109.3/RBU/82/07/02/0

JUL 16 1982

WMHT: 3109.3

DISTRIBUTION
WMHT r/f
NMSS r/f
WM r/f
CF
JB Martin
RE Browning
MJ Bell
P Altomare
J Surmeier
R Wright
S Coplan
J Greeves
HJ Miller
R Uleck & r/f
PDR (2)

Dr. Franklin E. Coffman
Deputy Assistant Secretary
for Nuclear Waste Management
and Fuel Cycle Programs
Office of Nuclear Energy
U.S. Department of Energy
Washington, DC 20545

Dear Dr. Coffman:

Thank you for providing a copy of the DOE "Facilities for National Waste Terminal Storage Project Plan" of April 15, 1982 (transmitted with your letter of June 11, 1982) in response to my previous request for information on your current schedules. This will be helpful to us in scheduling our site characterization review efforts over the next several years.

I would like to point out one matter regarding NRC requirements for site characterization reports (SCRs) which are not accurately reflected in the Project Plan. Statements on pages 5, 6 and 19 in the Project Plan are not clear on the preparation and review of the SCR to be submitted for each site. Our procedural rule does not require, as implied on pages 6 and 19 of the Project Plan, that DOE submit an "updated SCR" after completion of a substantial part of site characterization. As required under our procedural rule, 10 CFR 60.11, the SCR is to be submitted prior to site characterization and include DOE's plans for site characterization. The SCR will then be formally reviewed by NRC and, along with our analysis, be made available for public comment. Our only requirement for updates [under 10 CFR 60.11(g)] is that DOE inform NRC by semi-annual report of the overall progress of site characterization.

Orig. Not rtnd. fm. DCS
ACC. 8208030725

Susp. 221

OFC	WMHT	WMHT	WM			
NAME	RUleck:ls	HJ Miller	JB Martin			
DATE	82/07/02	7/ /82	7/ /82			

Thank you again for your schedules. Since the DOE is evaluating the NWTS program strategy and schedule in an effort to accelerate milestones as you mention in your letter, I appreciate your intentions to keep me informed of any changes in program schedules.

Sincerely,

Original Signed by
John B. Martin

John B. Martin, Director
Division of Waste Management

WM TICKET NO. 82-388

In put for developing this letter was obtained from WMPI and WMHL

OFC	WMHT	WMHT	WMHT			
NAME	RUleck:ls	HJ Miller	JB Martin			
DATE	82/07/02	7/15/82	7/15/82			

FROM	DATE OF DOCUMENT	DATE RECEIVED	NO.:
DOE	6/11/82	6/16/82	WM-82-388

LTR. XX w/report | MEMO: | REPORT: | OTHER:

TO: JBMartin, WM

ORIG.: XX | CC: | OTHER:

ACTION NECESSARY [X] | CONCURRENCE [ ] | DATE ANSWERED: 125

NO ACTION NECESSARY [ ] | COMMENT [ ] | BY: ~~6/30~~ 7/14

CLASSIF.: | POST OFFICE | REG. NO:

FILE CODE: 104.1

DESCRIPTION: (Must Be Unclassified)

Response to ltr of 3/17/82 re: DOE Planned Site Characterization Activities in the National Waste Serminal Storage (NWTS) Program

ENCLOSURES:

CLOSED 7/16/82
Ltr. to Coffman from Martin

REMARKS

REFERRED TO	DATE	RECEIVED BY	DATE
HJMiller	6/16		
Uleck	6/16		
① get comments from		② complete	
Sumner		draft by	
Bell		~~6/25~~	
Wright	rec'd	extended to 7/14	
Cyclan	comments		
Sutton	from		
Grace			
Miller			

U. S. NUCLEAR REGULATORY COMMISSION

MAIL CONTROL FORM

FORM NRC-326 (1-75)

Department of Energy
Washington, D.C. 20545

Tribut for Miller comments?

JUN 11 1982

Mr. John B. Martin, Director
Division of Waste Management
U.S. Nuclear Regulatory Commission
Washington, D.C. 20545

Dear Mr. Martin:

This letter is in response to your letter of March 17, 1982 regarding our planned site characterization activities in the National Waste Terminal Storage (NWTS) Program. Concerning our current reference program activities, I have enclosed a copy of a recently approved DOE project plan for the NWTS Program which summarizes this information and is not restricted for internal programmatic use. This information is consistent with our repository siting plans as outlined in the draft National Siting Plan released in March.

As you may know, the Department is evaluating this program strategy and schedule in an effort to accelerate both near-term and long-term milestones. We are considering several alternatives to achieve this acceleration, and have been interacting with your staff in investigating the feasibility of these alternatives from the standpoint of licensing. In addition, the waste management legislation currently under consideration in Congress also would affect the current DOE program schedule, and modifications to the schedule would be required. You will be advised of any changes to our reference schedule if a change is formally adopted by the Department.

Sincerely,

for Robert W. Ramsey Jr.

Franklin E. Coffman
Acting Director
Office of Terminal Waste Disposal
and Remedial Action
Office of Nuclear Energy

Enclosure

U.S. DEPARTMENT OF ENERGY

FACILITIES FOR NATIONAL WASTE TERMINAL STORAGE

PROJECT PLAN

Assistant Secretary
for
Nuclear Energy

April 15, 1982
Issue Date

TABLE OF CONTENTS

		Page
Section 1.	Mission Need and Objectives	1
Section 2.	Technical Plan	4
Section 3.	Risk Assessment	10
Section 4.	Management Approach	13
Section 5.	Acquisition Strategy	15
Section 6.	Project Schedule	19
Section 7.	Resources Plan	19
Section 8.	Scheduled Decision Points	20
Section 9.	Submissions and Approvals	21
Attachment 1.	Summary of Baseline Information	22
Attachment 2.	NWTS Work Breakdown Structure	23
Attachment 3.	NWTS Project Organization	24
Attachment 4.	NWTS Master Schedule	25
Attachment 5.	NWTS Resources Plan	26

FACILITIES FOR NATIONAL WASTE TERMINAL STORAGE PROJECT PLAN

1 Mission Need and Objectives

1.1 Applicable DOE Mission

Federal regulations require that highly radioactive wastes (including solidified high-level wastes (HLW) from fuel reprocessing and potentially unreprocessed spent fuel) from both commercial and defense sources be permanently disposed of in federally-licensed, federally-owned facilities. The U.S. Department of Energy (DOE) is assigned responsibility for developing the necessary programs for the treatment, management, storage, and disposal of nuclear wastes primarily through the Atomic Energy Act of 1954, the Energy Reorganization Act of 1974, and the Department of Energy Organization Act of 1977. Accordingly, DOE must establish temporary and permanent facilities for storage, management, and ultimate disposal of these wastes to isolate them from the biosphere for as long as they represent a significant hazard. This mission is ongoing as long as these wastes continue to be generated. The National Waste Terminal Storage (NWTS) project supports the DOE mission in compliance with the federal laws by developing the technology and facilities to permanently dispose of commercially generated high-level radioactive wastes.

DOE has prepared a Record of Decision* pursuant to the Regulations of the Council on Environmental Quality, 40 CFR 1505, on the selection of a strategy for the disposal of commercially generated radioactive wastes and the supporting program of research and development. The decision was based upon the analysis contained in the Final Environmental Impact Statement: Management of Commercially Generated Radioactive Waste, DOE/EIS-0046F, October 1980. The decision was to (1) adopt a strategy to develop mined geologic repositories for disposal of commercially generated high-level and transuranic (TRU) radioactive wastes (while continuing to examine subseabed and very-deep-hole disposal as potential backup technologies), and (2) conduct a research and development program to develop repositories and the necessary technology to ensure the safe long-term containment and isolation of these wastes.

*46 Federal Register 26677, May 21, 1981

This decision was given additional impetus through President Reagan's Nuclear Policy Statement of October 8, 1981, in which the President directed that the Secretary of Energy proceed swiftly toward deployment of a means of storing and disposing of commercial high-level radioactive waste. The President stated that it must be demonstrated to the public that problems associated with the management of nuclear waste can be resolved. The current NWTS project strategy is responsive to this direction.

1.2 Relationship to Other Programs and Projects

The NWTS project relates to DOE's overall nuclear energy goals in that it constitutes an important activity in the back end of the integrated commercial nuclear fuel cycle.* The last step in the nuclear fuel cycle is the ultimate isolation of immobilized high-level waste from the biosphere. The success of the entire commercial nuclear fuel cycle depends upon the successful performance of all of its component parts. Interfaces between the NWTS project and the other nuclear fuel cycle programs occur in contacts with spent fuel and waste treatment, and fuel cycle R&D programs. Coordination among these efforts is accomplished through the assignment of common management responsibility for commercial waste management and nuclear fuel cycles. High-level waste processing, transuranic waste processing, and transportation are not part of the NWTS project. However, there are active interfaces with these programs which are being documented through formal interface agreements, and the NWTS project does provide funding support to the transportation technology program. There is also technical information exchange with the Waste Isolation Pilot Plant (WIPP) project which is developing an R&D facility to evaluate disposal of defense wastes.

Throughout the course of the NWTS project, there is continual interaction between DOE and other federal organizations, state governments, and Indian tribes. Coordination is generally handled through memoranda of understanding and other appropriate formal arrangements.

*Statement of Kenneth Davis, Deputy Secretary of Energy before the Subcommittee on Energy and the Environment and the Committee on Interior and Insular Affairs, U.S. House of Representatives, July 9, 1981.

1.3 Project Objectives

The overall objective of the NWTS project is to plan, develop, and implement the technology and provide the facilities for the terminal isolation of commercially generated high-level and transuranic radioactive waste in a manner to demonstrate waste disposal capability, to assure the public and occupational health and safety, and to protect the environment. To meet this overall objective, the project will site, design, construct, and operate mined geologic repositories on a time frame consistent with waste disposal requirements. Approximately 72,000 metric tons* of spent fuel will have been discharged from commercial nuclear power plants by the year 2000. Waste from reprocessing this spent fuel, along with some quantity of spent fuel declared as waste and existing commercial high-level waste at West Valley, N.Y., will require disposal. Therefore, the NWTS project is planning to have the first repository available with the capacity to dispose of that quantity of waste by 1998. Subsequent repositories, although not included in the scope of this project plan, will be developed on a regional basis to meet national requirements.

Total project costs through completion of construction of the first repository (1998) is approximately $6 billion, of which about $4.4 billion will be for R&D and $1.6 billion will be for the construction of the repository. The total cost of operations and decommissioning is not included in the total project costs at this time because of current uncertainty in the length of the operation period. Estimates of the total project costs will be updated on an approximately annual basis. It is intended that all costs to the government for disposal of wastes generated from commercial activities be fully recovered. Mechanisms by which this can be done are being developed outside the scope of this project.

Attachment 1 provides a summary of project baseline information consistent with the objectives outlined above. This information includes baseline technical, schedule, and cost data.

*Spent Fuel and Radioactive Waste Inventories and Projections as of 12/31/80, September 1981, U.S. DOE, DOE/NE-0017.

2 Technical Plan

2.1 Project Description - What is to Be Done

In order to achieve its objectives, the NWTS project will provide for the isolation of nuclear wastes in deep, mined geologic repositories with a number of natural and man-made barriers between the wastes (emplaced in a host rock 1,000 to 4,000 feet below the earth's surface) and the uppermost strata of the earth's surface (the biosphere). These barriers include the waste form itself, the waste packaging, the backfills and seals of the repository, the host rock immediately surrounding the wastes, the geologic and hydrologic regime surrounding the host rock, and institutional barriers.

The design, licensing, construction, operation, and decommissioning of geologic repositories for radioactive waste will require (1) selection of suitable sites for development and (2) development of the technology to ensure permanent isolation of the wastes at these sites. The project activities to meet these requirements will be conducted in accordance with applicable U.S. Nuclear Regulatory Commission (NRC) and U.S. Environmental Protection Agency (EPA) regulations, and other applicable requirements. The work breakdown structure (WBS) that has been developed for the definition, budgeting, scheduling, and overall management of project activities provides for eight categories of tasks. This WBS is shown in Attachment 2 to the third level.

The NWTS project will be implemented in four phases in accordance with DOE systems acquisition guidelines. The phases are Advanced Development, Engineering Development, Operation, and Decommissioning. At the present time, the NWTS project is in the Advanced Development Phase. The objectives and activities of each of these phases are presented in this section.

2.1.1 Advanced Development Phase

The objectives of the Advanced Development Phase are:

- To conduct geologic studies to identify potential repository sites in varied geographic regions, considering alternative combinations of surface and geologic/hydrologic environments.
- To provide the capability to analyze behavior mechanisms for radioactive waste placed in such candidate environments.

- To develop other technologies, such as waste handling and packaging and repository sealing, necessary to support the design, construction, operation, and decommissioning of repositories.

The NWTS project began the Advanced Development Phase in 1976 and is presently planning and conducting efforts in this phase. During the remainder of the Advanced Development Phase, the following activities will take place:

- Three specific sites will be identified in the near-term at which exploratory shafts (ES), as part of detailed site characterization required by NRC's licensing regulations to establish repository site suitability, will be constructed to the proposed repository horizon. Site characterization reports will be prepared and submitted to NRC in accordance with 10 CFR 60. Presently planned locations for these three sites are:
  a. In basalt flows on the Hanford Site in the state of Washington
  b. In volcanic tuff on or near the Nevada Test Site
  c. In a salt formation at a site to be determined. Investigations are currently under way in salt domes in the Gulf Interior Region of Texas, Louisiana, and Mississippi, and bedded salt in the Paradox Basin in Utah and the Permian Basin in Texas.

  The three sites identified for detailed characterization will be potential candidates for selection of the first repository location. Exploratory shaft construction at all three sites will be completed in FY 1985. Steps will be taken to assure access and acquisition of land as necessary to allow these site characterization activities to be conducted and to preserve the integrity of the sites for future consideration as repository locations.
- Following initial (Phase I) at-depth examination of the three sites, selection of one of three sites for the development of an unlicensed Test and Evaluation Facility (TEF) will be made. This will be a major decision point in this phase, and the decision will be supported by a site-specific EIS.
- The TEF will be developed to refine repository construction methods, operation, and safety procedures, and to evaluate waste-handling and emplacement equipment under real-time, at-depth,

prototypical conditions. Placement of up to 300 packages of solidified high-level wastes will take place at the TEF. The packages will incorporate multiple barriers, will be completely monitored, and will be designed for retrieval. Methods for handling, emplacing, and retrieving large quantities of waste packages will be developed, and experience will be gained in keeping worker exposure to radiation as low as reasonably achievable. The information gained in the TEF will be used to assist DOE and NRC in establishing the conditions and requirements acceptable for repository licensing.

- Further work (Phase II) on detailed site characterization will continue in parallel at all three sites as a basis for determining suitability for a full-scaled, licensed repository. The knowledge acquired on each site will be used in the preparation of an updated site characterization report (SCR) that will be submitted to NRC in compliance with 10 CFR 60.
- Technology will be developed to support the ES, TEF, and the design, licensing, construction, operation, and decommissioning of repositories. Waste packages will be designed, tested, and qualified for use in the TEF and repositories; this effort involves design and development of waste package systems specific to each potential host rock, barrier materials testing, and laboratory and engineering scale testing of package systems to assure required long-lived integrity. Other technology development efforts include repository sealing, equipment and instrumentation development, performance assessment modeling for prediction of radionuclide containment, release, and migration, and support of transportation system development.
- Longer term evaluation of other potential repository sites in diverse geologic media will continue in order to provide additional siting alternatives for selection of the second and subsequent repository locations. Specifically, evaluations of granitic rocks and other salt formations (in addition to evaluation of the first salt site discussed earlier) within the conterminous U.S. will be conducted on a time frame as shown on the project schedule in Attachment 4 (see "sites 4 and 5").

- The evaluation of the technical and environmental feasibility of the subseabed disposal concept will continue. This concept will be studied to evaluate its role as a complement to mined geologic repositories and for supplementary disposal of particular waste components of the commercial nuclear fuel cycle.
- Findings that the candidate sites are technically suitable as alternatives for a repository siting recommendation will be made, based on site performance criteria and other technical considerations.
- A recommendation for public review and comment on the site for the first repository is scheduled in 1987. The candidate sites not selected for this repository may become alternatives for subsequent selections. A draft site-specific Environmental Impact Statement (EIS) will be prepared and published concurrently, assessing impacts of the repository at the chosen and alternative sites.

2.1.2 Engineering Development Phase

The objectives of the Engineering Development Phase are:

- To finalize repository site selection.
- To perform at-depth testing and the engineering design and development to derive site-specific designs needed for the repository facilities and waste packages.
- To construct the repository in accordance with approved designs and procedures, and the NRC construction authorization, and to license it for operations.

This phase of the project will involve the systematic use of the knowledge and understanding gained from the previous phase to achieve the detailed design, construction, and checkout of the repository, and licensing by NRC. Significant activities after the key decision to start engineering development include:

- Issuance of a final site-specific EIS and site selection report to document the decision to develop the site recommended by DOE for a repository.
- Conducting additional at-depth tests at the selected repository site to develop engineering data for design and construction of the repository.

- Naming of a site project manager and establishment of a site project office.
- Conduct of Title I and Title II designs (includes establishing design bases) for the repository.
- Preparation and submittal of a license application, including an environmental report (ER) and preliminary safety analysis report (PSAR), to NRC.
- Permanent protection, either through withdrawal or acquisition of land, in and around the selected site, as necessary. Measures to protect candidate sites reserved as alternatives for later repository site selections may also be taken.
- Following NRC granting of a construction authorization, proceeding with construction and checkout of the repository.
- Applying to NRC for and subsequently receiving an operating license.

2.1.3 Operation Phase

The objective of the Operation Phase is:

- To operate a nuclear waste repository to receive and emplace waste, thereby providing a capability for isolation of existing and future high-level and TRU radioactive waste generated by civilian activities from the environment so that it will not pose any significant threat to public health and safety.

Activities in this phase start after a key decision to initiate repository operation. During this phase, wastes will be emplaced, monitored, and maintained retrievable for some period of time. Following filling of the repository, preparation of documentation to decommission the facility will be initiated.

2.1.4 Decommissioning Phase

The objectives of the Decommissioning Phase are:

- To successfully isolate the emplaced nuclear waste by sealing the repository and restoring the site at the surface.

- To take whatever additional measures are needed to assure that the sealed repository will not constitute a threat to future generations.

Activities in this phase start after a key decision to decommission the facility. This final phase covers those operations for the closure/sealing of the repository and implementation of monitoring plans. This phase ends with the restoration of the surface environment to its near original state, the imposition of long-term institutional controls, and the initiation of any post-closure surveillance or monitoring systems which might be required.

2.2 Status of Technology

The status of technology* for site selection and characterization is adequate to meet the requirements for disposing of radioactive wastes in a mined geologic repository. Current mathematical models adequately represent the more important processes involved in radionuclide transport; they aid generic studies by identifying factors most important to waste isolation, and can be used in site selection and in preliminary performance assessments of site-specific repository designs. Modeling of flow in fractured rock and of waste form/waste package/host rock chemical reactions is under development.

Transportation and emplacement of spent fuel have been demonstrated. Although evaluation of potential barrier materials is not complete, available data indicate that several materials are capable of maintaining their integrity for extended periods under anticipated repository conditions. The current research and development program is expected to provide the remaining technical data needed for the design, licensing, construction, operation, and decommissioning of a waste repository, as required.

Performance assessment analyses performed to date have given no indication that a mined geologic disposal system cannot isolate radioactive wastes safely. Current technology is available to pursue testing required to

---

*Status of Technology for Isolating High-Level Radioactive Wastes in Geologic Repositories, DOE/TIC 11207 (Draft), October 1980.

determine the acceptability of a site for a nuclear waste repository. The level of scientific knowledge is highest for salt, followed by granite (crystalline rock), basalt, and tuff. Any of these rocks can prove to be acceptable for a mined geologic repository; however, no rock type, on strictly technical grounds, is a first choice at this time.

Successful isolation of radioactive wastes appears feasible for periods of thousands of years, provided the systems approach is used and the influence of future human activities is minimized through adequate precautions. Some uncertainties about long-term performance can be bounded or compensated for and, therefore, need not be resolved in detail before selecting a site or constructing a repository; other uncertainties can be resolved during repository construction. Although reliance on appropriate engineering practices and multiple barriers can compensate for a lack of complete knowledge and predictive capability, some residual uncertainties will always remain; however, it is believed that these uncertainties can be bounded and are well within the limits of acceptable risk.

3 Risk Assessment

The risks associated with the NWTS project can be considered in terms of the risk of not achieving the overall project objective, as well as the risks of not achieving this objective within the specified cost and schedule. An analysis of these risks and the measures being taken to control them is presented in this section.

A significant factor in the selection of mined geologic disposal as the disposal concept to be implemented in the NWTS project was the low risk that the technology would not be available when needed.* It was determined that the required technology for this concept was available or close at hand. This risk is being controlled by focusing project technical activities on timely satisfaction of the remaining technology needs.

---

*Final Environmental Impact Statement, Management of Commercially Generated Radioactive Waste, DOE/EIS-0046F, October 1980.

The control of health and safety and environmental risks resulting from the radioactivity of the waste is a key requirement in attaining the NWTS project objective. The health and safety aspects of normal and abnormal operation are being studied extensively. Events and processes which could potentially lead to hazardous conditions are being identified and criteria on siting and design established to control the risk. Conservative design practice and operational restrictions will be imposed to compensate for any residual uncertainties. An EIS will be prepared on the site-specific environmental impacts of the proposed repository, including an analysis of health and safety risks. Per 10 CFR 60.21, analyses of health and safety risks will be documented in the environmental report and preliminary/final safety analysis reports. Risks to construction and repository workers will be minimized by careful design and accident prevention programs. Measures will be taken to make future generations aware of the existence of repositories and to minimize the likelihood of accidental, human-induced loss of containment.

A principal source of risk in the NWTS project is institutional. One type of institutional risk being addressed in the project is related to satisfying the concerns and socioeconomic needs of affected states, local governments, and the public. If these concerns and needs are not adequately addressed, loss of a possible repository site or schedule delays are possible. The project seeks to develop sound working relationships with these parties through an interactive consultation process. Contacts are established with the affected state and local governments at the outset of site exploration activities. There are opportunities for exchange of information, review, and consultation on site characterization plans, activities, and decisions, and participation in the decision-making process. Grants to states are made available to support funding of these various efforts. However, the lack of a conflict resolution mechanism makes this area a source of moderate risk. The NWTS project is exploring various options, including legislation to address this issue.

Design, construction, operation, and decommissioning of NWTS facilities are required to be in compliance with yet-to-be promulgated EPA and NRC requirements and criteria designed to ensure public health and safety and protection of the environment. There is a risk of schedule delays and cost increases due to changes which may be necessary to bring the project in

compliance with the requirements when promulgated. The NWTS project is controlling this risk by being proactive in technical information exchange with the NRC and EPA and by developing its own soundly based technical criteria to control the health and safety risks.

Multiple sites are being explored and examined so that any site can be abandoned if the results show that the site is unsuitable for a repository, and to present a number of alternative sites from which to choose the site for the first repository. Longer term efforts at other sites and in other media are continued in parallel to provide alternatives for siting subsequent repositories. The ability of a particular repository site to satisfy health and safety criteria, and the cost and schedule for construction of a facility at that site, depend partially upon geologic and hydrologic characteristics that can only be confirmed by detailed characterization studies. The associated project risk decreases as these field studies are conducted. The project strategy controls this risk through extensive study and analysis of site characteristics prior to commitment of resources to develop the facility. There is also some risk related to the timely access or withdrawal/acquisition of land to support the development of a facility at a particular site; continuing to foster sound working relationships with the states in the siting program and planning for necessary land access, etc., with sufficient lead times should minimize this risk.

A variety of factors in addition to those discussed above contribute to an overall moderate risk in the area of project costs. The cost estimate presented in this plan is, consistent with the current advanced development status of the project, not based on a detailed site-specific repository design. Many decisions (e.g., repository geologic media, thermal loading, intermixing of TRU and HLW packages) which impact cost are yet to be made. The project cost risk will be reevaluated and estimates of costs updated on an approximately annual basis.

The NWTS project utilizes strong project management tools and techniques for the planning, monitoring, and control of project activities. These include controlled technical and cost/schedule baselines. Independent peer review is used extensively to supplement staff evaluation of technical planning and conclusions. Through these means timely identification of potential problems is made and corrective actions are taken to minimize overall project risk.

4 Management Approach

Responsibility for the NWTS mission has been assigned through the Under Secretary (US) to the Assistant Secretary for Nuclear Energy (ASNE). Under the ASNE, the Deputy Assistant Secretary for Nuclear Waste Management and Fuel Cycle Programs is responsible for executing policy and managing all aspects of DOE's commercial nuclear waste management program. Within the Office of Nuclear Waste Management and Fuel Cycle Programs (NWM & FCP), the Director of the Office of Waste Isolation (OWI) is responsible for overall direction of the NWTS project and, therefore, is designated as the Project Manager. The Director of OWI is W. Wade Ballard, Jr. Individual site project managers will be named when site project offices are established at each selected repository project site.

The Director of the Office of Waste Isolation provides: (1) program policy, objectives, and funding requirements; (2) the interface with the Nuclear Regulatory Commission on licensing matters; and (3) the planning and implementation of the Congressional and public affairs portion of the project in close coordination with the Assistant Secretary for Congressional, Intergovernmental, and Public Affairs. A program support contractor, the Office of NWTS Integration (ONI), supports the Office of Waste Isolation in the planning, organizing, integrating, and monitoring of activities across the project elements and interfacing programs. The responsibilities of ONI are delineated in the ONI Management Charter now under final DOE Headquarters (HQ) review.

The activities in the NWTS project are carried out by four project elements which operate through DOE Operations Offices, as shown in Attachment 3. The project elements and their major responsibilities are as follows:

- Basalt Waste Isolation Project (BWIP)
- Nevada Nuclear Waste Storage Investigations (NNWSI)
- Office of Nuclear Waste Isolation (ONWI)
- Subseabed Disposal Program (SDP).

BWIP and NNWSI are evaluating potential repository sites in basalt and tuff, respectively, at the DOE Hanford Site and the Nevada Test Site (NTS). ONWI is investigating other potential sites, in salt and crystalline rock, within the

RADIONUCLIDE MIGRATION: LABORATORY EXPERIMENTS WITH ISOLATED FRACTURES,* R. S. Rundberg, J. L. Thompson, and S. Maestas (Los Alamos National Laboratory).

Laboratory experiments examining flow and element migration in rocks containing isolated fractures have been initiated at Los Alamos National Laboratory. Techniques are being developed to establish simple fracture flow systems which are appropriate to models using analytical solutions to the matrix diffusion — flow equations, such as those of I. Neretnieks.[1] These experiments are intended to be intermediate steps toward larger scale field experiments where it may become more difficult to establish and control the parameters important to nuclide migration in fractured media. Hopefully, these experiments will validate the simple models and help to identify the more significant properties which need to be characterized in order to make accurate predictions.

Laboratory experiments have been run on fractures ranging in size from 1 to 20 cm in length. The hydraulic flow in these fractures was studied to provide the effective apertures. The flows established in these fracture systems are similar to those in the granite fracture flow experiments of Witherspoon et al.[2] Traced solutions containing ^{85}Sr and ^{137}Cs were flowed through a 15.6 μm fracture in Climax Stock granite. The results of the elution agree with a matrix diffusion calculation based on independent measurements of K_d and porosity. Similar experiments are currently underway on various rock types, including Climax Stock granite, Stripa granite, welded tuff (Tram member, Yucca Mountain, Nevada Test Site) and G-tunnel tuff (a highly zeolitized non-welded ash full tuff from the Nevada Test Site).

*This work was supported by the U.S. Department of Energy.
[1] I. Neretnieks, "Diffusion in the Rock Matrix: An Important Factor in Radionuclide Retardation?", J. Geophysical Research 85, 4379 (1980).
[2] P. A. Witherspoon, J. S. Y. Wang, K. Iwai, and J. E. Gale, "Validity of the Cubic Law for Fluid Flow in a Deformable Rock Fracture," Lawrence Berkeley National Laboratory Report (October 1979) LBL-9557.

EFFECT OF ORGANIC COMPLEXING AGENTS AND TEMPERATURE ON ADSORPTION OF RADIONUCLIDES: IMPLICATIONS FOR DISPOSAL OF RADIOACTIVE WASTE. A.Maest, S.Tréhu, E.Dillon, D.Crerar, Princeton Univ. Geology Dept. Princeton, NJ 08544; J.Means, Battelle-Columbus Labs.

Laboratory adsorption studies have been conducted using clays of varying cation exchange capacities and solutions containing an organic complexing agent (oxalic and gallic acids and EDTA at $2x10^{-3}$m) and four metals representative of nuclear waste: Cs,Sr,Co and U at $5x10^{-4}$m. Experiments were run from 25°-250°C in shaker baths and high T and P rocking autoclaves. The clay(mg) to solution(ml) ratios were 5:100, 1:100 and 0.5:100 for kaolinite, illite and montmorillonite respectively. The solutions were buffered at constant pH (pH=5.0 at 25°C). Adsorption onto the clays initially decreased with increasing temperature for all metals and then began to increase markedly at different temperatures, determined by the ionic radii and valencies of the metals. The adsorption increases for U,Co,Sr and Cs began at ~75°,100°, 150° and 200°C respectively and reached nearly 100% adsorption for all metals except Cs. We interpret this as the result of several competing mechanisms: an initial adsorption onto the clay's surfaces or in more internal, possibly Al, structural sites with later interlayer adsorption. Above 200°C the crystallinity of the clays is diminished, apparently opening more sites for adsorption. Results from solutions with complexing agents imply that they increase metal adsorption below their degradation temperatures (~100° for oxalic, 150° for gallic and 200°C for EDTA), possibly by creating a charge inbalance in the clays as a result of their high affinity for Al. Co, however was held strongly in solution with EDTA below the degradation temperature of the complexing agent the reached nearly 100% adsorption after the Co-EDTA complex broke down. Thermal degradation of chelating agents used in power plant clean-ups before burial with nuclear wastes would inhibit organically-bound radionuclides from remaining in solution and therefore increase the likelihood of sorption onto geologic media in the repository.

C1 DIFFUSION MEASUREMENTS IN COMPACTED BENTONITE. B. Torstenfelt, K. Andersson, B. Allard and U. Olofsson, Department of Nuclear Chemistry, Chalmers University of Technology, S-412 96 Göteborg, Sweden.

A cell for measurements of radionuclide diffusion in compacted clay has been designed. In this cell the diffusing element is introduced in the middle of a cylindrical clay body, and water is allowed to enter into the clay from both ends. From determinations of radionuclide concentration profiles in the clay diffusivities can be calculated, and the mass transfer rate of the radionuclide through compacted clay e.g. in a backfill barrier in a waste repository can be quantitatively described.

Diffusion data for some fission products and americium in compacted bentonite are given in the paper, and the possibilities of introducing additives to the clay that would enhance the retention of specific radionuclides, e.g. the actinides, are discussed.

C2 DISSOLUTION OF SIMULATED SPENT FUEL IN HYDROTHERMAL SOLUTION. W. Phelps Freeborn, Sridhar Komarneni, Barry E. Scheetz, and William B. White, Materials Research Laboratory, The Pennsylvania State University, University Park, PA 16802.

Spent fuel is difficult to simulate because of the unique microstructure that develops in reactor fuel pellets. However, the chemical reaction of spent fuel and water should be amenable to investigation. Three compositions were prepared: pure $UO_{2.08}$, $UO_{2.08}$ plus 3.5 wt% simulated fission products (corresponding to 33,000 MWd/MTU burnup), and $UO_{2.08}$ plus 40 wt% simulated fission products. The fission products (19 elements) were mixed, fired in hydrogen to simulate appropriate reducing conditions, mixed with UO_2 and fired again in sealed capsules. These materials were reacted at 100°, 200° and 300°C with water, a saturated NaCl brine and a $MgCl_2/CaCl_2/KCl/NaCl$ bittern brine at 300 bars pressure for periods of time from 1 to 5 weeks. Product solids were examined by SEM and X-ray methods; product solutions were analysed for extracted uranium and fission product elements.

The resistance of UO_2 to solution, especially under reducing conditions, however, does not prevent the extraction of most of the fission product elements even at low temperatures. The alkali and alkaline earth elements are very readily extracted; effectively all of the cesium has been removed into solution at 200°C. Molybdenum and the rare earths are also extracted although less effectively while little zirconium is removed. A substantial amount of uranium is taken into solution presumably through an oxidation mechanism and by chloride complexing. The bittern brine is a more effective solvent than water and particularly so for uranium.

IIGRATION RATES OF BRINE INCLUSIONS IN SINGLE CRYSTALS OF NaCl. I-Ming Chou, U.S. Geological Survey, 959 National Center, Reston, VA 22092. C3

ock-salt deposits have been considered as a possible medium for the permanent torage of high-level radioactive wastes and spent fuel. Brine inclusions present n natural salt can migrate if the temperature gradients in the vicinity of the adioactive waste are large enough. In order to better assess the problem, the iigration rate of these brine inclusions must be esimated under various repository onditions. Among the existing models of the migration process, the one presented y Anthony and Cline (1971, J. Appl. Phys. 42, 3380-3387) is considered as being he most complete, because it accounts for most of the phenomena known to occur in he migration process. However, application of their model is difficult because of n insufficient data base. By utilizing recent data for halite saturated brines and he model of Anthony and Cline, the effect of brine composition on the migration ate of inclusions was evaluated at 50° and 100°C. Because Soret coefficients (σ) if salt in brines are not known, the Soret coefficient must be treated as a variable. n order to simplify the calculation, the interface kinetics are neglected. Then he isothermal maximum migration rate predicted by Anthony and Cline's model reduces o a simple linear function of σ for each brine composition. Although the estimated aximum migration rates for WIPP-A, NRT-6, and 2.41 m $MgCl_2$ brines are of the same rder of magnitude, they differ markedly from those of pure NaCl solution (except in narrow region of σ values). At 100°C, these calculated migration rates are considerably higher than those measured experimentally. This discrepancy indicates ither that the neglected interface kinetics or the presence of a gas phase in the easured inclusions might be important retarding factors for the migration rates of nclusions, or that the model used is not perfect.

CORROSION OF STRUCTURAL MATERIALS IN CLAY ENVIRONMENTS, J. Dresselaers, F. Casteels, H. Tas, SCK/CEN, Metallurgy B.2400 Mol, Belgium. C4

Disposal of radioactive waste in stable geological formations is presently considered as one of the most safe methods for the isolation of radionuclides contained in the waste from the ecosphere. In Belgium the use of a geological clay formation is studied for the burial of radioactive waste.

The corrosion rate and mechanism of construction materials of galleries and casings for boreholes (eventually to be used as consumable overpack for 12 HLW canisters) has been evaluated after corrosion experiments in direct contact with clay, in a humid clay atmosphere and in a gas corrosion chamber. The experiments in direct contact with clay have been carried out "in situ" and in closed stainless steel cylinders.

The experiments have been carried out in the temperature range between 13°C (temperature of the clay formation) and 70°C.

The exposed specimens consists of a series of alloyed and non-alloyed cast nodular irons protected with different anti-corrosive protections (hot dipping, paintings, plastics, bitumen).

The experiments lasted up to several years.

The corrosion attacks are interpreted in terms of the Cr, Ni and Si content of the alloyed cast nodular irons and the weight, adhesion and porosity of the coating.

The aggressiveness of the different clay environments is interpreted in terms of the presence of corrosive anions and cations and characteristics of the clay such as electrical resistivity, pH, redox potential and redox capacity, sulphide and moisture content, bacterial activity and the nature of corrosive products escaping from the clay under influence of a temperature effect.

C5 THE DISSOLUTION OF IRRADIATED FUEL UNDER HYDROTHERMAL CONDITIONS. L.H. Johnson, Atomic Energy of Canada Limited, Whiteshell Nuclear Research Establishment, Pinawa, Manitoba, ROE 1LO

Uranium dioxide, the major component of irradiated CANDU fuel, shows good hydrothermal stability under the appropriate redox conditions. The thermodynamic stability of UO_2 in brine and in low ionic strength granite groundwater at 150°C under oxidizing and reducing conditions is briefly reviewed in order to provide a basis for the discussion of results of irradiated fuel dissolution experiments. Fuel chemistry characteristics that influence the kinetics of fission product release are also discussed.

Experimental studies have demonstrated the influence of redox chemistry on irradiated fuel dissolution rates at 150°C, where matrix dissolution rates are decreased by a factor of 10-100 times in going from oxidizing to reducing conditions. Under oxidizing conditions, UO_2 matrix dissolution rates are not strongly temperature dependent. Observed rates, inferred from ^{90}Sr release to solution, indicate that the rate increases by a factor of about 10 between 25 and 150°C. These studies have indicated the need for further work in certain areas, in particular on the relative amounts of important fission products released through leaching and matrix dissolution.

C6 RETARDATION, PERMEABILITY, AND SWELLING OF CANDIDATE BACKFILL MATERIALS, J. H. Westsik, Jr., L. A. Bray, F. N. Hodges, and E. J. Wheelwright, Battelle Pacific Northwest Laboratory, Richland, WA 99352

Batch distribution coefficients (Kd) for I-125, Sr-85, Cs-137, and Tc-99, with candidate backfill material and synthetic Grande Ronde groundwater have been measured for contact times up to 28 days. The results show that cesium and strontium are retained by the clays and zeolites (Kd - 10^{-10}). Iodine was retained by silver-treated Na-bentonite (Kd = 270), and technetium was not retained on any material tested.

Selected flow-through experiments have been directed at evaluating sodium and calcium bentonite clays and clay/sand mixtures. Synthetic Grande Ronde groundwater shows a significant change in composition as cations and anions are being sorbed and/or released during flow through the clay/sand compacts.

Hydraulic conductivities and swelling pressures are being determined as functions of the density of the compacted backfill, temperature, radiation dose, hydraulic head, the chemical composition of the permeating fluid. Bentonite clays and bentonite/sand mixtures have received initial emphasis in the program. For samples compacted by the same pressure, the hydraulic conductivity increased and the swelling pressure decreased as sand content increased in the mixture. In experiments using synthetic Grande Ronde groundwater, sodium bentonite has shown the lowest hydraulic conductivity of 7×10^{-13} cm/sec. Calcium bentonite has a hydraulic conductiviity roughly twice that of the sodium bentonite. A 1:1 mixture of sodium bentonite and 80-120 mesh quartz sand had a hydraulic conductivity of 4×10^{-12} cm/sec, and a 1:3 clay/sand mixture had a hydraulic conductivity of 7×10^{-10} cm/sec. In all cases the measured density was 2.1 g/cm^3.

PNL-SANDIA HLW PACKAGE INTERACTIONS TEST: PHASE ONE. Martin A. Molecke, Sandia National Laboratories, Albuquerque, New Mexico 87185, Don J. Bradley and John W. Shade, Battelle Pacific Northwest Laboratory, Richland, Washington, 99352

The first phase of a high-level waste (HLW) package interactions test in a salt environment has been completed. The test system consisted of PNL 76-68 HLW glass (loaded with inactive fission products and ^{238}U) surrounded by a stainless steel waste canister, a TiCode-12 overpack, a bentonite/sand backfill, excess brine leachant, and a bedded rock salt container, all held within a 19-liter autoclave. All components were physically compromised in order to force wasteform-barrier-salt geology interactions to occur during this 95-day, 250°C overtest. The empirical test data obtained (analyses of leachant and all barrier surfaces) will be compared with supporting data from simpler test systems previously conducted. Contrary to previous predictions, the glass wasteform did not totally dissolve under the severe overtest conditions. The data presented will be used to predict interactions and performance under realistic salt repository environmental conditions.

8 EXPERIMENTAL STUDIES OF PROCESSES GOVERNING THE RELEASE AND SUBSEQUENT SORPTION OF NUCLIDES IN GRANITE REPOSITORIES. N. A. Chapman, P. J. Dudson, I. G. McKinley, D. Savage and J. M. West, Institute of Geological Sciences, Harwell Laboratory, Harwell, Oxfordshire, England

Progress over the last 18 months in laboratory and modelling studies of near-field behaviour in a granite repository is reviewed. Data are presented on the hydrothermal interactions of glass and granite, and the factors controlling the release source term. These include temperature dependent solubility of the glass and leachability of various nuclides, availability of groundwater, and formation of a crystalline secondary waste form. Estimates are made of release rates of significant nuclides to the rock. Data are presented on the interaction of Cs, Ce, Sr and Co with weathered granite, replicating altered fissure surfaces. The conventional K_D assumptions are replaced by parameters reflecting dependency of sorptive behaviour on temperature, concentration, pH and reaction direction. A simple model is used to demonstrate a means of synthesising these hydrothermal and waste-rock interaction data. Data suggest that disposal at temperatures initially in the order of 200°C may be feasible.

DEGRADATION OF ROCKS, THROUGH CRACKING CAUSED BY DIFFERENTIAL THERMAL EXPANSION, IN RELATION TO NUCLEAR WASTE REPOSITORIES. R.W. Davidge, J.R. McLaren and I. Titchell, Materials Development Division, AERE Harwell, OX11 ORA, U.K.

Heating to moderate temperatures, 50-500°C, gives a significant reduction in Young's modulus of rocks and it is likely that a major reason is grain boundary cracking. The cracking of grain boundary facets in polycrystalline multiphase materials showing anisotropic thermal expansion behaviour is controlled by several microstructural factors in addition to the intrinsic thermal and elastic properties. Of specific interest are the relative orientations of the two grains meeting at the facet, and the size of the facet; these factors thus introduce two statistical aspects to the problem. The criteria for facet fracture are critically reviewed. The statistical factors are then introduced to give quantitative data on crack density versus temperature. The theory is compared with the limited experimental measurements of Young's modulus for various rocks as a function of temperature. There is good qualitative agreement, and the additional (mainly microstructural) data required for a quantitative comparison is defined. Comment is made about the relevance of the effects to underground nuclear waste repositories.

C10 DEVELOPMENT OF ENGINEERED STRUCTURAL BARRIERS FOR NUCLEAR WASTE PACKAGES. Richard E. Westerman, Stanley G. Pitman, and Rebecca P. Elmore, Pacific Northwest Laboratory, Richland, Washington, 99352.

A program directed toward the development of licensable structural engineered barrier elements for long-lived nuclear waste packages has been underway at PNL since January, 1979. Emphasis is currently placed on metallic materials for canister, overpack, and hole sleeve applications. These barrier elements are expected to contribute substantially to a barrier system that will hold radioisotope release rates to near-zero values for time periods of approximately 1000 years.

The development of structural barriers involves the initial selection of candidate materials; their screening by means of mechanical and corrosion testing; their rigorous in-depth testing in repository-relevant environments, using accelerated tests that can be extrapolated to long time periods; and ultimately, their evaluation in conjunction with other package elements in site-equivalent environments for final package qualification.

Corrosion screening of alloys in 250°C autoclave tests, with basaltic groundwater environments, constitutes a major part of the program effort, with cast irons and cast steel presently receiving the most attention. The stress corrosion cracking of alloys is being investigated in basaltic groundwater by means of slow-strain-rate tests and cyclic loading of pre-cracked specimens. Emphasis is placed on titanium alloys in these studies.

E1 A STUDY OF ZIRCONOLITE FROM SRI LANKA AND SOUTH AFRICA. Richard F. Haaker, Rodney C. Ewing, Department of Geology, University of New Mexico, Albuquerque, NM; Thomas J. Headley, Sandia National Laboratory, Albuquerque, NM.

The long-term stability of radioactive waste forms such as SYNROC are difficult to assess from leaching and irradiation experiments which are of limited duration. One approach to the evaluation of crystalline radioactive waste forms is to examine the chemical alteration and radiation effects in natural materials of great age. Since natural analogues are the product of a long-term experiment, they may serve as useful standards against which the results of short-term experiments can be compared. Specimens of zirconolite from Sri Lanka and South Africa have been characterized by optical microscopy, TEM, SEM, the electron microprobe and x-ray diffraction. Both zirconolites are partially metamict but the original structure is restored by heating in air (1130°C - 4 hours). Optical examination of the Sri Lanka zirconolite suggests that little chemical alteration has occurred. Since the South African zirconolite is intergrown with other minerals, an estimate of the degree of alteration from optical data is difficult. SEM and TEM studies indicate that the Sri Lanka zirconolite has 1 micron and smaller spherical pores, perhaps due to helium accumulation. Similar porosity was not detected in the South African zirconolite. Both zirconolites have regions of crystalline and amorphous material. Optical, TEM, SEM, microprobe and x-ray diffraction data are presented and discussed. In light of these data, the long-term stability of zirconolite is evaluated. Limitations on the use of natural analogues for the prediction of long-term effects in synthetic phases are addressed.

E2 WITHDRAWN

(PERIMENTAL AND THEORETICAL ANALYSES OF SMALL-SCALE RADIONUCLIDE MIGRATION FIELD
(PERIMENTS,* K. L. Erickson and D. R. Fortney, Sandia National Laboratories,**
lbuquerque, New Mexico, 87185.

ecently completed analyses define criteria for selecting sites for small-scale
adionuclide migration field experiments and for choosing optimum experimental con-
itions for analysis of fundamental processes dominating radionuclide migration.
etailed characterization of geomedia and experiments to identify dominant chemical
henomena were used to develop a mathematical model for migration in an idealized
oint in nonwelded tuff. The model was applied to a hypothetical experiment. Criteria
or optimum operating conditions were developed in terms of the physical and chemical
arameters of the system, and effects of perturbations in parameter values were
xamined. The principal results of these analyses will be summarized. For example,
f sorption occurs by a process such as ion exchange, the parameters dominating
adionuclide transport in jointed rock are the sorption equilibrium distribution co-
fficient, the plate spacing of the joint, and the average fluid velocity. The
riteria for optimum operating conditions will be described using these parameters,
nd it will be shown that perturbations in the fluid velocity and plate spacing would
ave a larger impact on experimental results than the distribution coefficient. The
nalyses are being evaluated experimentally, and initial results are encouraging.
he experiments will be described, and available data discussed. Much of the work
nder consideration also would be applicable to other types of jointed rock, and
xamples will be given to illustrate how the physical and chemical properties of the
ock could affect experimental results. Extension of the small-scale analyses to
arger joint systems also will be considered.

This work supported by the U. S. Dept. of Energy under contract DE-AC04-DP00789.

* A U. S. Dept. of Energy facility.

EFERENCE LABORATORY TESTING FOR BACKFILL, Riley M. Chung and Felix Y. Yokel,
eotechnical Engineering Group, Structures and Materials Division, Center for
uilding Technology, National Bureau of Standards, Washington, D.C. 20234

elatively high magnitude of swelling and low hydraulic conductivity are two of
he performance requirements for the backfill placed around the radioactive waste
ackage for the underground nuclear waste storage scheme. Some studies have been
onducted in national laboratories and in other countries where the candidate
backfill materials were tested under many different conditions to determine the
expected range of these properties.

This paper briefly examines the variables that can be significant in the eval-
uation of these two properties and emphasizes the ones, including compaction
effort, methods of specimen compaction, and the moisture content at the time
of compaction, which for one reason or another have not been considered in the
test programs.

E5 STUDIES OF LOCALIZED CORROSION AT HIGH TEMPERATURES OF PIT AND CREVICE MICRO-ENVIRONMENTS. V. Kay Hardman and Jerome Kruger, National Bureau of Standards, Bldg. 223, Rm. B254, Washington, DC 20234.

Studies of canister materials in nuclear waste repository environments have been undertaken to determine the susceptibility of these materials to localized corrosion for long times (> 1000 years). Since the local environments in pits and crevices can differ greatly from that found in the bulk solution, electrochemical and pH measurements in these microenvironments are taken to determine the corrosion rate and passive regions of stainless steel and Ti alloy materials. These metals are tested in sodium ion solutions with various concentrations of chlorine ions simulating possible environments found in nuclear waste repository sites. Repassivation properties of these materials are examined by scratch tests at several temperatures between 25 and 200°C.

This work is partially supported by the Office of Nuclear Waste Management.

E6 STRONTIUM-BASALT REACTIONS UNDER NUCLEAR WASTE REPOSITORY CONDITIONS. Sridhar Komarneni, Materials Research Laboratory, The Pennsylvania State University, University Park, PA 16802.

Hydrothermal interactions of Sr as strontium chloride or hydroxide with basalts or basalt phases were investigated at 300°C for 4 and/or 12 weeks under a confining pressure of 30MPa. The nature and extent of Sr immobilization varied with the Sr source and with the basalt or basalt phase as determined by the characterization of solid and solution reaction products. Basalts or basalt phases immobilized up to 99% of the added Sr. The main mechanism of Sr fixation with basalts or basalt phases excepting with the glass phase seems to be ion exchange when $SrCl_2$ was the Sr source. However, when $Sr(OH)_2$ reacted with basalts or basalt phases Sr was immobilized by the crystallization of strontium aluminosilicates such as $SrAl_2Si_2O_8$. These results of Sr-near-field interactions in a basalt repository indicate that basalt can be a good barrier for Sr migration from nuclear wastes.

E7 BACKFILL-WASTE INTERACTIONS UNDER RADWASTE REPOSITORY CONDITIONS. Noriaki Sasaki, Tokai Works, Power Reactor and Nuclear Fuel Development Corporation, Tokai-Mura, Ibaraki-Ken, Japan; Sridhar Komarneni, Barry E. Scheetz and Rustum Roy, Materials Research Laboratory, The Pennsylvania State University, University Park, PA 16802.

Alteration of backfill materials such as montmorillonite and clinoptilolite with and without the presence of simulated nuclear waste solids such as borosilicate glass, glass ceramic, sintered ceramic and supercalcine ceramic was investigated under hydrothermal conditions of 300°C and 300 bars for 4 weeks. Experiments were conducted under dry as well as wet repository conditions. Montmorillonite and clinoptilolite did not seem to alter under both dry and wet repository conditions as determined by XRD. However, the above backfill materials altered extensively in the presence of various simulated wastes under both dry and wet repository conditions. Backfill materials mainly altered to feldspar (oligoclase) in the presence of borosilicate glass and to analcime in the presence of sintered ceramic under both dry and wet repository conditions. Montmorillonite did not alter either in the presence of glass ceramic or supercalcine ceramic under both dry and wet repository conditions. However, clinoptilolite altered to analcime in the presence of glass ceramic or supercalcine ceramic under wet repository conditions. The above reactions of backfill materials with simulated wastes immobilized waste elements such as Cs, Mo, etc. by forming new phases such as analcime, oligoclase and powellite. In fact, analyses of product solutions from interaction runs indicated that the presence of backfill materials during the alteration of waste solids served to immobilize many waste elements. These results show that backfills can be effective barriers for nuclide migration if they come in contact with the wastes during the thermal period of the life of a repository.

_PHA DECAY SELF-DAMAGE IN CUBIC AND MONOCLINIC ZIRCONOLITE. F. W. Clinard, Jr., . C. Land, D. E. Peterson, D. L. Rohr, and R. B. Roof, University of California, os Alamos National Laboratory, Los Alamos, NM 87545.

he zirconolite phase of the SYNROC waste form will be a host to actinide isotopes hich generate self-damage by alpha decay. Natural zirconolite is often found in he partially-metamict condition from self-damage over geologic times. In this ondition the mineral exhibits a defect fluorite cubic structure which transforms o the normal monoclinic form on heating.

tudies are under way to investigate accelerated damage in zirconolite by doping ith $^{238}PuO_2$ (half-life=88 y). Samples containing 25 mol % PuO_2 in place of the 5 mol % ZrO_2 constituent have been fabricated and the resulting material, which s fluorite cubic, stored at ambient temperature for 200 days. Macroscopic welling saturated at 4.6 vol % at a dose of 1.8×10^{19} α decays/cm^3, while X-ray welling saturated at 2.2 vol % and 6×10^{18} α/cm^3. Aging studies are also being arried out on 5 mol % PuO_2--20 mol % ZrO_2 zirconolite, which is primarily onoclinic. Macroscopic swelling at 3×10^{18} α/cm^3 is approximately the same as hat for the cubic form. Other physical property observations for these materials, uch as degree of metamictization, defect content, fractographic characteristics, nd extent of microcracking, will also be described. Results are compared with hose obtained by others from natural and neutron-irradiated synthetic irconolite.

HEMICAL, PHYSICAL, AND ENGINEERING CHARACTERIZATION OF CANDIDATE BACKFILL CLAYS ND CLAY ADMIXTURES FOR A NUCLEAR WASTE REPOSITORY. Dr. S.K. Singh, President, aterials Research Laboratory Ltd., 9B Caesar Avenue, Nepean, Ontario K2G 0A8.

he investigation includes the characterization of Fourteen Canadian Clay and Clay dmixtures on the basis of their mineralogic, chemical, physical and engineering ehaviour under simulated environments of a nuclear waste repository located in a ranitic pluton.

he experimental data obtained during the study warrants further research on elected samples to prepare composites suitable for backfill and buffering purposes.

ADIATION INDUCED SODIUM METAL COLLOID FORMATION IN NATURAL ROCK SALT FROM SEVERAL EOLOGICAL LOCALITIES*--J. M. Loman, P. W. Levy and K. J. Swyler, Brookhaven lational Laboratory, Upton, New York 11973

he principal radiation damage induced in natural rock salt from various localities, ncluding potential repository sites, consists of point defects and, in the 115 to 250C range, colloidal sodium metal particles. With increasing dose the radiation nduced point defects, primarily F centers, grow to a saturation level, which is eached at doses of 10^7 to 10^8 rad, and that decreases with increasing temperature o a negligible level at 300C. Colloid growth follows a nucleation and grow curve ccurately described by a C(dose)^n expression at large times. For numerous atural rock salt samples irradiated at 150C, the temperature of the maximum colloid formation rate, the exponent n lies between 1.5 and 2.3 and the constant C varies y a factor of more than 30. The constant C, which is a measure of the total colloid that would be formed in a given repository, is related to the sample strain and the impurity and void content. The currently available data indicates that repository rock salt adjacent to planned waste canisters will develop 0.001 to 0.1 mole percent sodium.

*Supported by DOE Office of Nuclear Waste Isolation and DOE Division of Basic Energy Sciences under contract DE-AC02-76CH00016.

E11 STUDY OF POLYHALITE FROM THE WIPP SITE, NEW MEXICO. Douglas G. Brookins, Dept. of Geology, University of New Mexico, Albuqerque, NM 87131.

Polyhalite ($K_2Ca_2Mg(SO_4)_4 \cdot 2H_2O$) is an important mineral present in the bedded evaporites of the WIPP site, southeastern New Mexico. Because of the two structurally bonded molecules of water, it is important to know if this mineral formed at the time of evaporite formation or, as proposed by Tremba (1969) and Bodine (1978), much later and more or less continuously in post-evaporite time. If the latter mode of information is the case, then this implies a source of new water in order for polyhalite to form; in turn raising serious questions about water mobility and abundance in the bedded evaporites in the post-Late Permian. Polyhalite is not well suited for Rb-Sr geochronology as it contains too much common Sr; but it has been shown to be well suited for K-Ar geochronology (Brookins et al., 1980). Samples of polyhalite representing very pure material, mixtures with halides (NaCl and KCl), from xenoliths from rubble chimneys, from distorted layers near rubble chimneys, and from the recrystallized contact zone of a 33 MYBP lamprophyre dike have been studied. The K-Ar apparent ages range, for pure polyhalite, from 190 - 216 MYBP whereas impure polyhalites range from 155 - 205 MYBP. Both show a correlation of K-Ar age lowering as a function of halide content and, when age vs. % Na (proportional to halide content) plots are used, extrapolated ages of formation of 220 - 235 MYBP are obtained. These data convincingly show that the polyhalites formed at the time of sedimentation, except for the dike contact zone material, and not at much younger times (op. cit.). Finally, petrographic and thermodynamic information show that polyhalite is earlier than carnallite and some sylvite, which supports the K-Ar information.

E12 URANIUM-LEAD RADIOMETRIC AGE DETERMINATIONS OF NATURALLY OCCURRING U(VI) MINERALS: APPLICATION TO RADWASTE STORAGE. Douglas G. Brookins, Dept. Geology, Univ. of New Mexico, Albuquerque, NM 87131.

U(VI) bearing species are commonly formed under laboratory rock-water-HLW experimental conditions. Artificially produced Na-Cs-Rb weeksite (alkali uranyl hydrosilicate), and other U(VI) species, may be metastable for the Eh-pH conditions under which the chemical reactions may take place; and it is not known from the experimental work if these species will be subjected to alteration such that U is released. To address this problem naturally occurring U(VI) bearing minerals (uranophane and tyuyumanite) from the Grants mineral belt, New Mexico, have been studied for their U-Pb systematics. Samples from highly oxidized rocks and from overall chemically reducing environments have been studied. The samples from oxidized rocks are mixed with hematite-barite-gypsum-calcite-kaolinite gangue whereas those from reduced rocks are mixed with goethite-calcite gangue and are close to, but not in contact with, pyritiferous-organic carbon-uraniferous (U(IV))-chloritic ore. The reduced rock uranophane yields nearly concordant ^{238}U - ^{206}Pb and ^{235}U - ^{207}Pb ages of 8.32 and 8.33 MYBP (millions of years before the present); the oxidized rock uranophane and tyuyumanite yield more discordant ages (n = 8) between 7.4 and 2.7 MYBP. These data argue for long range stability of U(VI) hydrosilicates under both oxidizing and reducing conditions and imply that U(VI)-rich phases produced during experimental rock-water-HLW reactions are also stable under a wide range of Eh in the pH range of interest.

DIFFUSION OF NEPTUNYL(V)- AND PERTECHNETATE IONS IN MARINE SEDIMENTS*. F. Schreiner, S. Fried, and A. Friedman, Chemistry Division, Argonne National Laboratory, Argonne, IL 60439.

The diffusion of the $^{235}NpO_2^+$ and $^{95m}TcO_4^-$-ions has been measured directly in sample cylinders of two different sediments from the floor of the deep sea. In smectite-rich sediment not shielded from contact with atmospheric oxygen the following values were obtained for the effective diffusion coefficients of neptunyl- and pertechnetate-ions, respectively: $D_{eff}(NpO_2^+) = 1.5 \times 10^{-12} m^2 s^{-1}$ and $D_{eff}(TcO_4^-) = 3.2 \times 10^{-10} m^2 s^{-1}$. Under anoxic conditions in sediment with known reducing properties, the pertechnetate ion appears to undergo slow reduction and the effective diffusion coefficient of the reduced species of $D_{eff}(Tc,red) = 1.1 \times 10^{14} m^2 s^{-1}$ reflects a substantial decrease of the mobility of the lower-valent technetium.

*Work performed under the auspices of the Sandia National Laboratory Program Contract #74-1160.

RADIOGEOLOGICAL ASSESSMENT OF CANDIDATE SITES FOR NUCLEAR WASTE REPOSITORIES, EXEMPLIFIED BY STUDIES OF THE STRIPA PLUTON, SWEDEN. H. A. Wollenberg, S. Flexser and L. Andersson, Lawrence Berkeley Laboratory, University of California, Berkeley, CA 94720.

Investigation of candidate sites for nuclear waste isolation will require an assessment of their radiogeologic settings. Studies at the Stripa research facility in granitic rock of central Sweden incorporated the distribution and abundance of naturally occurring radioelements in rocks encompassing the underground experiments and in the accompanying fracture-controlled groundwater system. These studies showed that besides defining the natural radioactivity baseline upon which the effects of radioactive waste will be superimposed, radioelement distributions can be used to determine the apparent age of the groundwater and its flow paths. In crystalline rocks, where the groundwater systems are confined to the joints and fractures, the uranium daughter element, radon-222 in the water serves as a natural tracer to locate fractures along which significant flow is occurring and to measure the flow rates. The heat production from radioactive decay of uranium-238, thorium-232 and potassium-40 in the rock, combined with measurements of regional and local geothermal heat flow, permit calculation of the apparent size of the rock mass that will encompass the repository. This method is especially useful in terranes such as at Stripa where the contacts between plutons and older rocks are concealed.

NOVEL EXPERIMENTS FOR UNDERSTANDING THE SHALLOW LAND BURIAL OF LOW-LEVEL RADIOACTIVE WASTES. Gerald L. DePoorter, Los Alamos National Laboratory, Environmental Science Group, Los Alamos, NM 87545

Data on the basic processes that occur in the shallow land burial of low-level radioactive wastes are needed to engineer facilities with guaranteed performance, to validate models for system predicitons, and to provide input to models that consider contaminant pathways out of the facility. Two types of novel experiments that are providing experimental data on the basic processes in shallow land burial facilities are described in this paper. Generic experiments that give data on the movement of water and radionuclides and an experiment that is particularly important for semi-arid sites are described.

E16 IDENTIFICATION OF POSSIBLE HIGH INTEGRITY CONTAINERS FOR NUCLEAR WASTE DISPOSAL, J. Williams, Hittman Nuclear & Development Corporation, 9190 Red Branch Road, Columbia, MD 21045.

Increased concern by the State of South Carolina over the condition and capacity of the low-level radioactive waste burial site at Barnwell has prompted them to promulgate new regulations on waste burial containers. As of September 30, 1981, ion exchange resin and filter media waste with an activity of 1 uCi/cc or greater, or longer than five-year half-life isotopes disposed of at Barnwell, shall be solidified or confined in a "high integrity container." The materials and designs of these containers are required to provide waste isolation from the environment for a period of 300 years, and provide the structural integrity specified in 49 CFR 173.398. HITTMAN has been active in the design and development of containers suitable for this purpose with this paper detailing the analyses involved. Material selections were limited to stainless steel, fiberglass, and polyethylenes. Structural concerns focused on overpressure requirements, drop-testing requirements, and lifting capabilities. With a lifetime dose of up to 10^8 rads, the possibilities of radiation damage were considered. Preliminary selection of polyethylene was based on satisfactory resolution of these issues and economic factors. A brief discussion of the regulatory process encountered in this project is also included.

E17 PREDICTIONS OF RADIONUCLIDE MIGRATION RATES FOR A SUBSEABED REPOSITORY. Laurence H. Brush, Seabed Programs Division 4536, Sandia National Laboratories Albuquerque, NM 87185

Geochemical research supporting the U. S. Subseabed Disposal Program is described. Data from studies of high temperature interactions between sediments and porewater (seawater), and of sorption and diffusion of radionuclides in oxidized deep sea sediments are used, along with results from heat transfer studies, to predict migration rates of radionuclides in a subseabed repository.

Preliminary results for most radionuclides in oxidized sediments are very encouraging: fission products with moderate values of K_D (generally 10^2 to 10^5), and actinides with high values of K_D (generally 10^3 to 10^6) would not migrate significant distances before decaying to innocuous levels. ^{137}Cs, ^{90}Sr, and ^{239}Pu are among this group. The results for anionic species in oxidized sediments are less encouraging, but preliminary work with reduced sediments indicates that Tc can be effectively isolated.

Planning for a field verification of these laboratory and modeling studies is also described.

STOCHASTIC ANALYSIS OF GROUNDWATER FLOW AND CONTAMINANT TRANSPORT IN A FRACTURED ROCK SYSTEM. F. W. Schwartz, Dept. of Geology, Univ. of Alberta, Edmonton, Canada, L. Smith, Dept. of Geological Sciences, Univ. of British Columbia, Vancouver, Canada and A. S. Crowe, Dept. of Geology, Univ. of Alberta, Edmonton, Canada.

It has been possible to develop a stochastic model for groundwater flow and transport in a fractured rock system. Using a Monte Carlo method, a large number of statistically independent realizations of a fracture network are generated from a single set of distributions on parameters defining the fracture geometry. These parameters are the number of fractures, their location, orientation, length and aperture. By solving for hydraulic head at the fracture intersections, and using data on fracture aperture and porosity, seepage velocities may be calculated and the transport equation solved for each realization using a moving particle technique. Output from the Monte Carlo simulation consists of distributions on concentrations and various exit times. The applications are specifically designed to highlight the usefulness of the modeling concept. One very important application is to investigate the validity of the equivalent porous medium concept and the diffusional model of dispersion which are key assumptions in conventional continuum models. In addition, we illustrate the variety of plume shapes which can develop in fractured systems, and examine how parameters of fracture geometry act to control transport.

SOLIDIFICATION OF RADIOACTIVE WASTES WITH THERMOSETTING RESIN. M. Hayashi[1], K. Kobayashi[2], O. Okamoto[3], T. Kagawa[4], K. Wakamatsu[5], H. Irie[6], H. Matsuura[7], and Y. Nakayama[7]. Tokyo Electric Power Co.[1]; Tohoku Electric Power Co.[2]; Chubu Electric Power Co.[3]; Chugoku Electric Power Co.[4]; Japan Atomic Power Co.[5]; Toshiba Corp.[6]; NAIG Lab.[7]; Japan.

Dried simulated radioactive wastes were solidified with thermosetting resin and their properties were investigated with laboratory scale and real scale products through extensive testings; mechanical resistance, resistance to leaching and swelling in water, radiation resistance, fire resistance, resistance to temperature cycle. The typical results were as follows: over 500 kg/cm^2 of compressive strength, 10^{-6} -- 10^{-5} cm^2/day of diffusion constant for ^{137}Cs leaching from concentrated wastes solidified products, no remarkable change up to 5×10^8 RAD irradiation, thermal damage limited to the surface after the thermal test of IAEA transportation guide.

NUCLEAR WASTE DISPOSAL: THE INTERFACE BETWEEN PERFORMANCE ASSESSMENT AND RESEARCH R.B. Lyon, Whiteshell Nuclear Research Establishment, Atomic Energy of Canada Ltd., Pinawa, Manitoba ROE 1LO

The potential impact of the post-closure phase of a nuclear fuel waste disposal project is radiation dose to man. Radiation dose is estimated as the end product of a total systems analysis. Field and laboratory research must be assimilated in a form which can be accepted by the total systems analysis procedure. A central focus of this assimilation must be the consideration of uncertainties in the analysis and data used. Irreducible uncertainty arises because of the wide variability in natural systems and the unprecedented extrapolation into the distant future.

A procedure called systems variability analysis, incorporated in the SYVAC computer program, provides a framework for assimilation of the results of the field and laboratory research with a systematic treatment of uncertainty.

A SYVAC assessment of the post-closure performance of a Canadian nuclear waste disposal facility is presented with particular illustrations of the interface between the assessment models and data, and the field and laboratory research.

F4 PROCESSING OF TITANATES AND ZEOLITES, AND USE OF A CARTRIDGE SYSTEM FOR "REACTOR" WASTES, S. Forberg, T. Westermark, K. Svärdström, RIT, Stockholm; and L. Fälth, LIT, Lund, Sweden.

Hot isostatic pressing, hot uniaxial pressing, and cold pressing followed by simple sintering were technically compared as methods for conditioning of titanates and zeolites with radioactive and corrosion products having been transferred from organic ion exchangers, used for parallel purification of PWR water.

The pre-pressing conditioning, i.e. pre-heating, of the inorganic ion exchangers was studied, e.g. the bulk density obtained and the limitations from volatilization of radioactive species.

The density, the crystallographic composition and the leachability at Soxhlet tests and in cold (40°C) granitic ground water were studied.

Aiming at a simple, hygienic and fool-proof handling of radioactively charged inorganic ion exchangers, a cartridge system was designed for the whole route from packing of channel-free beds, via charging and emptying of the columns, to drying and preheating of the beds, and packing of the canisters for final conditioning (hot isostatic or hot uniaxial pressing or sintering).

After suitable measures, HLW trapped in titanate in such cartridges could form bodies with rutile as a continuous matrix and as a protective outer layer, a MEGANNITE, with an optional million-year warranty.

F5 PERFORMANCE OF ASPHALT AND CLAY LINERS AS A URANIUM MILL TAILINGS LEACHATE BARRIER. J. L. Buelt, V.Q. Hale and S. M. Barnes, Pacific Northwest Laboratory, Richland, Washington 99352.

Pacific Northwest Laboratory is evaluating the long-term effectiveness of various asphalt and clay liner materials as a radionuclide and process chemical barrier from uranium mill tailings. A field test is being conducted by monitoring asphalt and clay liners installed at the Grand Junction, Colorado tailings site. In addition eight prospective liners have undergone three months exposure to accelerated conditions to predict their behavior over a 1000 year period. High calcium leachates have been forced through thin layers of clay to determine the ability of the clay to resist ion exchange, which reduces its swelling capabilities. Asphalt liners are being exposed to elevated temperatures and increased strengths of oxidizing agents to accelerate their aging processs. The permeability coefficients measured during this exposure is then used to predict the expected lifetime. The analyses thus far show that clay soils with bentonite amendments and most asphalt compositions have good long-term performance characteristics.

EHAVIOR OF RADIONUCLIDES IN FUSED WASTE-DISPOSAL-SITE SOIL PRODUCED BY IN-SITU ITRIFICATION,* C. L. Timmerman, and J. L. Ruelt, Pacific Northwest Laboratory, ichland, Washington 99352

Radioactive wastes buried at waste disposal sites may require further stabilzation to secure the isolation of these wastes from the environment. One method of aste stabilization proposed is in-situ vitrification. In-situ vitrification is the n-place melting of the soil and encapsulation of the wastes in this molten soil to orm a glass and crystalline waste form. This paper will present analytical results nd predictions pertaining to the behavior of radionuclides in fused soil.

Solidification of contaminated soil is achieved by establishing an electric urrent between electrodes placed in the soil. The electric current produces a oule-heating effect that melts the soil and immobilizes any contaminants. Benchcale demonstrations of this concept with soil contaminated with nonradioactive esium, strontium, and ruthenium were performed. These demonstrations have provided preliminary indication of migration effects, off-gas contamination potential, and levitrification predictions. In addition to the nonradioactive data and tests, a rucible-scale melt of soil spiked with radioactive uranium, plutonium, and cesium as been leach tested. These nonradioactive and radioactive test results show good adionuclide retention in the fused soil, which indicates an excellent potential for pplication of in-situ vitrification to burial wastes and contaminated soils at aste disposal sites.

*Work performed for the U. S. Department of Energy under Contract DE-AC06-76RLO 1830

A COMPARATIVE ASSESSMENT OF TRU IMMOBILIZATION SYSTEMS - W. A. Ross, C. O. Harvey. R. O. Lokken, R. P. May, F. P. Roberts, C. L. Timmerman, R. L. Treat, and J. H. Westsik, Jr., Battelle-Pacific Northwest Laboratory, Richland, WA, 99352.

The results of a comparative assessment of six alternative TRU waste forms and seven processes for application at Rocky Flats will be reported. This assessment has the objective of evaluating both waste form containment properties and economics and safety of processes. The waste forms which are being evaluated include cast cement, pressed cement, borosilicate glass, aluminosilicate glass, sintered ceramics, and basalt glass ceramics. The waste forms have been prepared with sludge and plutonium containing incinerator ash from Rocky Flats. The waste form properties which are being evaluated include leachability, impact behavior, pressurization potential, and thermal stability. The waste form processes which are being evaluated for a new facility at Rocky Flats include batch cement mixing, cement mixing and pressing, in-can melting, joule heated melting, glass marbles, basalt glass ceramics, and pressed and sintered ceramics. The overall economics are based on capital cost, operating costs, D & D costs, transportation costs, and repository disposal costs. The importance of occupational exposure, industrial hazards, criticality safety, and quality assurance are also being determined.

F8 LOW-LEVEL WASTE DISPOSAL RISK ANALYSIS, S.G. Oston, C.M. Koplik, D.A. Ensminger, J.Y. Nalbandian, M.F. Kaplan, The Analytic Sciences Corporation, One Jacob Way Reading, Massachusetts 01867

The Analytic Sciences Corporation (TASC) is assessing the safety of low-level waste disposal. A computer code, LOTRAN, is under development to model the various pathways along which radionuclides can reach man via inhalation, ingestion, and direct exposure to gamma rays. These pathways include contaminated river water, well water, airborne dust, locally grown crops, and radon diffusion. The study considers the geologic and hydrologic properties of the disposal site and the properties of engineered barriers such as trench liners, caps, tailored waste forms, and high integrity containers used to inhibit the release of radionuclides.

Radiation doses to off-site individuals and to future land reclaimers are computed for a number of generic disposal sites located in both dry and humid environments. The predicted doses are to be compared to illustrative performance criteria developed under this project and to draft NRC criteria. Pesults will provide insight into those characteristics of low-level waste disposal systems most important to public safety and will allow trade-offs to be performed among natural and engineered features.

F9 LEACH STUDIES OF RADIONUCLIDES FROM SOLIDIFIED WASTES WITH THERMOSETTING RESIN. Kazunori Suzuki and Hiroshi Kuribayashi; JGC Corporation, Wataru Morimitsu and Isamu Ono; Industrial Research Institute of Kanagawa, Yokoham, Japan.

In Japan, low and intermediate-level radioactive wastes generated in nuclear power plants are solidified with cement or asphalt. Meanwhile, solidification processes using thermosetting or thermoplastic resin have recently been developed and are attracting much attention from the standpoints of volume reduction , immobilization of radionuclides and the physical and chemical qualities of the solidified products.

We, in collaboration with CEA/ECOPOL(France), have investigated and developed a new solidification process using thermosetting resin. This paper reports studies on the leachabilities of Co-60 and Cs-137 from speciemens solidified with thermosetting resin and evaluates the effects of chemical fixations on leach rate. This study concludes that insolubilization by nickel-ferrocyanide offers an effective chemical fixation of those radionuclides and is a recommendable pretreating method for radiowastes to be solidified.

PROCESSING TAILORED CERAMIC NUCLEAR WASTE FORMS, J. Flintoff, D. R. Clark, A. B. Harker, P. E. D. Morgan, and C. M. Jantzen, Rockwell Science Center.

Tailoring of Savannah River Defense Waste into a four-phase assemblage can be accomplished with minimal tailoring by hot isostatic pressing (HIPping). A fully dense fine grain ceramic is produced with chemical host sites for all radionuclides and no continuous glassy phases present. The tailoring formulations and resulting phase assemblages are given. The phases produced all have natural mineralogic analogues which are known to be geologically stable.

Processing of SRL Defense Waste into a ceramic monolith with the desired combination of mineralogic phases is dependent on the tailoring additives, redox control and consolidation conditions. Optimally, the Fe^{+3}/Fe^{+2} ratio should be <2/1 to take maximum advantage of the elemental content of the waste while uranium is desired to be in the tetravalent state. Use of wet chemical reducing agents before cold compaction avoids powder handling problems. The reduction occurs at 250°C and the redox control of the phase development has been examined at a sequence of temperatures up to 1200°C. Samples produced by this method show the same phases as ceramics made by calcining and reduction at the Fe/FeO buffer before and after final consolidation.

ON-LINE MEASUREMENTS OF THE VOLATILIZATION OF RUTHENIUM IN A VITRIFICATION PROCESS, R. Odoj, KFA-ICT, Postfach 1913, D-5170 Julich, FRG.

In the course of high level waste solidification, the volatilization of ruthenium causes some problems. Two mechanisms of release are possible: mechanical entrainment of particles and volatilization of constituents. To measure both factors, the investigations were carried out on-line with radioactive RuI03 als tracer.

Preliminary work indicates very low volatility at temperatures 100°C. Between 100°C and 200°C, the main fractional release of ruthenium takes place and amounts to more than 10%. Even at high temperatures during the glass melting, the fractional release was lower. The overall release of ruthenium during concentration, calcination and glass melting was about 16%. Denitration before calcination reduces the release of ruthenium about one order of magnitude. By adding iron powder as an additional agent, the volatilization was futher decreased even below 0.5%. The studies indicate that there should be other mechanisms and compounds besides the vaporization of RuO_4, which are responsible for the volatilization of ruthenium. The volatility of ruthenium in nitric acid solution is proportional to the volatilization of NO_3 — at temperatures of 100°C. This result is not related to the equilibrium vapor pressure of RuO_4 in this range of temperature.

CEMENTITIOUS RADIOACTIVE WASTE HOSTS FORMED UNDER ELEVATED TEMPERATURES AND PRESSURES (FUETAP CONCRETE), L. R. Dole, J. G. Moore, G. C. Rogers, G. A. West, H. E. Devaney, M. T. Morgan and J. H. Kessler, Chemical Technology Division, Oak Ridge National Laboratory, P. O. Box Y, Oak Ridge, TN 37830.

Concretes formed under elevated temperatures and pressures (FUETAP concretes) are effective hosts for transuranic (TRU) and high-level defense and commercial wastes.

Tailored cement formulations developed at ORNL use Portland cement, fly ash, sand, and clay additives. These FUETAP concretes are cured under mild autoclave conditions then dewatered to remove the unbound water. The resulting solids are strong (60-100 MPa, compressive strength), leach resistant (Pu at a rate of 10^{-11} g/cm^2-d) and radiolytically stable (lambda total = 0.005 molecule/100 eV) monolithic waste hosts.

This paper summarizes the results of the four-year old FUETAP development program. These continuing studies give data addressing the major questions concerning radioactive waste forms. These are: (1) leachability, (2) radiation stability, (3) thermal stability, (4) thermal conductivity, (5) impact strength, (6) permeability, (7) phase complexity and (8) effect of waste composition.

G4 BEHAVIOR OF RADIONUCLIDES AT SMELTING OF NONCOMBUSTIBLE SOLID WASTES. M. Osaki and S. Yokoi, Daido Steel Co., Ltd., Mutuno, Atuta-ku Nagoya Japan and M. Miyagawa, The Chubu Electric Power Co., Thoshin-cho, Higashi-ku Nagoya Japan.

Melting process of noncombustible solid wastes seems one of the most promising processes in point of volume reduction and immobilization. We have been developing the melting furnace of noncombustible solid wastes. However, melting process needs high temperature around 1500 °C, but the behavior of radionuclides in such process and at such temperature has not been fully examined. In part of developing stage, we melted small contaminated samples to examine the behavior of radionuclides. And melted samples were presented to leaching tests.

Samples were carbon steel, SUS304, silicaboard, concrete and zeolite and melted by laboratoryscale plasma arc melting furnace. After melting, radioactivity of melted samples and dusts were counted to calcurate the distribution of radionuclides. Melted samples were submerged to estimate leaching rate.

Many assorted samples were melted and we found that the vapour pressure of radionuclide determined the distribution between melted products and dusts. The standerd free energy of oxide formation and the quantity of stable nuclide of that radionuclide determined the distribution between metal and slag products. Leaching rate of melted products were sufficient low.

These works have been developed by Chubu, Tohoku, Tokyo, Kansai and Chugoku Electric Power Companies, Japan Atomic Power Co. and Daido Steel Co.

G5 VITRIFICATION OF HIGH-LEVEL RADIOACTIVE WASTE IN A SMALL-SCALE JOULE-HEATED CERAMIC MELTER. Gerald B. Woolsey, Russell E. Eibling, and Lien-Mow Lee, E.I. duPont de Nemours & Co., Savannah River Laboratory, Aiken, SC 29808.

Vitrification in borosilicate glass is the current reference process for the immobilization of Savannah River Plant high-level radioactive waste. Much of the chemistry and the design data for the process have been developed using experiments with nonradioactive, simulated waste. To demonstrate the equivalence of radioactive and simulated waste in the process, an integrated, shielded process has been constructed on a small scale at the Savannah River Laboratory. This process has established the equivalence of actual and synthetic waste in the reference process and yielded information on off-gases which was not readily accessible with simulated waste. This paper presents the results of vitrification studies on actual waste fed as a water slurry directly to the glass melter.

---

*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.

G6 METASTABLE LIQUID IMMISCIBILITY IN NUCLEAR-WASTE-GLASSES. John E. Engell and G. Roed, Institute of Mineral Indystry, The Technical University of Denmark, DK-2800 Lyngby, Denmark.

Changes in chemical durability of borate glasses upon phase separation is well-known. A similar behaviour of nuclear waste-glasses is clearly unwanted. The data presented concern potential nuclear-waste-glasses containing 9 wt% of simulated fission products and presently under consideration by the Swedish Nuclear Energy Association (KBS/SKBF). The glasses have compositions closely related to the ternary system Na_2O-B_2O_3-SiO_2. In this system a broad spectrum of glass compositions show metastable phase separation upon heat treatments in the range 550°C to 775°C. The 600°C-isotherm for the extension of this immiscibility dome into the multi-component system relevant to nuclear-waste-glasses has been located experimentally. The results give the lower limit for the alkali oxide content in useful waste-glasses of this type.

MERCURY REDUCTION AND REMOVAL DURING HIGH-LEVEL RADIOACTIVE WASTE PROCESSING AND VITRIFICATION.* Russell E. Eibling and John R. Fowler, Savannah River Laboratory, E. I. du Pont de Nemours & Co., Aiken, SC 29808.

At the Savannah River Plant, a process for immobilizing the high-level radioactive waste in borosilicate glass has been developed. This waste contains a substantial amount of mercury from separations processing. Since mercury will not remain in borosilicate glass at the processing temperature, mercury must be removed either before vitrification or must be handled in the off-gas system. A process has been developed to remove mercury by reduction with formic acid prior to vitrification. Studies with both simulated and actual waste are discussed. Additional benefits of formic acid treatment include improved sludge handling and glass melter redox control.

*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.

INTERIM WASTE FORMS FOR HIGH-LEVEL RADIOACTIVE WASTES: PROCESSING AND PROPERTIES,* G. Bandyopadhyay and S. M. Gehl, Materials Science Division, Argonne National Laboratory, 9700 South Cass Avenue, Argonne, Illinois 60439

Several interim waste forms have been investigated in the laboratory using simulated materials to study their suitability for application in handling and transportation of high-level radioactive wastes to terminal processing sites. In the fused-salt/sludge option, the neutralized supernatant liquid and the precipitated sludge are treated simultaneously to form fused-salt cakes. Silicate-based options, in which sodium silicate or cement-silicates act as binders for the calcined sludge, require prior separation of the sludge and the soluble radioactive constituents from the supernatant before the waste form can be prepared. The results from tests on simulated fused-salt waste forms indicated that the process simplicity of this option is partially offset by the high water solubility and hygroscopicity, which would make special precautions necessary during transportation and storage. The most promising silicate-based option is the cement-silicate process, in which the sludge is mixed with sodium silicate and $Ca(OH)_2$ powder and subsequently exposed to a controlled-humidity environment at room temperature to form the cementitious bond. Waste forms containing up to 90 wt % synthetic sludge and with sufficient mechanical and chemical stability have been prepared by the cement-silicate process.

*Work supported by the U. S. Department of Energy.

H3 IMMERSION AND LEACH TESTS ON SOLIDIFIED DECONTAMINATION WASTES FROM UNIT 1.* Robert E. Barletta, J. W. Adams, and R. E. Davis, Division of Waste Management, Brookhaven National Laboratory, Upton, NY 11973

In order to provide technical support to the NRC, Brookhaven National Laboratory has performed leach tests and immersion tests using organic liquids and organic saturated water of concentrated decontamination waste solidified using a vinyl ester-styrene binder.

The leach tests measured the release of Fe, Ni, and Co from these forms in deionized water, groundwater, and seawater. After 64 days, the mean fraction released normalized by V/S for iron was $5.1 \pm 1.9 \times 10^{-3}$ cm in deionized water, $7.1 \pm 1.1 \times 10^{-3}$ cm in groundwater, and $1.0 \pm 3.2 \times 10^{-3}$ cm in seawater. For nickel, 64 day release rates observed were $4.6 \pm 1.6 \times 10^{-3}$ cm, $4.6 \pm 1.0 \times 10^{-3}$ cm, and $5.9 \pm 0.6 \times 10^{-3}$ cm in deionized water, groundwater, and seawater, respectively. After 50 days, the ^{59}Fe release rates are $5.9 \pm 0.7 \times 10^{-3}$, $4.8 \pm 2.4 \times 10^{-3}$, and $2.8 \pm 1.0 \times 10^{-3}$ cm in deionized water, groundwater, and seawater, respectively. For ^{60}Co, the 50 day releases in the three respective leaching media are $6.0 \pm 1.7 \times 10^{-3}$ cm, $6.8 \pm 1.0 \times 10^{-3}$ cm, and $2.3 \pm 0.2 \times 10^{-3}$ cm.

Immersion tests of waste forms prepared at a solidification demonstration held at the Dresden Nuclear Power Station were conducted in toluene, xylene, and water saturated with toluene and xylene. Upon immersion of samples in the pure organics, large changes in sample volume and weight were observed. Total weight changes upon immersion of $9.6 \pm 0.3\%$ and $21.6 \pm 0.7\%$ were observed after 839 hours of immersion in xylene and toluene respectively. Air drying of the samples led to an overall weight loss of $23.5 \pm 0.7\%$ for xylene and $35.6 \pm 0.6\%$ for toluene. Qualitatively, similar changes were observed for immersion tests using organic saturated water. Severe sample deterioration was observed in this case, however. The cause of this deterioration is not known.

*Work carried out under the auspices of the Nuclear Regulatory Commission.

H4 TAILORED ION EXCHANGE RESINS FOR COMBINED CESIUM AND STRONTIUM REMOVAL FROM SOLUBLE DEFENSE HIGH-LEVEL WASTE*. Martha A. Ebra and Richard M. Wallace, Savannah River Laboratory, E. I. du Pont de Nemours & Co., Aiken, SC 29808.

Novel organic resins that achieve high selectivities for both cesium and strontium have been synthesized. They are condensation polymers of resorcinol and formaldehyde with attached chelating groups. Their column performance compares favorably with that of commercially available resins for either cesium or strontium removal. By combining Cs^+ and Sr^{2+} removal in the same bed, these resins can significantly reduce the size and complexity of proposed facilities for processing defense high-level waste.

*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.

H5 BIOLOGICAL INTRUSION OF LOW LEVEL WASTE TRENCH COVERS-T.E. Hakonson, Environmental Science Group, Los Alamos National Laboratory, Los Alamos, NM 87545

The long-term integrity of low-level waste shallow land burial sites is dependent on the interaction of physical, chemical, and biological factors that modify the waste containment system. Past research on low-level waste shallow land burial methods has emphasized physical (i.e. water infiltration, soil erosion) and chemical (radionuclide leaching) processes that can cause waste site failure and subsequent radionuclide transport.

The purpose of this paper is to demonstrate the need to consider biological processes as being potentially important in reducing the integrity of waste burial site cover treatments. Plants and animals not only can transport radionuclides to the ground surface via root systems and soil excavated from the cover profile by animal burrowing activities, but they modify physical and chemical processes within the cover profile by changing the water infiltration rates, soil erosion rates and chemical composition of the soil. Examples of the effects plants and animals have in modifying soil profiles are drawn from the ecological literature and from operating low-level waste sites where plants and animals have caused radionuclides to be transported from the waste trenches.

One approach to limiting biological intrusion thru the waste cover is to apply a barrier within the profile to limit root and animal penetration with depth. Experiments in the Los Alamos Experimental Engineered Test Facility were initiated to develop and evaluate biological barriers that are effective in minimizing intrusion into waste trenches. The experiments that are described employ four different candidate barrier materials of natural geologic origin. Experimental variables that will be evaluated, in addition to barrier type, are barrier depth and soil overburden depth.

H6 STABILITY OF I AND SR RADIOPHASES IN CEMENT MATRICES, M.W. Barnes, B.E. Scheetz, L.D. Wakeley, S.D. Atkinson and D.M. Roy, Materials Research Laboratory, The Pennsylvania State University, University Park, PA 16802.

The following study is a portion of a comprehensive research program that is examining the stability of a variety of nuclear waste forms. In addition to the bulk waste forms, important individual radiophases are being studied to obtain a more complete understanding of the behavior of the components of complex multiphase radwaste systems. The stability of the Sr-radiophase in supercalcine ceramic and a I-radiophase will be discussed.

The strontium radiophase in this study was a Sr-powellite and the iodine radiophase was I-sodalite. Each radiophase was incorporated into bulk compositionally adjusted portland and aluminate cements. Two processing variables were studied: curing at 60°C and warm pressing at 150°C and 345 MPa.

The I-sodalite appears to approach congruent dissolution in the portland cement. However the mechanism is masked by the presence of Na, Al, and Si in the aluminate cement. Soxhlet leach rates for I^- are 5 and 7.5 $gm^{-2}d^{-1}$ at three weeks in the aluminate and portland cement respectively. The Sr-powellite leaches incongruently at short times but the mechanism appears to change and trends to congruent dissolution with increasing leachtimes.

H7 THE SYSTEM $SrMoO_4$-$BaMoO_4$-$CaMoO_4$: COMPATIBILITY RELATIONS, THE IMPLICATIONS FOR SUPERCALCINE CERAMICS. Barry E. Scheetz, W.P. Freeborn, J. Pepin and W.B. White, Materials Research Laboratory, The Pennsylvania State University, University Park, PA 16802.

The stability of the tailored ceramic waste form designed for high level commercial nuclear waste was examined under hydrothermal conditions. This study indicated that the phase with the scheelite structure, dissolved and reprecipitated. The compositions of the initial and final phases were different, although of the same structure. A schematic phase diagram was proposed to explain the observations.

This study is then the experimental determination of those phase relations. Samples were heated following the schedule for the supercalcine ceramic. Characterization of these runs by x-ray diffraction suggests that the two phase region between the calcium and barium end members extends into the ternary system to about sixty weight per cent of the strontium end member.

Hydrothermal treatment of critical compositions at 300°C resulted in further separation of the compositions of the coexisting phases.

The change in composition observed in the previous work is thus confirmed to be due to the instability of the intermediate compositions formed at high temperature at the lower temperatures of the hydrothermal treatment.

H8 IRON-ENRICHED BASALT AND ITS APPLICATION TO THREE-MILE ISLAND AND WEST VALLEY RADIOACTIVE WASTE DISPOSAL. P. V. Kelsey, Jr., J. M. Welch, D. E. Owen, and J. E. Flinn, EG&G Idaho, Inc., P.O. Box 1625, Idaho Falls, ID 83415

Demonstration tests have been performed with iron-enriched basalt (IEB) as a dissolution/immobilization medium for some TMI and West Valley rad-waste disposal needs. Zeolite used for Cs and Sr decontamination of TMI containment water was mixed and melted with 20 wt. % additives to form IEB. Studies show that low Cs volatility during melting can be expected. The Cs leach rate from controlled-cool monoliths is 4×10^{-6} $g/cm^2 \cdot d$ (in 263 K deionized water). Dissolution of simulated TMI reactor core debris (UO_2 pellets clad with zircaloy) in IEB occurred in 2 h at 1773 K.

Simulated calcine from the neutralized high-level liquid waste stored at West Valley, N.Y., was incorporated into IEB at 25, 35, and 50 wt. % loadings. Leach tests on the resultant waste form materials showed no dependence on waste loading.

H9 LEACHING STUDIES OF Na-BEARING CRYSTALLINE PHASES IN NUCLEAR WASTE FORMS. E.R. Vance and T. Adl, Materials Research Laboratory, The Pennsylvania State University, University Park, PA 16802.

Since the most soluble element in nuclear waste forms will likely be Na, various possible Na-bearing crystalline phases which might occur in nuclear waste forms were studied. The following phases were investigated for leach behavior in various aqueous media: $NaAlSiO_4$, $NaNbO_3$, $Na_{0.5}La_{0.5}TiO_3$, $NaFeSi_2O_6$, $NaZr_2(PO_4)_3$, $NaFeO_2$, Na_2TiSiO_5, $Na_2Ti_3O_7$, $Na_5Ti_8O_{14}$, $Na_2Ti_2Si_2O_9$, Na_2ZrSiO_5, $Na_2ZrSi_4O_{11}$, $Na_4Al_3Si_3O_{12}X$ (X = Cl or I) and $(Cs,Na)AlSi_2O_6 \cdot xH_2O$.

Ease of ceramic synthesis and the likelihood of the various phases actually occurring in crystalline nuclear waste forms are discussed.

0 IMMOBILIZATION AND LEAKAGE OF KRYPTON ENCAPSULATED IN ZEOLITE AND GLASS, J. E. Tanner, J. A. Del Debbio, A. B. Christensen, and D. A. Knecht, Exxon Nuclear Idaho Company, Inc.

Federal regulations will limit release of krypton-85 to the environment due to reprocessing of nuclear fuel irradiated after January 1, 1983. Immobilizing krypton in a solid may offer significant safety advantages over high pressure gas cylinder storage. High pressure/temperature encapsulation of krypton in zeolite 5A and "thirsty" Vycor glass results in typical loadings of 50 STP cc/g for zeolite and 20 STP cc/g for glass. Below 825°C, krypton leakage measured by thermogravimetric analysis (TGA) is 1-3% for zeolite 5A and less than 1% for glass.

Under 1000 atm of krypton optimum encapsulation temperatures are 700°C for zeolite 5A and 900°C for glass. Initial water loadings of 6 wt % or greater causes recrystallization to an anorthite type feldspar resulting in no krypton entrapment.

The final waste form was characterized by x-ray diffraction, differential thermal analysis (DTA), infrared and transmission electron microscopic techniques.

Leakage into vacuum of krypton trapped in zeolite 5A or porous Vycor "thirsty glass" is reported for temperatures of 500-800°C. The krypton is initially forced in at 1000 atm, and trapped by sintering at 700°C (zeolite) or 900°C (glass). A small amount, <2% in a well-prepared sample, is evolved in a few hours at 300-500°C. Leakage of the remainder is very slow. Extrapolation of leakage measurements over a few days or weeks at temperatures of 600-750°C to longer times and lower temperatures yields 13% leakage in 10 years at 600°C or 1.3% at 500°C. Above 800°C the structure breaks down in a few hours, and rapid leakage continues even after the temperature is lowered.

1 SOLUBILITY EFFECTS IN WASTE GLASS/DEMINERALIZED WATER SYSTEMS. Harold T. Fullam, Battelle, Pacific Northwest Laboratory, P. O. Box 999, Richland, WA 99352.

A study was carried out to determine the solubility limits of various elements found in waste glasses in demineralized water as a function of temperature. The work was sponsored by the Office of Nuclear Waste Isolation under contract to the Department of Energy. Solubility measurements were carried out at 35°, 65°, 95° and 150°C using four nonradioactive waste glass compositions. Subsaturation and supersaturation methods were used to determine the solubility limits. The two methods gave markedly different values for most glass components. The results obtained indicate that it is impossible to assign solubility limits to most glass components without thoroughly describing the glass-water system. This includes not only defining the glass type and initial solution composition and temperature, but also the glass surface area-to-water volume (S/V) ratio of the system and its thermal history.

H12 THE IMMOBILIZATION OF CARBON-14 FORMED DURING REACTOR OPERATIONS*. Sherman Fried, Felix Schreiner, Donald Cohen, John Hines and Arnold Friedman, Chemistry Division, Argonne National Laboratory, 9700 South Cass Avenue, Argonne, Illinois 60439.

Carbon-14 is formed during reactor operations by various nuclear reactions. During fuel reprocessing, decommissioning, or other operations this radionuclide is released to the atmosphere largely as $^{14}CO_2$ and will enter the biosphere through plant metabolism, animal respiration or through the water supply. Thus carbon-14 is considered to form a definite airborne biological hazard.

Carbon dioxide has been passed over hot magnesium turnings at approximately 600°C. The reaction

$$CO_2 + 2\,Mg \longrightarrow 2\,MgO + C$$

takes place readily and elemental carbon is deposited on the magnesium. Since carbon is both very inert and very insoluble (deposits of graphite have existed unchanged for hundreds of millions of years) this reaction provides a good method of immobilizing ^{14}C in a geological repository. By disposing of the ^{14}C in this manner it can be assured that ground water intrusion cannot mobilize it and carry it into the biosphere.

*Work performed under the auspices of the United States Department of Energy under contract number W-31-109-ENG-38.

H13 LANTHANIDE ORTHOPHOSPHATES AS A MATRIX FOR SOLIDIFIED RADIOACTIVE DEFENSE AND REACTOR WASTES.* M. Petek, M. M. Abraham, and L. A. Boatner, Solid State Division, Oak Ridge National Laboratory, Oak Ridge, Tennessee 37830.

The lanthanide orthophosphates are being evaluated as a waste form for the storage of high-level defense and nuclear power reactor wastes. Pure lanthanum orthophosphate was employed as the host material for the incorporation of simulated composite Savannah River waste (SRW) without aluminum removal and simulated PW-4b reprocessed LWR wastes. In both cases the solidified waste forms were prepared from a nitric acid solution of the mixed, calcined simulated waste oxides and La_2O_3. The composite phosphate material was formed by the addition of $(NH_4)_2HPO_4$ or by combining $(NH_4)_2HPO_4$ with precipitation from molten urea. In the case of SRW the effects of waste loading (10 to 50 wt. % simulated SRW relative to La_2O_3) on the sintering properties of the composite material and on the formation of other crystalline phases were investigated. Dilatometry measurements on cold-pressed powders of the SRW material showed that sintering occurs in the range of 700 to 1000°C. Cold-pressed, sintered pellets of the SRW composite material had densities of 3.9 and 3.2 g/cm^3 for loadings of 20 and 50 wt. % SRW, respectively. In the case of $LaPO_4$ plus simulated PW-4b, the effects of urea precipitation conditions on the particle size and compaction properties were studied. Waste form pellets with 92 ± 2% of the theoretical density of pure $LaPO_4$ were formed by cold pressing and sintering (T_{max} = 1200°C) Hot pressing at 4000 psi and 1050°C produced pellets with a density of 96% of the theoretical value. Leaching results for Cs, Sr, and U will be presented.

*Research sponsored by the Division of Materials sciences, Office of Basic Energy Sciences, U.S. Department of Energy under contract W-7405-eng-26 with Union Carbide Corporation.

4 HOT-PRESSED BARIUM SULPHATE CERAMICS FOR IMMOBILISATION OF MEDIUM LEVEL MAGNOX WASTE. A. Briggs, D.V.C. Jones and G.B. Cole, Materials Development Division, AERE Harwell, OXTI ORA, U.K.

A possible method of treatment for Magnox waste is by dissolution in nitric acid and conversion to barium sulphate floc by floc precipitation. Radwaste ions are co-precipitated with the floc.

This paper describes the results of work aimed at conversion of barium sulphate floc, containing simulated radwaste, into a synthetic, ceramic version of the natural mineral barytes by a hot-pressing route. By variation of the parameters pressure, temperature and time, optimum conditions for consolidation of the floc to > 90% theoretical density on a laboratory scale are found to be 22.5 MPa, 900°C for 10 minutes. Using these conditions, hot-pressed billets of $BaSO_4$ have been made on a 5 kg scale. In going from the Magnox sludge to the hot-pressed barium sulphate a volume reduction factor ∿ 50 is achieved. The principal phases in the product are found to be $BaSO_4$, MgO and Fe_3O_4, and the degree of consolidation achieved depends on the MgO content.

The leaching behaviour of the hot-pressed materials in 100°C, 3 day Soxhlet tests also depends on the MgO content, and on the consequent level of open porosity. If there is porosity accessible to the leach water, MgO at the internal surfaces is converted to $Mg(OH)_2$, which deposits within the pores, and a weight gain is registered in the Soxhlet test. If, however, there is no open porosity, a weight loss occurs, and leach rates ∿ 4×10^{-7} kg/m^2/sec are found. In contrast, pure $BaSO_4$, hot-pressed to similar densities, shows no variation in leaching behaviour over a wide range of open porosities, and gives Soxhlet leach rates ∿ 8×10^{-7} kg/m^2/sec. Leaching tests of both pure $BaSO_4$ and floc containing simulated radwaste, spiked with ^{137}Cs and hot-pressed under conditions giving similar densities, show that ...

5 IMMOBILIZATION OF HIGH LEVEL NUCLEAR REACTOR WASTES IN SYNROC: A CURRENT APPRAISAL A.E. Ringwood, V.M. Oversby and S.E. Kesson, Research School of Earth Sciences, Australian National University, Canberra, A.C.T. 2600, Australia.

Leaching tests carried out under dynamic rather than static conditions provide the best available means of predicting the comparative performances of wasteforms buried in a geological repository. Glasses dissolve quasi-congruently whereas leach rates obtained for the titanate ceramic wasteform SYNROC fall of drastically with time. Leach rates also vary considerably according to the species monitored. In water at 95°C the long-term leach rates of Cs, Ca and U from SYNROC are respectively 500, 2500 and 100,000 less than those from borosilicate glass. Even more pronounced differences prevail at 200°C and 300°C.

The effects of radiation damage on SYNROC minerals have been investigated. Natural zirconolites and perovskites have quantitatively retained α-emitting radionuclides even after radiation doses far greater than SYNROC would receive over a million years.

New developments in processing technology, to be described, make SYNROC manufacture simpler, less hazardous and probably less costly than borosilicate glass fabrication.

H16 THE INFLUENCE OF GAMMA IRRADIATION ON THE LEACHING BEHAVIOR OF MODIFIED SYNROC-B CERAMIC WASTE FORM. Ahmad Gabr Solomah, Engineering Research Services Division, North Carolina State University, Raleigh, NC 27650.

The effects of γ-irradiation on the chemical stability and particularly, the leaching behavior of sintered Modified SYNROC-B ceramic waste form[1] loaded with 10 wt % simulated high level radioactive waste (PW-4b) have been investigated. Using a Co^{60} as a gamma irradiation source, leaching experiments have been conducted according to the Materials Characterization Center Leach Test procedures and requirements (MCC-1) at the ambient temperature and pressure (25°C and 1 atmosphere). Deionized water was the leachant medium. Similar leaching experiments have been carried out in the absence of γ-irradiation and the leach rates of U, Ce and Fe are presented for comparison studies. Acid/Base formation has been monitored through pH measurements of the solution. Figure 1 shows the effects of γ-irradiation on the leaching behavior of sintered Modified SYNROC-B ceramic waste form by indicating the leach rates of Fe^{+2} and Ce^{+3} as a function of the leaching time (i.e., the accumulated radiation dose). Data from leaching experiments for longer periods (up to 6 months) are available for other high level radioactive waste constituents (e.g., U).

[1] A.G. Solomah, T.M. Hare and H. Palmour, III, Trans. Am. Nuc. Soc. 34, 197 (1980).

H17 MIGRATION OF RADIONUCLIDE CHAINS IN SUBSEABED DISPOSAL; A. K. Ray; Department of Chemical Engineering; University of Kentucky; Lexington, KY 40506, and H. E. Nuttall; Department of Chemical and Nuclear Engineering; University of New Mexico; Albuquerque, NM 87131.

The two dimensional (r-z) migration of radionuclides released from a canister located in a sedimentary layer bounded at the top by the ocean and at the bottom by an impermeable basalt zone is analyzed to determine the transport rates of radionuclides to the ocean floor. The radionuclide in the canister is assumed to decay in the following chain form

$$A_1 \xrightarrow{\lambda 1} A_2 \xrightarrow{\lambda 2} A_3 \xrightarrow{\lambda 3}$$

Analytical solutions were obtained for the concentration distributions, for the flux at the sediment/ocean interface and for the discharge rate to the ocean floor for each chain member. Using the properties of chain members present in actinide decay systems, the effects of canister placement depth, half-life, adsorption equilibrium and other relevant parameters are elucidated.

H18 STANDARD TESTS FOR THERMAL STABILITY OF NUCLEAR WASTE FORMS*, W. J. Gray, R. P. May and W. J. Weber. Pacific Northwest Laboratory, Richland, WA 99352.

Three standard tests for evaluating the thermal stability of nuclear waste forms have been proposed by the Materials Characterization Center (MCC). The effects of thermally-activated phase transformations on the properties of nuclear waste forms are measured by the MCC-7 Thermal Phase Stability Test. The MCC-8 High Temperature Vaporization Test measures the vaporization of fission products and other condensable elements from nuclear waste forms at high temperatures. The potential pressurization of a canister at elevated temperatures is determined by the MCC-9 Thermal Gas Generation Test which measures pressure of noncondensable gases from various waste forms as a function of temperature. The three tests will be described in detail and results for several waste forms will be presented.

* This work was performed for the Department of Energy under Contract DE-AC06-76RLO 1830.