsidence. One open issue appears to be subsidence. Should it be accelerated to get it over as quickly as possible, or should it be resisted as much as possible with the understanding that problems in the future are possible. Subsidence is an important consideration in

performance prediction and perhaps a position should be developed as to which is the preferred option.

$K_d$ . Consistent with experiences and expectations, the role of  $K_d$  and our ability to measure it was discussed. The variability of  $K_d$ 's is such that a site which heavily depends upon  $K_d$  to meet performance requirements can expect to have a troubled future.

## II. Does the program as structured address the issues?

Without a systems analysis (optimization, engineering evaluation, or whatever term you prefer) a clear answer is not possible. Group members who have considerable experience in low-level waste matters expressed concerns about the status of four different topics.

Humid Site Water Management. Because several humid area sites have experienced water problems in the past, the ability to avoid or to control water problems will properly be carefully reviewed in any new site proposal. Not very much work seems to be in place in this area. This leads to the conclusion that perhaps this topic deserves more emphasis.

Biotic Transport. One pathway which may not have been adequately investigated and modeled is the biotic pathway. Most, if not all, sites have encountered some surface contamination incidents caused by plants or animals. Vegetation litter containing radionuclides, contaminated materials and soils brought to the surface and/or dispersal by animals (rabbits, gophers, birds) cannot be assumed to be an insignificant pathway because of insufficient data and analysis. Some evidence seems to indicate that this pathway is as significant as the intruder pathway after 100 years.

Analysis. Some analytical effort needs to be directed to determining what can be controlled and what cannot, what can be done to circumvent the uncontrollable, and does the technology exist to effectively do this. Different waste types and different environments present different sets of controllables and uncontrollables. Some method needs to be developed to determine weaknesses as well as when, "enough is enough." The presence of political, emotional and financial factors in such considerations does not make this problem a simple one.

Actual Test Data/Model Verification. If one asks the question as to whether we have enough data to apply to the Nuclear Regulatory Commission for a shallow land burial site license, one quickly realizes that real data is a bit limited to handle all of the possible "what ifs." In some cases where site conditions indicate no reasonable transport pathways, perhaps present data is adequate. However, in complex situations, the data base is insufficient for model verification.

### III. Are the tasks being performed technically sound?

This question cannot be answered. The limited presentations and discussion times are not adequate to fairly make a judgement. Nothing bad was apparent. However, from a program standpoint, the group had the following comments.

Development Completed by 1984. The group had grave reservations that all technology for all possible site conditions will be available. For some sites the technology will be adequate, but for others it will not. In cases where water control at humid sites or biotic pathways are important, the technology will be inadequate.

Development Completed by 1986. It is possible that necessary technology will be available. However, Environmental Protection Agency criteria or more restrictive regulations (Resource Conservation and Recovery Act) could drastically change the situation.

### IV. Can the technology which should be transferred, potential users, and appropriate mechanisms for transferring technology be identified?

There are so many people and technical issues potentially involved that there is no easy answer. Normal formal information dissemination methods (reports to libraries, journal articles) take too long and represent too many data sources to monitor. Nobody can keep up with all of the literature today. The group submits the following for consideration.

Quarterly Newsletter. The semi-informal format and wide distribution of this vehicle lends itself to be a general clearing house for what is going on and who is doing it. Interested people can easily identify and contact the appropriate people. No other suitable vehicle was identified.

Transfer Mechanisms. Because it is inappropriate for every researcher to disseminate "Department of Energy technology," the mechanisms of Handbooks and Topical Reports from DOE seem to be the best mechanism. Informal mechanisms in existence seem to be working OK.

Peer Review. Ultimately, the technology developed will be subject to review by the public. Past failures to have peer reviews have resulted in difficulty in obtaining acceptance. The sooner the technology is published in journals (or equivalent) the easier it will be to obtain acceptability.



SUMMARY REPORT  
SESSION B-3: CORRECTIVE MEASURES

J.D. Sease, Chairman  
Oak Ridge National Laboratory

Discussion in Session B-3 began with an assessment of what constituted corrective measures. It was determined that they included activities designed to 1) obtain short-term improvements in order to buy time to study problems; 2) affect long-term improvements, ideally passive, minimum maintenance measures; and 3) maintain site integrity, e.g., filling holes, fixing erosion, and planting vegetation. In general, corrective measures applied at disposal sites have been in response to

- o good engineering practices (to prevent the development of worse problems)
- o licensing requirements of the Nuclear Regulatory Commission and states

For sites operated by the Department of Energy, future corrective actions will include responses to "trigger levels" observed via the monitoring program and specified in Department of Energy criteria currently under review for publication.

In addressing the primary questions itemized by the Low-Level Waste Management Program, the order of presentation was reversed and attention was first focused on what the technology needs were believed to be. It was agreed that the following list of causative conditions adequately represented the source of problems at current sites:

- o Unstable Trench Cover
- o Permeable Trench Cover
- o Subsidence
- o Intersection with Ground Water System
- o Intrusion by Deep-Rooted Plants
- o Intrusion by Burrowing Animals
- o Chemical and Physical Conditions in Trench

Concerning technology transfer, it was determined that engineering-scale demonstrations of effective correction of these conditions would be very instructive. The demonstrations would need to be of integrated systems which treated these conditions together as they occurred at disposal sites. The system would need to enable monitoring

and evaluation of the results of changes on interrelated conditions. For example, at humid sites it is particularly important to know the effects of changes in permeability of the trench cover on surface runoff and resulting problems with stability of adjacent surfaces.

Based on review of work on corrective measures known to be underway at Department of Energy facilities and on perceived urgency of current problems, the following prioritization was constructed for components of demonstration systems:

- o Highest Priority
  - Subsidence
  - Permeable Trench Cover
  - Unstable Trench Cover
- o Intermediate Priority
  - Chemical and Physical Conditions in Trench (causing radionuclide migration)
  - Intrusion by Deep-Rooted Plants
- o Lower Priority
  - Intersection with Ground Water System
  - Intrusion by Burrowing Animals

Additional important considerations for designing demonstration systems included long-term stability and subsurface fracture flow.

Endorsement was given to the design and current conduct of the Low-Level Waste Management Program with respect to corrective measures, with the proviso that the demonstrations previously discussed be incorporated.

The need was cited for cost-to-benefit analyses which included evaluation of the long-term consequences of corrective actions. It was observed that short term corrections currently being applied or under consideration may not be based on adequate attention to long-term return on investments.

In addition, it was believed that more attention needed to be paid to the form of the waste put into the trench. Important considerations included reductions in void volume and in decomposable material. This would assist the process of anticipating and responding to subsidence and, consequently, in developing long term, cost effective corrective actions.

In summary, it was concluded that:

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- o Technology transfer needed more emphasis on engineering-scale demonstrations, with consideration given to the priorities discussed.
- o Cost-to-benefit analyses for evaluating corrective measures should consider long-term as well as short-term costs and benefits.
- o More attention should be given to the form of the waste disposed for decreasing disposal site problems.

SUMMARY REPORT  
SESSION B-4: GREATER CONFINEMENT DISPOSAL

K.V. Gilbert, Chairman  
Transuranic Waste Storage Office  
Rockwell International, Rocky Flats Plant

The Greater Confinement Disposal workshop was attended by participating site contractors, Low-Level Waste Management Program staff, and Department of Energy and Environmental Protection Agency representatives. Attendees were requested at the outset to consider and respond to the following questions during the course of the workshop:

- o Has the Department of Energy adequately defined the problem?
- o Is the program designed to answer the questions?
- o Is the program being conducted properly?
- o Who needs the technology and how do we get it to them?

These problem areas are addressed in the following discussion.

Potential Benefits

The potential benefits of the greater confinement disposal concept for low-level wastes were identified and discussed. It was recognized at the beginning of the session that greater confinement disposal is a commercial as well as a defense waste item of interest.

Greater confinement disposal affords a reduced potential for intrusion compared to shallow land burial. Plant root uptake of radionuclides and inadvertent intruder scenarios were discussed at length as being extremely important factors in disposal facility design considerations. Greater confinement disposal would increase the depth of waste burial and thickness of cover, therefore providing a means to avert biological or human intrusion.

Disposal limits might be increased by making it possible to accommodate a wide range of wastes. Wastes exceeding the proposed 10 CFR 61 Class C limits would be candidate greater confinement disposal wastes, as well as some Class B and Class C wastes.

Greater confinement disposal appears to be a cost-effective alternative to shallow land burial. Greater confinement disposal may be cheaper in the long run, by making it possible to bury certain low-level waste without expensive waste processing and corrective measures that may be required at shallow land burial disposal sites. According to work conducted by REECO at the Nevada Test Site, the operational costs

for greater confinement disposal and shallow land burial are similar, depending upon the greater confinement disposal method. Development costs and closure costs, on the other hand, were shown to vary, with some being greater for greater confinement disposal and vice-versa.

Criteria for greater confinement disposal are currently in preparation. Greater confinement disposal site selection criteria will be developed and issued prior to the end of the 1984 fiscal year.

In addition, the Low-Level Waste Management Program waste classification system addresses greater confinement disposal. Waste classification criteria will identify wastes requiring disposal at a site providing greater confinement. A preliminary draft of these criteria will be available early in the 1983 fiscal year.

### Cost Effectiveness

In the area of cost effectiveness, a number of parameters are worth noting. First of all, do low-level waste volumes support the need for greater confinement disposal application? A Low-Level Waste Management Program survey of low-level waste exceeding the proposed 10 CFR 61 Class C regulations was recently conducted. Results indicated that the total volume of defense and commercial wastes requiring non-shallow land burial disposal was on the order of 1300-1500 cubic meters per year. This does not include alpha-contaminated (transuranic) waste, of which perhaps 35 percent could be disposed of as low-level waste, using greater confinement disposal.

Are shallow land burial and greater confinement disposal costs similar enough that greater confinement disposal should be considered for Class B and C wastes? It may be more cost-effective to provide greater confinement for these wastes rather than conducting expensive waste stabilization or waste form development and processing for disposal at shallow land burial sites. Intensive discussion was generated relative to the co-location of greater confinement disposal facilities with shallow land burial facilities, or the development of greater confinement disposal facilities in lieu of shallow land burial facilities. Several greater confinement disposal options examined were apparently associated with total costs similar to those of shallow land burial facilities.

Does the cost of assay and sorting affect greater confinement disposal economic attractiveness? Instrumentation for waste assay and segregation to make greater confinement disposal a cost-effective alternative to shallow land burial is costly or not readily available otherwise. An interesting idea was posed as to whether the activity level and radionuclide content of low-level waste packages received at the disposal site can really be accurately determined and provided by the generator. The Savannah River Laboratory has conducted a valuable waste segregation study related to this issue.

### New or Modified Efforts

Some areas of the greater confinement disposal program have been identified that need new, or at least modified, effort. Better inventory data on candidate wastes is required; this was one of the main problem areas identified in the workshop. For instance, do we really have a sufficient grasp of the volume(s) and type(s) of commercial and defense waste requiring greater confinement disposal? Also, where (regionally) are these wastes located? Work is indicated to resolve these gaps in the program.

More quantitative cost analyses are needed. This was another main area of need identified. Is greater confinement disposal really worth the costs incurred? What are the costs? Cost-benefit analyses should include such factors as corrective measures costs or conversion of shallow land burial to greater confinement disposal costs. How much of the Low-Level Waste Management Program budget is directed towards corrective actions at shallow land burial sites? It might prove valuable to compare greater confinement disposal facility closeout costs with shallow land burial facility closeout costs. Does greater confinement disposal then become less expensive in the long term?

A conscious effort to address waste acceptance criteria is needed. The program needs to consider waste acceptance criteria with respect to greater confinement disposal needs and applications.

More directed effort toward site selection criteria is indicated. Site selection methodology and criteria must be developed for matching low-level waste needs with potential greater confinement disposal facility sites.

Quantitative assessment of greater confinement alternatives is required. Alternative greater confinement disposal methodologies have been examined on a qualitative basis in the past. Quantitative studies are needed now. A new initiative addressing this concern will be conducted at Argonne National Laboratory in the 1983 fiscal year.

The Low-Level Waste Management Program needs to assess the schedule modifications to influence the regulatory framework. Will the schedule for greater confinement disposal-related deliverables be timely? The final Greater Confinement Disposal Handbook is due to be issued in March, 1986. Will this be soon enough to influence actions undertaken by the Nuclear Regulatory Commission and the Environmental Protection Agency? The general consensus was that critical portions of the Handbook will be available in a timely fashion.

### Users of Greater Confinement Disposal Technology

Potential users of greater confinement disposal technology have been identified as: state compacts, Department of Energy sites, disposal site operators, Environmental Protection Agency/Nuclear Regulatory Commission, and foreign governments. Greater confinement

disposal waste classification and greater confinement disposal facility site selection were identified as a real need for the state compacts user category. States may be restricted users, however, because they will be closely tied to 10 CFR 61.

The Department of Energy low-level waste sites are the obvious initial users of greater confinement disposal technology developed by the Low-Level Waste Management Program. Department of Energy sites require near-term attention and can more quickly apply the developed technologies.

Potential commercial disposal site developers and operators comprise another user category. It is expected that they will be major users of the information resulting from Low-Level Waste Management Program greater confinement disposal tasks.

An important area of interchange and technology transfer exists with the Environmental Protection Agency, Nuclear Regulatory Commission and other federal agencies. The Low-Level Waste Management Program contractors are also potential users of other federal agencies' work related to greater confinement disposal.

Foreign exchange applications might be expected with the Federal Republic of Germany, Japan, Great Britain, or other countries. The Low-Level Waste Management Program would also benefit by their experiences.

#### Workshop Suggestions

The Low-Level Waste Management Program seems to be generating meaningful technology transfer documentation, but a couple of suggestions were presented. First of all, more aggressive communication is required. There are still communication gaps, even between Low-Level Waste Management Program participating contractor sites. No definitive conclusion arose as to how to solve this problem, but more frequent meetings of various site personnel might be indicated. A suggestion was made to promote site visits. A periodic revision of the UC-70 distribution list may be necessary for document issue to enhance better communication of technology development.

The program was encouraged to explore timely review and publication of greater confinement disposal technology reports and documents. For the technology to be optimally useful, it must be made available in time for application. In addition, some program tasks might be escalated so as to make the results available sooner and promote technology transfer.

#### Summary

Workshop attendees seemed to be in general agreement that the Low-Level Waste Management Program greater confinement disposal program is providing a useful technical base for technology development and application. Apparently, it was felt that the areas of technical need

previously identified (water management, radionuclide migration, prediction capabilities, monitoring performance and evaluation, and management practices) were being addressed by the Low-Level Waste Management Program; no new areas of need were identified.

The experimental program utilizing the greater confinement disposal facility at the Nevada Test Site was cited as an example of progress. Results of the field studies undertaken as a cooperative effort between Los Alamos National Laboratory and REECO will provide useful data in the 1983 fiscal year.

Humid site field data are also being generated and evaluated. Work being conducted towards a greater confinement disposal facility at the Savannah River Plant will be forthcoming.

The Environmental Protection Agency is developing a generic assessment. The agency's PRESTO model will be useful for site performance prediction.

As was previously discussed, some modification of the Low-Level Waste Management Program tasks and contractor activities is suggested as a means to facilitate technology transfer in the area of greater confinement disposal. Although definite courses of action to achieve this goal may not have been identified in this workshop, the general status and direction of program efforts was established.



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SUMMARY REPORT  
TECHNICAL GROUP A: WASTE TREATMENT, PACKAGING, AND HANDLING

D.P. Ferrigno, Chairman  
Gilbert Associates

Overview of Technical Papers

The technical papers are referenced relative to subject material. The following are commentaries of formal presentation material.

- o It appears that the Low-Level Waste Management Program is focusing on the mission of technology of research and potential information transfer. It further appears that the program could try to establish formal information transfer mechanisms with potential industry users (Edison Electric Institute, Electric Power Research Institute, American Nuclear Society, American Society for Mechanical Engineers, utilities) when data is in the preparation stage as opposed to information dissemination when work is complete.
- o Waste classification indexing seems to have a useful purpose; but demonstration of how the user will apply the index in a standard way needs to be addressed. The hazard index, due to potential subjective interpretation, may require concurrent Department of Energy, Nuclear Regulatory Commission, Environmental Protection Agency and industry acceptance and rulings in order that implementation is applied universally in the same way.
- o Generation of a waste treatment handbook is a formidable task. Some thought should be discussed concerning the methodology of "peer review" prior to formal circulation. Groups such as Edison Electric Institute, Electric Power Research Institute, American Nuclear Society, American Society for Mechanical Engineers, and private industry representatives may provide the added "user" experience necessary to make the handbook more useful.
- o Fuel cycle waste characteristics should reflect actual experience, not just reports from Nuclear Regulatory Commission documentation. Possible information exchange with the Electric Power Research Institute and Edison Electric Institute could provide more specific information.

- o Nonfuel cycle waste problems seem to be limited in scope and are a new area of study.
- o Properties of radioactive waste and containers is an area that needs further development. Although this further development is required, the program efforts should be cognizant of previous data developed under Nuclear Regulatory Commission, Electric Power Research Institute, and Edison Electric Institute funding.

#### Summary of Discussion Topics from the Panel and Working Group

The following notes are not intended to represent an agreement or majority consensus on issues or resolution of issues. These notes are intended to highlight those topics that were discussed at the meeting. Neither the panelists nor the Session Chairpersons endorse any or part of the following, but rather present the material for information documentation.

- o There was general consensus that today, given the basis on which utilities are making decisions, volume reduction in the nuclear plants may not be cost effective due to payback requirements of 1 to 3 years. If payback is extended, then application of volume reduction would in most cases be cost-justified. The Department of Energy is continuing to develop volume reduction technology for two reasons: (1) if interim (or temporary) onsite storage is necessary, the economics will shift; (2) volume reduction may become necessary for continued operation based on governmental or regulatory mandate.
- o Cost information is needed for the various competing technologies. To be useful, cost data must be developed using common or identifiable bases.
- o Commercial industry considers short payback periods and costs, which include (for plant backfitting):
  - 20-30% (1) Capital Equipment (the hardware)
  - 15-25% (2) Engineering
  - 45-65% (3) Building and Construction

The reason for the range is due to site specific variances.

- o Costs of volume reduction are shifting; examples are \$2.4 M in 1974 and \$24 M in 1982; in 1977, volume reduction equipment costs ran \$1 M and \$1 M for

building, but today volume reduction is \$5 M and the building and accessories runs \$20 M.

- o Disposal site operators noted that disposal is more difficult and costly for volume reduced waste: more overburden, less trench efficiency, more exposure.
- o With only two or three disposal sites and differing disposal requirements, if one site closes or a generator has to switch sites, the generator may not have the capability to treat waste commensurate with requirements.
- o 10 CFR 61: A, B, and C present problems; the Nuclear Regulatory Commission expects to take about one year to implement. Costs in plants may be \$20 M to \$40 M per plant (over 20 years) for achieving segregation capabilities. There needs to be a coordination of effort (Department of Transportation, Nuclear Regulatory Commission, Environmental Protection Agency) for waste and container requirements.
- o High integrity containers and licensing were discussed; qualification testing and procedures (standardized tests?) were explored. The Nuclear Regulatory Commission sees that standardized testing should be the subject for development programs by the Department of Energy.
- o 10 CFR 61 versus Department of Energy Order 5820: Are they compatible? Does technology developed for the Department of Energy fit into 10 CFR 61?
- o Quality assurance for burial sites: "What is the benefit?"
  - (1) U.S. Ecology indicates that the reduction of incidents after institution of a quality assurance program was in direct proportion to the reduction in business.
  - (2) Chem-Nuclear experienced reduced discrepancies when at-the-gate-checking began.
- o Question: Is current practice of solidification and packaging a good level of safety?

Answer: Practice in the United States is commensurate with world practice where shallow land burial is used.

- o Data in EG&G data base may be limited because the utilities do not have an accurate way to get the information contained in the semi-annual release reports that are sent to the Nuclear Regulatory Commission. This information varies in quality. Since EG&G is using the Nuclear Regulatory Commission reports as a mechanism to log the data base, the EG&G report may reflect speculative data.

SUMMARY REPORT  
TECHNICAL GROUP B: WASTE DISPOSAL

G.B. Levin, Chairman  
EG&G Idaho

First I would like to go on record as thanking Don Oakley for not showing up so that I could have the honor of doing this. I hope you guys from Los Alamos will tell him that.

What I would like to do is a little different. You can read the proceedings and find out what happened. I think I would like to share with you what I gleaned out of it -- as lessons learned.

In the area of technology transfer, in listening to what Dennis, Howard, Ralph and the other speakers have said, I think there is one thing that comes through: technology transfer is from people to people, and paper doesn't always work as a very effective way to communicate. You can say it like Ralph did, that, after the final report, you have to maintain some funding so that the people who know what's going on can share it with the people who need it. However, when you get down to it, it is people to people.

We have to build some things into the system to allow people to talk to people and share technology. Examples of how technology transfer is not working were given throughout the meeting and I'm not sure everyone was aware of these.

One of the things we had a long discussion about in the Tuesday afternoon session, was modeling and it came up again today. There were some real gems that came up that I'll share with you. One thing that was said two or three times was that we don't know how to model fracture flow systems. In fact, the Oak Ridge people have put together a model for doing fracture flow systems and we in the Department of Energy system don't know about that. Our technology transfer is not working. We have to get people talking to people so that they know what's going on. That's what I learned from that discussion.

In terms of the technology aspects, a couple of things came out. One of the important things that showed up dealt with water management at humid sites. Two sessions picked it up independently. It appeared to them that this is one of the problems at a humid site that the program was not devoting enough attention to. The other thing that the two shallow land burial sessions picked up was identifying properties that enhance/inhibit migration. One place they called it  $K_d$  and another place they called it something else, but it's the same thing. Those properties of the radioisotopes that affect migration have to be understood with some degree of precision.

The big area of discussion on Tuesday afternoon and it came through again today in the summary sessions was modeling. A lot of good things

came out of it and I can't repeat them all. I encourage you to read it. One of the things that came out is that a model is not an answer unto itself. It is not a panacea. A model is a tool for decision makers and others to use and the results from the model have to be evaluated along with the other information from the sight using sound, good judgement. A question was posed to both the Nuclear Regulatory Commission and the Department of Energy while discussing site selection, and was not really answered. Both groups say that a site has to be describable by some kind of model. What does that really mean? What kind of levels of precision are we talking about? When one says you have to show by a model that your exposure is less than 25 mrem per year to an individual, what does that really mean? Is it exactly 25 carried out to the fifth decimal place or is it 25 plus or minus 5 or 10. A model provides an estimate and it has to be evaluated in that way. The other thing I think the modelers and some of the listeners who are non-modelers agreed with is that models have been oversold as providing an answer. A more realistic perspective should be communicated by the modeler to the non-modeler like myself on how much and what can a model really tell you. I think that's important. It is also important, I think, in maintaining credibility for the system with "the real world," the outside world. One of the more technical things that came out on modeling was a lot of discussion on validation/verification being the key words. And as one of the modelers pointed out those are kind of like consultation and concurrence, you don't know what they really mean. They mean different things to different people. What it came down to is, we should have a handle on what models can do and what they can't do, where you can apply them, what systems they work with, and some kind of sensitivity studies that give you a handle on what they will do under abnormal situations. There seemed to be agreement that current models may in fact be outperforming, in terms of significant figures they produce, the confidence of the parameters that go into them. The input parameters are probably less accurate than the answers that are coming out, and we all need to be aware of that.

The other two things that had to do with modeling were heterogeneity and fracture flows that I mentioned earlier. We need to develop the capability to do it. I'm not sure from my personal knowledge whether we need to develop that capability or just need to tell people more about it. I'm not a modeler so I can't judge.

The other point that came up this morning, which I thought was extremely nice, was the integration of the theory, laboratory results, and field testing and putting it all together into a model. Possibly some more analysis of that would be useful to increasing our understanding.

One of the other things that merited attention and stuck out as having some validity was the disposal of decontamination waste. Some day they may decontaminate the resin. They certainly use it day to day. We don't really have good information on that, and then of course cost benefit analysis came up, and the need for that. We ought to do it better.

I think a good point was made today, that one has to look at the long term benefits, not the short term benefits. It may be cost effective, as the FRAM man says, to pay the seven dollars today rather than the \$7000 for a new engine. Build a good site up front and decrease long term care.

In terms of corrective measures, the ideas about trench caps being critical for subsidence and permeability, we need to work on that. Some of that work, in fact, is being done. The concerns had to do with long term stability of the trench cap and some subsurface fracture flow. They seemed to advocate Dr. DeLong's encouragement from the first day about engineering scale demonstrations -- again focused on cost benefit analysis -- and gave a strict endorsement for waste form work. And we ought to focus on waste form work. In terms of technology transfer, there was more of an endorsement for what's been happening than any new kind of breakthrough ideas.

The last area that we dealt with was separate from shallow land burial. I think I've covered most of the near surface items. There are some other things that you will see in the report -- some other suggestions -- which I'm not talking about. It's not that they are less important, it's just a personal sorting to save time.

Greater confinement was discussed in a separate section and is kind of a separate thing. There was a lot of information. There was some endorsement that the program is producing useful information, and it suggested accelerated publication and a more aggressive communication posture. I detected some confusion between the concept of providing greater confinement possibly by a better waste form and packaging, which didn't seem to be discussed in the greater confinement disposal facility testing which is being done at Nevada and Savannah River. The discussions seemed to focus in on the facilities as opposed to some other options. That needs to be documented better.

In listening to the summary this morning I wasn't sure they reached any real conclusion. There were some suggestions that greater confinement might be a cost effective alternative to shallow land burial in some parts of the country by saving money on waste forms and some things like that. I didn't really see anything that stuck out as a new concept. They did indicate that a comparison of the costs and benefits of alternative greater confinement concepts is needed in the short-term to assist the Environmental Protection Agency and the states in standard setting and disposal site planning.

I think that touches most of the things that stood out in my mind.

I'd like to do one other thing before I give up the microphone. Some of you have come up to me and shared your nice feelings about the conference. I thank you for that. I also think you ought to thank Tom Row, whose idea this was a year and a half ago, and Lance Mezga and Leroy Stratton, who really did the dog work and carried it out. I think



those good feelings you shared with me should also be shared with them.  
I would like to thank you all for coming.

## CLOSING REMARKS

L. J. Mezga  
Oak Ridge National Laboratory

The question is: Where do we go from here? We had some great summaries. George Levin and I indicated when the meeting began that we were not going to work at this meeting. This was your job -- you were going to work hard. After the meeting is when our work begins. We will take the results, distill them somewhat, maybe expand on some of them, sit back, analyze and evaluate them, and see how they should be factored into the program. That is what we are going to be doing within the next few months.

I would like to say that my observations, and those of the rest of the people whom I have talked with, are that everyone worked hard. We said that this was your meeting and we are pleased that everyone did a good job.

I have reviewed the critique sheets, and one overwhelming theme throughout was the desire for the continuation of involvement and expansion of information exchange by all of the agencies represented here. Representatives of the Environmental Protection Agency, the U.S. Department of Energy, the Nuclear Regulatory Commission and the United States Geological Survey have indicated that they would like to encourage their contractors to continue the communication that took place here.

George identified some people to be thanked and there are others whom I need to acknowledge who did an outstanding job in putting this session together. I would like to thank John Starmer, Nuclear Regulatory Commission, who worked very hard. I would also like to thank Lou Meyer, U.S. Environmental Protection Agency, for his time and hard work. The technical session chairmen performed beautifully. I was accused of getting blood from a turnip in one of the sessions. What we really did was to identify the people who had the knowledge, experience, savvy, and the drive to get the job done, and gave them an opportunity to carry the ball for us -- and that is what they did.

George and I really did not do much. You did it all, and I would like to again express our appreciation to the technical session chairmen for working so hard to see that the meeting was successful.

Others whom I wish to thank are Norma Callahan, Thelma Patton, Brenda Baber, Pat Viles, Don Wilkes, and Leroy Stratton, all from Oak Ridge National Laboratory. Special thanks to Leroy for his efforts in getting the arrangements and program finalized. As I indicated previously, I did not have much to do with the organization of the meeting. I just provided the introductory comments. The people designated did the work, and I would like to acknowledge their successful efforts.

Thank you.

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FOURTH ANNUAL DOE LLWP PARTICIPANTS' INFORMATION MEETING  
August 31 - September 2, 1982  
Denver, Colorado

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