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An Annotated Bibliography for the Design of Waste Packages for Geologic Disposal of Spent Fuel and High-Level Waste

Technical Report

November 1982

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ONWI
Office of Nuclear Waste Isolation

BATTELLE Project Management Division

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ABSTRACT

This bibliography identifies documents that are pertinent to the design of waste packages for geologic disposal of nuclear waste. The bibliography is divided into fourteen subject categories so that anyone wishing to review the subject of leaching, for example, can turn to the leaching section and review the abstracts of reports which are concerned primarily with leaching. Abstracts are also cross referenced according to secondary subject matter so that one can get a complete list of abstracts for any of the fourteen subject categories.

All documents which by their title alone appear to deal with the design of waste packages for the geologic disposal of spent fuel or high-level waste were obtained and reviewed. Only those documents which truly appeared to be of interest to a waste package designer were abstracted. The documents not abstracted are listed in a separate section. There was no beginning date for consideration of a document for review. About 1100 documents were reviewed and about 450 documents were abstracted.

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ABBREVIATIONS AND ACRYONYMS

ANL	- Argonne National Laboratory
ASEA	- Allmaenna Svenska Electriska Aktiobolaget
BCL	- Battelle Columbus Laboratories
BNWL	- Battelle Pacific Northwest Laboratories
BWR	- Boiling-Water Reactor
CRNL	- Chalk River Nuclear Laboratories
DOE	- U.S. Department of Energy
EMAD	- Engine Maintenance Assembly and Disassembly Area
EPA	- U.S. Environmental Protection Agency
ft	- foot
FUETAP	- Formed Under Elevated Temperatures and Pressure
FY	- Fiscal Year
Gal	- Gallons
HLW	- High-Level Waste
hr	- hour
HTGR	- High-Temperature Gas-Cooled Reactor
IAEA	- International Atomic Energy Agency
ICPP	- Idaho Chemical Processing Plant
in	- inches
l	- liter
LLL	- Lawrence Livermore Laboratory
KBS	- Swedish Nuclear Safety Project
kW	- kilowatt
LWR	- Light-Water Reactor
MPa	- Megapascal

MTHM	- Metric Tons of Heavy Metal
MTU	- Metric Tons of Uranium
MW	- Megawatts
MWD	- Megawatt Days
NAS	- National Academy of Sciences
NWTS	- National Waste Thermal Storage (program)
ONWI	- Office of Nuclear Waste Isolation
ORNL	- Oak Ridge National Laboratory
OWI	- Office of Waste Isolation
PNL	- Battelle Pacific Northwest Laboratories
PWR	- Pressurized-Water Reactor
SRP	- Savannah River Plant
TRU	- Transuranic
WIPP	- Waste Isolation Pilot Plant
WISAP	- Waste Isolation Safety Assessment Program
WSEP	- Waste Solidification Engineering Prototypes (program)
yr	- year

I. INTRODUCTION

Purpose and Objectives

This bibliography identifies documents that are pertinent to the design of waste packages for geologic disposal of spent fuel or high-level waste. The waste package has been defined by the Office of Nuclear Waste Isolation as the waste form, and the enclosing envelope that separates the waste form from the host rock. The package components may include the waste form, stabilizer, canister, overpack, sleeve, and backfill materials.

There are a multitude of reports, journal articles and other types of written information which deal with packaging of spent fuel and high-level waste for deep geologic disposal. The intent of this bibliography is to provide a literature search for the package designer and to summarize the identified information that is pertinent to the current ONWI waste package design effort.

The citations for the reports identified in the literature search as containing information on the disposal of spent fuel or high-level waste in deep geologic formations are listed in Section 2, the Citations Section. The citations for the documents that contain pertinent information to the waste package designer are listed with an abstract number. The abstracts for these documents are in Section 3, the Abstracts Section, and are grouped according to the primary subject of the document. The subject categories represent various characteristics, effects, and requirements of the waste package. At the end of each subject category is a list of abstract numbers for the documents which contain information on that subject of a secondary nature. The resulting cross referencing of documents provides the packager access to a list of documents and abstracts for major subject areas of interest. The exception to this cross referencing method is the documents in the criteria subject category. The criteria documents deal with every subject category and are therefore not cross referenced.

Literature Search Sources and Dates

The literature search was a combination of computerized and manual searches. There was no beginning date for consideration of a document for

review. The ending date is the date of publication of this bibliography. Every effort was made to include the most recent reports.

The complete Nuclear Science Abstracts and the Energy Research Abstracts through Volume 5, Number 8, were searched by computer. Also searched by computer was the ORLOOK data base located at Oak Ridge National Laboratory. The ORLOOK data base contains report citations directed towards radioactive waste management. The manual literature search included the Office of Nuclear Waste Isolation's library. Also searched was the reference list of each reviewed document.

Document Review Guidelines

Each document identified in the literature search required a decision concerning whether or not the document should be acquired and reviewed. The decision was based solely on the title of each document. If by the title alone the document appeared to contain information concerning the disposal of spent fuel or high-level waste in deep geologic repositories, the document was obtained, reviewed, and listed in the Citations section.

Various guidelines were established to aid in this decision. If by the title, the document appeared to deal strictly with chemical processing of waste the document was not reviewed. If the document appeared to include discussion of waste form properties obtained by a particular processing method, the document was reviewed. Documents which appeared to deal only with temperature or radiation effects on the host rock were not reviewed because the effects were not within the confines of the package. Upon review, only those documents which contained information relevant to the needs of a waste package designer were abstracted and listed in both the Citations section and the Abstracts section. All reviewed documents are listed in the Citations section.

Approximately 1200 documents were identified in the computerized literature search. About 1100 of the documents identified in the computerized and manual searches were reviewed. About 450 of the reviewed documents were abstracted.

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Winchester, J.W., "Long-Term Geochemical Interactions of High Active Waste with Crystalline Rock Repository Media", Florida State University, Scientific Basis for Nuclear Waste Management", Vol. 1, McCarthy, G.J., (ed), Plenum Press, New York, (1979), pp 345-347. No Abstract

Witherspoon, P.A., "Cooperative Work Program Between ERDA/OWI and the Swedish State Power Board on Waste Storage in Mined Caverns", Lawrence Berkeley Laboratory, National Waste Terminal Storage Program Conference on Waste-Rock Interaction, July 6-7, 1977, Y/OWI/SUB-77/14268, pp 81-82. No Abstract

Witherspoon, P.A., et al., "Geologic Storage of Radioactive Waste Results from Field Investigations at Stripa, Sweden", Lawrence Berkeley Laboratory, Waste Management '80, Post, R.G., (ed), University of Arizona, Tucson, (1980), pp 463-502. No Abstract

Wodrich, D.D., "Retrievable Surface Storage of High-Level Radioactive Waste", Atlantic Richland Hanford Co., American Nuclear Society Winter Meeting, San Francisco, California, CONF-731101, (November 1973), p 326. No Abstract

Wood, B.J., "Estimation of Waste Package Performance Requirements for a Nuclear Waste Repository in Basalt", Rockwell Hanford Operations, National Waste Terminal Storage Program Information Meeting, Columbus, Ohio, December 8-11, 1980, ONWI-212, (December 1980), pp 335-338. 12.166

Wood, B.J., "Backfill Performance Requirements, Estimates from Transport Models", Rockwell Hanford Operations, RHO-BWI-SA-58, (November 1980). 22.015

Wranglen, G., "Gold and Selective Storage Solves Nuclear Waste Problem", Royal Institute of Technology, Annals of Nuclear Energy, Vol. 4, (1977), pp 527-528. No Abstract

Wurm, K.J., "Conceptual Design of Packaging For Disposal of Spent Pressurized Water Reactor Fuel Assemblies in a Salt Formation Repository", Ohio State University, Unpublished Thesis, (1978). 20.006

Wurm, K.J., and O'Hara, F.A., "Spent-Fuel Processing and Packaging Options", Transactions of the American Nuclear Society, Vol. 32, (1979), pp 382-383.
13.017

Yen-Bower, E.L., et al., "An Approach Towards Long Term Prediction of Stability of Nuclear Waste Forms", University of Florida, Ceramics in Nuclear Waste Management, Chikalla, T.D., and Mendel, J.E., (eds), Cincinnati, Ohio, April 30 - May 2, 1979, CONF-790420, (1979), pp 41-46. 12.167

Zemlyanukhin, V.I., et al., "Clay Phosphate Ceramics and Vitromets: Alternatives to Monolithic High Level Waste Glass Products", V.G. Khlopin Radium Institute, USSR. Scientific Basis for Nuclear Waste Management, Vol. 1, McCarthy, G.J., (ed), Plenum Press, New York, (1979), pp 195-200.
12.168

Zerwekh, A., et al., "Experimental Studies of the Degradation of RAD-Wastes for the Sandia Laboratories Waste Isolation Pilot Project (WIPP) April 1-June 30, 1978", Los Alamos Scientific Laboratory, LA-7478-PR, (October 1978), 5 p.
No Abstract

Zima, G.E., "An Evaluation of Potential Chemical/Mechanical Degradation Processes Affecting Fuel and Structural Materials Under Long-Term Water Storage", Battelle Pacific Northwest Laboratories, (PNL-2379), NUREG/CR-0668, (May 1979). No Abstract

10. Criteria

10.001

Christensen, R.N., et al., "Waste Characteristics That May Require Specification in Waste Acceptance Criteria", Ohio State University, Transactions of the American Nuclear Society, Vol. 32, (1979), pp 383-385.

This paper lists those waste characteristics that may require criteria. The waste characteristics are divided into four general categories: nuclear, chemical, physical, and thermal. Also included is an extensive matrix (33 x 33) showing the interrelationships of these characteristics in terms of independent and dependent variables.

10.002

Davis, J.M., and Smith, R.E., "Standardized Experimental Package for Spent Fuel", Rockwell Hanford Operations, RH0-SA-141, (October 1979).

The conceptual design criteria of the Standardized Experimental Package for Spent Fuel, plus some results of the preliminary testing is described in this paper. The canister will be used to emplace spent fuel assemblies in geologic-repository demonstration sites. The packaged fuel will be received from the EMAD packaging facility located at the Nevada Test Site in presently licensed truck-mounted shipping casks.

Conformance to the ONWI Criteria. Many of the criteria are similar to the ONWI criteria. Deviation of the criteria comes from its use of only a 10-year minimum life for container integrity, as opposed to 1000 years, as in the ONWI criteria.

CRITERIA

III-2

10.003

Heckman, R.A., and Holdsworth, T., "Determination of Performance Criteria for High-Level Solidified Nuclear Waste", Lawrence Livermore Laboratory, UCID-18121, (May 1979), 281 p.

Lawrence Livermore Laboratory provides broad technical support to the NRC for developing performance criteria for HLW. This document is the briefing document on performance criteria.

Conformance to ONWI Criteria. These criteria pertain more to the handling and transportation of radioactive wastes than to specific waste properties.

10.004

LaRiviere, J.R., et al., "Preliminary Canister Size Criteria for Commercial Solidified High-Level Waste", Atlantic Richfield Hanford Co., ARH-ST-151, (April 1977), 23 p.

Criteria for the shape, size, and weight of HLW disposal packages are discussed. Emphasis is placed on designing the package to be compatible with present reprocessing and transportation facilities, as well as proposed repository sites. HLW canister acceptance criteria, as well as waste solidification methods are presented. Additional waste canister acceptance criteria are suggested.

Conformance to the ONWI Criteria. Most of the performance criteria in this study conform to the ONWI criteria.

III-3

10.005

LaRiviere, D.C., et al., "Development of National Waste Terminal Storage Program Waste Packaging Criteria", Atlantic Richfield Hanford Co., Transactions of the American Nuclear Society, Vol. 26, (1977), pp 247-248.

The waste acceptance criteria program managed by OWI in the mid-1970's is briefly described. The canister produced for use at WIPP is described, as is the thermal power of the waste to be disposed.

Conformance to ONWI Criteria. As this document is a brief abstract, few specific criteria are presented.

10.006

Larson, D.E., "Interim Package Performance Criteria for Spent Fuel Disposal in a Geologic Repository", Battelle Pacific Northwest Laboratories, SFPD-T1-R-79-1, (November 1978).

This report presents the state of packaging criteria for PWR and BWR spent fuel disposal in a geologic repository as of November 1978. The criteria relate to the form of the spent fuel assemblies, the design of the canister, and the compatibility of the package with the storage and disposal environments.

Conformance to the ONWI Criteria. Most, but not all, of the proposed criteria are in conformance to the ONWI packaging criteria.

10.007

Mendel, J.E., "High-Level Waste Glass", Battelle Pacific Northwest Laboratories, Nuclear Technology, Vol. 32, No. 1, (January 1977), pp 72-87.

CRITERIA

III-4

This paper discusses the HLW disposal techniques which are being investigated at PNL. The criteria discussed are those for the material properties of the canister, and of the glass, particularly borosilicate glass. The spray calciner and in-can melting processes are discussed.

Conformance to the ONWI Criteria. The criteria of the HLW glass discussed in this paper conform fairly well with the ONWI criteria.

10.008

Moore, E.L., et al., "Radioactive Waste Package Criteria: Spent Fuel Elements, High-Level Waste, Cladding Hull Waste, Intermediate-Level TRU Waste, Low-Level TRU Waste", Rockwell Hanford Operations, RHO-CD-568, (October 1978).

This document discusses the acceptance criteria for containers of spent fuel, HLW, cladding waste, and TRU waste which must be stored or disposed of by the DOE. These acceptance criteria are intended to ensure that the waste containers are compatible with all elements of the waste management system, and provide a base for handling, transporting and disposing of commercial waste.

Conformance to the ONWI Criteria. Most of these criteria conform to the ONWI criteria; however, many of the quantitative values were arbitrarily established and hence subject to change.

10.009

Moore, E.L., "Interim Report: Commercial Waste Packaging Studies Task 2, High-Level Waste Package Acceptance Criteria Study", Rockwell Hanford Operations, RHO-ST-10, (March 1978), 47 p.

III-5

A set of preliminary HLW package acceptance criteria have been developed. Areas requiring additional study are identified.

Conformance to the ONWI Criteria. The criteria developed here are an early attempt to establish and identify the eventual criteria required for the waste package. Some but not all of the criteria are in conformance with the ONWI criteria.

10.010

NUS Corp., "Proposed Annotated Outline for Safety Analysis Report for Radioactive Waste Repositories in Rock Salt Formations", Y/OWI/TM-56, (1978).

This report deals principally with the geologic, hydrologic, meteorologic, and demographic aspects of waste disposal in a salt formation. However it also discusses this disposal concept with respect to its conformance with general NWTs design criteria, including protection against fires, explosions, and nuclear criticality, and the retrievability of the waste.

10.011

O'Hara, F.A., et al., "Development of Comprehensive Waste Acceptance Criteria for Commercial Nuclear Waste", Battelle Columbus Laboratories, Ceramics in Nuclear Waste Management, Chikalla, T.D., and Mendel, J.E., (eds), Cincinnati, Ohio, April 30 - May 2, 1979, CONF-790420, (1979), pp 36-40.

A detailed methodology is presented for the identification of the characteristics of commercial nuclear waste which may require criteria. This methodology is analyzed as a six-step process which begins with identification of waste operations and proceeds until the waste characteristics affecting the potential release of radionuclides are determined. All waste types and opera-

tions were analyzed using the methodology presented. Several illustrative examples are included. It is found that thirty-three characteristics can be identified as possibly requiring criteria.

Conformance to the ONWI Criteria. The paper presents a methodology for identifying characteristics which may require criteria but not the criteria themselves.

10.012

O'Hara, F.A., et al., "Preliminary Report, Waste Acceptance Criteria Study, Phase 1--Identification of Characteristics", Battelle Columbus Laboratories, ONWI-6(1), (September 1978).

This report is an early attempt to identify the characteristics of the waste package which may require criteria. The interrelationships between characteristics are developed and the concept of the "most restrictive limit" for eventual quantification of criteria is discussed.

10.013

Rockwell International Corp., "Commercial Waste and Spent Fuel Packaging Program; Package Acceptance Criteria, Remote Handled Waste, Contact Handled Waste", Rockwell Hanford Operations, RHO-CD-711, (June 1979).

These criteria were prepared in support of the ONWI program to develop criteria for radioactive waste disposal. These are second level criteria, which are responsive to the Waste Management System Performance Criteria, developed by ONWI, which are the primary level criteria. These acceptance criteria are qualitative statements defining package characteristics which are acceptable for receipt and emplacement in a geologic repository.

III-7

Conformance to the ONWI Criteria. These criteria are a generalized and earlier version of the ideas in the ONWI criteria.

10.014

Rockwell International Corp., "Package Design Criteria for Commercial Waste and Spent Fuel Packaging Program", Rockwell Hanford Operations, RHO-CD-772, (August 1979).

The design criteria presented in this document are a preliminary set of package design guidelines which will be clarified and modified as more detailed policy decisions and technical information become available. These criteria discuss the various chemical, physical and nuclear characteristics of the waste form and the canister.

Conformance to the ONWI Criteria. These criteria are similar to the ONWI criteria; however, these criteria predate the more definitive ONWI criteria.

10.015

Science Applications, Inc., "An Approach to Equipment Criteria and Specification Development with Canister Evaluation as an Example", (SAI-OR-629-1), ONWI/SUB/78-E513-0042-1, (November 1978), 55 p.

This report discusses a method which could be used to generate design specifications for equipment and hardware from preliminary plant designs and general plant or facility requirements and criteria. The method is called the Functional Flow Method.

10.016

Steffan, R.J., et al., "Equipment Designs for the Spent LWR Fuel Dry Storage Demonstration", Westinghouse Advanced Energy Systems Division, Transactions of the American Nuclear Society, Vol. 34, (1980), pp 835-837.

The design and fabrication of components and equipment for receipt, inspection, lag storage, encapsulation, testing, and emplacement of spent fuel assemblies at the EMAD facility at the Nevada Test Site are discussed but details are not given. Specific components considered are: fuel canisters and closure lids, remote handling tools, a shielded transfer cask, drywell liners, and shield plugs. Some of the design parameters which were established for the canister include: material (type 304 stainless steel), design pressure (17.0 kg/cm^2), design temperature (340°C), closure method (threaded with seal weld), backfill medium (helium), and maximum allowable leak rate ($10^{-5} \text{ std cm}^3/\text{sec}$). A cutaway view of the drywell/canister is included in the paper. Also, results of actual encapsulation of spent fuel assemblies are listed.

Conformance to the ONWI Criteria. The conformance of the few design parameters presented in this paper to the ONWI packaging criteria cannot be determined since the specific design limits related to the ONWI packaging criteria have not been established yet. Also, the container designed in this study is not intended for permanent disposal in a geologic repository and therefore is not constrained by any ONWI packaging criteria.

10.017

Union Carbide Corp., "Conceptual Design Criteria for Facilities for Geologic Disposal of Radioactive Wastes in Salt Formations", Y/OWI/TM-9, (1977).

Conceptual design criteria for a facility for the long-term storage and disposal of commercial radioactive wastes are presented. Canister and

III-9

waste package materials for spent fuel and HLW and proposed dimensions are discussed. A maximum total weight of the canister, overpack, and contents is specified. Handling attachments are briefly discussed. The maximum allowable thermal power, neutron dose rate, and gamma dose rate for a single canister are also specified.

Conformance to the ONWI Criteria. These criteria are more specific than the ONWI criteria.

10.018

U.S. Department of Energy, "Strategy Document, Long-Term High-Level Waste Technology Program", Savannah River Operations Office, DOE/SR-WM-79-3, (December 1979).

This document is a comprehensive overview of the DOE program to isolate all of the United States nuclear wastes from the human environment. Discussed are the various programs which are under way in the United States. The various programs involving the development and characterization of the waste form are categorized by the institutions and the waste form under development. The Waste Materials Characterization Center has been established at PNL to provide quality assurance and provide system-wide concurrence. Some waste acceptance criteria are presented.

Conformance to the ONWI Criteria. The views in this report conform fairly well with the intent of the ONWI criteria.

see also

12.064

12.166

14.019

16.027

18.002

18.011

III-10

11. Corrosion11.001

Abrego, L., and Braithwaite, J.W., "Corrosion of Titanium in Saline Nuclear Waste Isolation Environments", Sandia Laboratories, Transactions of American Nuclear Society, Vol. 34, (June 1980), p 196.

Susceptibility of titanium alloys Ti-50A and Ti Code-12 to crevice corrosion and stress corrosion cracking in both nonirradiated and gamma irradiated saturated brine and seawater environments is investigated. The effect of crevice gap width (from 0.01 mm to 0.25 mm) is also being investigated in oxygenated and deoxygenated, 250°C solutions for both Ti Code-12 and Ti-50A.

11.002

Berreth, J.R., et al., "Stabilization and Storage of Solidified High-Level Radioactive Wastes", Allied Chemical Corp., Nuclear Technology, Vol. 32, (1977), pp 16-24.

Corrosion measurements on stainless steel bins used to store high-level waste calcines at the Idaho Chemical Processing Plant indicate a corrosion rate of 0.13 mm over 500 yr. Stabilization of commercial-type calcine is done prior to storage in sealed canisters. The impact of waste form and heat generation rate on practical canister configuration and on permissible waste decay time at initial storage is discussed.

11.003

Braithwaite, J.W., and Molecke, M.A., "High-Level Waste Canister Corrosion Studies Pertinent to Geologic Isolation", Sandia Laboratories, High-Level

Radioactive Solid Waste Forms, Casey, L.A., (ed), Denver, Colorado, December 19-21, 1978, (CONF-781241), NUREG/CP-0005, (April 1979), pp 241-281.

Media considered for waste isolation include bedded and domed salt, sub-seabed sediment, and different kinds of hard rock. Corrosion effects were examined as a function of temperature (25 to 250°C), pressure, solution composition, oxygen concentrations, and moisture content. Corrosion rates of steel and other candidate alloys are given. Of 20 candidate metals, eight were selected for further study. They are: 1018 mild steel, Corten A steel, lead, 90-10 Cupronickel, SS-Ebrite 26-1, Nomen 400, Inconel 600, and Ti Code-12.

11.004

Braithwaite, J.W., and Molecke, M.A., "High Level Waste Canister Corrosion Studies Pertinent to Geologic Isolation", Sandia Laboratories, SAND-78-2111, (December 1978).

This report presents results of several series of laboratory tests which evaluated the effects of environmental parameters on corrosion rates for canister materials proposed for use in geological repositories. The cost, mechanical properties and estimated lifetimes of various canister and overpack metals are compared.

11.005

Braithwaite, J.W., and Magnani, N.J., "Corrosion Considerations for Nuclear Waste Isolation Canisters", Scientific Basis for Nuclear Waste Management, Vol. 1, McCarthy G.J., (ed), Plenum Press, New York, (1979), pp 283-287.

The eventual selection of a HLW canister material will be based upon compatibility, strength, material cost and availability, and canister

III-13

fabricability. The corrosion resistance is indicated to be perhaps the most important criteria in canister material selection. Corrosion rates for ten different candidate overpack alloys in three different seawater and brine systems are presented.

11.006

Braithwaite, J.W., and Molecke, M.A., "Nuclear Waste Canister Corrosion Studies Pertinent to Geologic Isolation", Sandia Laboratories, Nuclear and Chemical Waste Management, Vol. 1, No. 1, (1980), pp 37-50.

The effect of environmental (temperature, pressure, solution chemistry, radiolysis products, and water intrusion) and processing (stress, and sensitization) parameters on canister corrosion are discussed. References are given to pertinent work in each area. The applicable corrosion studies from Project Salt Vault, several geothermal energy extraction programs, and light water nuclear reactor cooling cycles are referenced along with conclusions regarding aluminum alloys, mild low strength steels, stainless steels, nickel based alloys and titanium alloys. Corrosion rates are given for twenty-two candidate alloys in deoxygenated solutions. As a result of the present work, five alloys plus lead are chosen for further investigation as possible overpack materials; Ti-Code-12, Hastelloy C-276, and 90-10 Cupronickel, Inconel 625, and Incoloy 825.

11.007

Casteels, F., et al., "Corrosion of Materials in Clay Environment", S.C.K./C.E.N., Scientific Basis for Nuclear Waste Management, Vol. 2, Northrup, C.J.M., Jr., (ed), Plenum Press, New York, (1980), pp 385-393.

The corrosion resistance of candidate canister materials for the storage of radioactive waste has been tested in the presence of clay.

Corrosion tests were carried out at 25, 50, 100, 150, 200 and 300°C on potential canister materials in a wet atmosphere simulating pyrolysis of clay at 300°C. Carbon steel, AISI 304, AISI 304L, AISI 316, AISI 316L, UHB 904L, AISI 430, 2 ferritic steels with low interstitial content, Inconel 600, Inconel 625, Incoloy 800, nickel 200, Hastelloy B, Hastelloy C, aluminum alloys, titanium, titanium alloyed with lead, titanium alloyed with aluminum and tin or aluminum and vanadium were evaluated in tests lasting up to 16 months.

11.008

Gies, H., "Experiments on the Corrosion of Materials For Borehole Casings and for Waste Packaging Canister in Salt", Institute fuer Tieflagerung, Waste Isolation Performance Assessment and In-Situ Testing, US/FRG Bilateral Workshop, Berlin, October 1-5, 1979, ONWI-88, (December 1979), pp 405-416.

For the test storage initially planned for HLW-canisters at the Asse salt mine, provisions for safe recovery must be provided. For this purpose the borehole must be cased. The catalog of requirements to be met by the materials for the canisters and for the borehole casings is discussed. Selected types of material for container material and borehole casings are given.

11.009

Henriksson, S., et al., "Measurement of Galvanic Corrosion Between Titanium and Lead and Measurement of Corrosion Potential of Titanium Under Gamma Radiation", KBS Technical Reports 1-120, Report No. 67, (1977-1978).

The corrosivity of gamma-irradiated simulated groundwaters for titanium is discussed. The galvanic corrosion of lead in contact with

titanium is reported for water-saturated air at 80°C. All measurements were conducted in either the presence or absence of a bentonite/quartz buffer material.

11.010

Henriksson, S., and Pourbaix, M., "Corrosion Testing of Unalloyed Titanium in Simulated Deposited Environments for Reprocessed Nuclear Fuel Waste", KBS Technical Reports 1-120, Report No. 96, (1977-1978).

Three hundred day corrosion tests were carried out on unalloyed titanium, which is planned to be used as an outer corrosion resistant canister for HLW. The corrosive medium was modified Baltic water at 100 and 130°C with various oxygen contents.

11.011

Hunter, T.O., et al., "Canister Material Studies for Waste Isolation at Sandia Laboratories", Sandia Laboratories, Waste Isolation Performance Assessment and In-Situ Testing, US/FRG Bilateral Workshop, Berlin, October 1-5, 1979, Berlin, October 1-5, 1979, ONWI-88, (December 1979), pp 417-418.

The report reviews the effects of environmental parameters on corrosion rates canister materials. The variables tested include: temperature (25 to 250°C), pressure, time, radiation, solution composition, oxygen concentrations, welding and crevice effects, and dry and moist salt effects.

11.012

Hyden, L., "Studies of Ceramic Material for Encapsulation of High-Level Waste", KBS Technical Reports 1-120, Report No. 103, (1977-1978).

The stability and corrodability of major ceramic-forming minerals are reviewed. Minerals include diopside, feldspars, and cordierite.

11.013

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

This document deals with the effects of gaseous radioactive nuclides from spent fuel on the design and operation of a geologic repository in salt. The sources of gas in a spent fuel assembly are identified. Corrosion of the canister is one method for release of the gases in the canister. An equation which gives the rate of inleakage through capillary breaches in partially excavated canisters and another equation which gives the leakage from pressurized canisters are presented. The corrosion of carbon steel in salt is discussed and backfill materials which can protect the canister from brine are identified. The corrosivity of Zircaloy and stainless steel cladding in brine is discussed.

11.014

LaRiviere, J.R. and Moore, E.L., "Conceptual Design Considerations for the Storage of Solidified High-Level Waste in Canisters at a Commercial Fuel Reprocessing Plant", Atlantic Richfield Hanford Co., ARH-SA-221, (November 1975), 19 p.

Onsite storage of canisters of solidified HLW generated by the commercial fuel reprocessing plants may be required prior to shipping these containers to a federal repository. The most likely storage concept is to hold the waste-filled canisters in water storage basins. This report presents

III-17

conceptual designs for the canisters and water storage basins. Design considerations discussed include: type of corrosion that may develop, conditions that promote stress corrosion cracking, prevention of stress corrosion cracking, routine water basin cleanup, cleanup of a grossly contaminated water basin, effluent discharge discussions, storage basin integrity, and designing for decommissioning.

11.015

Larker, H.T., "Densification of Calcines and Direct Containment of Spent Nuclear Fuel in Ceramics by Hot Isostatic Pressing", ASEA, Sweden, Ceramics in Nuclear Waste Management, Chikalla, T.D., and Mendel, J.E., (eds), Cincinnati, Ohio, April 30 - May 2, 1979, CONF-790420, (1979), pp 169-173.

The corrosion rate of alumina container materials formed by hot isostatic pressing was investigated and results from the tests on the densified materials are given. The problem of obtaining relevant long-term leach rates for crystalline materials with mechanically disturbed surface layers is discussed.

11.016

Magnani, N.J., and Braithwaite, J.W., "Corrosion-Resistant Metallic Canisters for Nuclear Waste Isolation", Sandia Laboratories, Scientific Basis for Nuclear Waste Management, Vol. 2, Northrup, C.J.M., Jr., (ed), Plenum Press, New York, (1980), pp 377-384.

This paper describes a program currently under way at Sandia Laboratories involving the estimation of the potential of a material to survive 300 years as a nuclear waste canister. The principal properties of candidate materials which were studied are corrosion and stress corrosion cracking. Stainless steel Type 304 is compared with the leading candidate, Ti Code-12.

Four barrier lifetimes are identified, those being: (1) 0 Year Lifetime; (2) ~25 Year Lifetime; (3) ~300 Year Lifetime; (4) $>10^4$ Year Lifetime.

11.017

Mattsson, E., "Corrosion Resistance of Canisters for Final Disposal of Spent Nuclear Fuel", Swedish Corrosion Institute, Scientific Basis for Nuclear Waste Management, Vol. 1, McCarthy, G.J., (ed), Plenum Press, New York, (1979), pp 271-281.

A group of Swedish scientists have evaluated the corrosion resistance of three potential canister materials. These were titanium lined with lead, copper, and alpha-alumina. Corrosion rates have been measured as a function of temperature, chloride content, and pH.

11.018

Molecke, M.A., and Abrego, L., "Status of Sandia HLW Canister/Overpack Program Studies", Sandia Laboratories, National Waste Terminal Storage Program Information Meeting, Columbus, Ohio, 9-11 December, 1980, ONWI-212, (December 1980), pp 230-233.

Candidate alloys for container materials are examined for corrosion behavior under salt repository conditions. The effect of temperatures from 70 to 250°C on the uniform corrosion rates of the titanium alloys Ti-50A, Ti Code-12, and titanium-lead are given. The corrosion rates of 21 alloys in seawater and three kinds of brine at 250°C have been determined. The effect of dissolved oxygen on the uniform corrosion rate of titanium alloys at 250°C is also given.

11.019

Neilson, R.M., et al., "The Corrosion of Mild Steel Waste Form Containers", Brookhaven National Laboratory, Transactions of the American Nuclear Society, Vol. 32, (1979), pp 395-397.

This paper considers the interior wall corrosion of zinc-lined mild steel containers (55 gal drums) resulting from contact with urea-formaldehyde and Portland cement waste forms. Results are reported in terms of quantitative weight loss measurements ($\text{mg}/\text{cm}^2/\text{day}$) using mild steel specimens.

11.020

Posey, F.A., et al., "Corrosivity of Geothermal Brines Progress Report for Period Ending September 1977, Final Report", Oak Ridge National Laboratory, ORNL/TM-6308 (March 1978).

This final report summarizes the results of studies on the corrosion of ferrous materials in synthetic geothermal brines. A literature review on the electrochemical aspects of the corrosion of iron and carbon steel in chloride solutions is included.

11.021

Scott, T.E., "Issues Relevant to Nuclear Waste Canister Materials", Ames Laboratory, Corrosion/80 NACE Meeting, Chicago, Illinois, March 6-10, 1980, CONF-800305-7.

This paper focuses on the canister as a component of multibarrier systems for the containment and isolation of HLW and spent fuel. It provides perspective for the role of canisters in the overall nuclear fuel cycle with special emphasis on the multibarrier concept of waste isolation. Attention is

given to the chemical composition of wastes, waste forms, interim storage environments, and geologic isolation repository environments; these will all have to be considered in canister materials section. In addition, other factors such as mechanical loads, radiation effects, and radioactive decay-heat generation rates are cited.

11.022

Shannon, D.W., "Corrosion of Iron-Base Alloys Versus Alternate Materials in Geothermal Brines", Battelle Pacific Northwest Laboratories, PNL-2456, (November 1977), 25 p.

A series of 30 autoclave tests and one field test were completed to define how various chemical components in geothermal brines affect uniform corrosion of 35 materials. The times of exposure were 135 and 270 hrs. The average corrosion rates of 10 carbon steels in 1 percent NaCl and CO₂-bicarbonate brines are plotted. Tests were conducted as a function of pH, temperature, pressure and oxygen concentration.

11.023

Sundquist, J.D., "Preliminary Corrosion Test of a Glass-Ceramic Candidate for a Nuclear Waste Canister", Corning Glass Works, Scientific Basis for Nuclear Waste Management, Vol. 1, McCarthy, G.J., (ed), Plenum Press, New York, (1979), pp 289-296.

Preliminary corrosion studies of Corning Glass Works glass ceramic Code 9617 was undertaken. Corrosion rates were measured as a function of time, temperature, solution composition, and test solution volume/sample surface area ratio.

11.024

Swedish Corrosion Institute and Its Reference Group, "Evaluation of Corrosion Resistance of Material Intended for Encapsulation of Nuclear Fuel Waste Status Report", KBS Technical Reports 1-120, Report No. 31, (1977-1978).

The corrosion characteristics of three different canister materials under the geochemical conditions expected in a repository in crystalline rocks in Sweden are presented. The three different canister materials include: titanium lined with lead, copper, and alumina (Al_2O_3).

11.025

Swedish Corrosion Institute and Its Reference Group, "Lead-Lined Titanium Canister for Reprocessed and Vitrified Nuclear Fuel Waste-Evaluation from the Viewpoint of Corrosion", KBS Technical Reports 1-120, Report No. 107, (1977-1978).

The estimated life of a 6 mm thick titanium wall around a 100 mm thick lead canister is conservatively estimated. The probability of fracturing in the 100 mm thick lead wall is also assessed.

11.026

Vesterlund, G., and Olsson, T., "Degradation Mechanisms in Connection with Pool Storage and Handling of Spent Nuclear Reactor Fuel", KBS Technical Reports 1-120, Report No. 68, (1977-1978).

The extent of corrosion and hydrogen embrittlement on spent fuel and fuel rods in underwater storage for up to 50 yrs is assessed.

see also

12.007	12.021	12.032
12.033	12.042	12.045
12.083	12.089	12.094
12.101	12.107	12.111
12.114	12.115	12.123
12.137	12.140	12.144
12.145	13.003	13.010
13.016	13.017	14.001
14.013	14.019	14.032
14.038	15.001	15.007
15.013	16.035	16.046
16.050	16.059	16.066
16.073	18.009	18.011
19.006	20.001	20.006
21.001	21.007	21.018
22.001	22.002	22.005

12. Leaching

12.001

Aaron, W.S., et al., "Cermet High Level Waste Forms", Oak Ridge National Laboratory, ORNL-TM-6404, (June 1978), 19 p.

Samples of SRP simulated waste, hot-pressed cermet pellets, and extruded-sintered rods were boiled in saturated NaCl solutions for 168 hours at atmospheric pressure. Subsequent leach data are given.

12.002

Adams, P.B., "Relative Leach Behavior of Waste Glasses and Naturally Occurring Glasses", Corning Glass Works, Ceramics in Nuclear Waste Management, Chikalla, T.D., and Mendel, J.E., (eds), Cincinnati, Ohio, April 30 - May 2, 1979, CONF-790420, (1979), pp 233-237.

The leach behavior of simulated nuclear waste glasses of the sodium-borosilicate type with a low waste loading, and a zinc-borosilicate type with a high waste loading, have been compared with that of obsidians. Activation energies for reactions of the two waste glasses with pure water were determined. Leach rates were measured at 30 and 90°C.

12.003

Barkatt, A., et al., "Stability of Fixation Solids for High-Level Radioactive Wastes", Catholic University of America, High-Level Radioactive Solid Waste Forms, Casey, L.A., (ed), (CONF-781241), NUREG/CP-0005, (April 1979), pp 81-154.

Borosilicate and high-silica glass stability against devitrification are given under simulated conditions. Heat capacity was measured at 467 and 597°C, and electron microscope studies were conducted.

12.004

Barnes, S.M., et al., "Processing of High Temperature Simulated Waste Glass in a Continuous Ceramic Melter", Battelle Pacific Northwest Laboratories, Scientific Basis for Nuclear Waste Management, Vol.2, Northrup, C.J.M., Jr., (ed), Plenum Press, New York, (1980), pp 859-866.

A joule-heated ceramic lined melter has been developed at PNL to operate with glass compositions requiring electrode temperatures in excess of 1400°C. At these temperatures, glasses of superior durability are produced. The process was designed to vitrify the residual liquor component of the defense waste stored at Hanford. The resultant glass was found to have a Soxhlet leach rate of 4.75×10^{-5} g/cm²/day with a 20 percent waste loading. Further leach rate, crystallization and other characterization data can be found in the text of this report.

12.005

Barney, G.S., "Fixation of Radioactive Waste by Hydrothermal Reactions with Clays", Atlantic Richfield Hanford Co., ARH-SA-174, (January 1974), 35 p.

A means of converting radioactive wastes to solid, relatively insoluble, aluminosilicate minerals is described. This process can be applied to caustic sodium wastes such as neutralized Purex wastes, sodium salts and oxides produced by evaporation or calcination of liquid wastes. Leach rates of the resulting solids at 30°C are presented.

12.006

Barton, G.B., "Effects of Compositional Variation on 'Drip Temperature' and Leachability of Phosphate Solids Under Consideration for Fixation of Radioactive Residues", Battelle Pacific Northwest Laboratories, Solidification and Long-Term Storage of Highly Radioactive Wastes, Richland, Washington, February 14-18, 1966, CONF-660208, (April 1967), pp 549-565.

Phosphate glasses that offer promise as desirable waste forms can be made with a variety of compositions. It is reported that the thermal history of the melt has very significant effects on the properties of the final solid. Slow cooling to produce crystallized material generally reduces the "drip temperature" and increases the leachability of the glass.

12.007

Battelle Pacific Northwest Laboratories, "Quarterly Progress Report - Research and Development Activity - Waste Fixation Program", BNWL-1908, (June 1975), 78 p.

Corrosion rates are given for 304L stainless steel in simulated HLW glass melt from 950 to 1100°C. Leach rates are given for zinc borosilicate glasses by the Soxhlet leach test for cesium, strontium, and molybdenum. Three tests each lasting 24 hours were conducted.

12.008

Berreth, J.R., "Status Report: Development and Evaluation of Alternative Treatment Methods for Commercial and ICPP High-Level Solidified Wastes", Allied Chemical Corp., ICP-1089, (May 1976).

General alternative solid waste forms such as sintered glass-ceramics, metal matrices, and calcine-concretes are described and compared. Leach rates of zirconium calcine ceramics and the amount of sodium and chromium leached from zirconium-2nd-cycle calcine are determined. The strength of the sintered glass ceramic and calcine concrete is discussed.

12.009

Blanco, R.E., and Lotts, A.L., (eds), "High-Level Waste Program Progress Report for April 1, 1979, through June 30, 1979", Oak Ridge National Laboratory, ORNL-TM-7013, (1979).

Studies continued on the leachability of strontium, cesium, and plutonium from FUETAP (formed under elevated temperature and pressures) concretes into distilled water, spring water and Sandia brine at ambient temperatures. Laboratory experimental time was 12 days.

12.010

Blomqvist, G., "Leaching of French, English, and Canadian Glass Containing High-Level Waste", KBS Technical Reports 1-120, Report No. 8, (1977-1978).

A survey has been made of typical leach rates for French and English glasses used for incorporating highly radioactive waste. Calculations have been made to the time- and temperature-dependencies of the leach rates of cesium and strontium in the range 25 to 110°C using published values for the diffusion coefficients up to 70°C.

12.011

Blomqvist, G., "Calculation of Leaching of Certain Fission Products and Actinides from a Cylinder Made of French Glass", KBS Technical Reports 1-120, Report No. 24, (1977-1978).

The probable total leaching of the most important fission products and actinides have been tabulated for a cylinder of French HLW glass with a waste loading of approximately 9 percent fission products. The calculations cover the period between 30 and 10,000 yrs after removal from the reactor.

12.012

Blomqvist, G., "Leaching Trials with High-Level French Glass at Studsvik", KBS Technical Reports 1-120, Report No. 50, (1977-1978).

Glasses containing 20 percent waste oxides were experimentally leached using simulated groundwaters from Swedish rocks. The effect of temperature on leach rates was determined.

12.013

Boatner, L.A., et al., "Monazite and Other Lanthanide Orthophosphates as Alternate Actinide Waste Forms", Oak Ridge National Laboratory, Scientific Basis for Nuclear Waste Management, Vol. 2, Northrup, C.J.M., Jr., (ed), pp 289-296.

Leaching tests of orthophosphate crystals resulted in IAEA leach indices that were smaller than those reported for borosilicate glasses by a factor of 16 for CePO_4 in 4 M NaCl at 200°C and 250 psi, and a factor of 295 for La(Am) PO_4 in distilled water at 200°C and 250 psi.

12.014

Bonner, W.F., "High-Level Waste Immobilization Program: An Overview", Battelle Pacific Northwest Laboratories, PNL-3094, (September 1979), 42 p.

The release rates of various fission products from top glass sections due to leaching from 1 to 100 days are presented.

12.015

Bonnaud, R.A., et al., "Glasses as Materials Used in France For Management of High-Level Wastes", Commissariat a l'Energie Atomique, Ceramics in Nuclear Waste Management, Chikalla, T.D., and Mendel, J.E., (eds), Cincinnati, Ohio, April 30 - May 2, 1979, CONF-790420, pp 57-61.

Two kilogram blocks of waste glass containing 100 to 1000 curies of beta-gamma activity were leached by tap water at room temperature. The leach rates for ruthenium and cerium were found to be between 1 and 5×10^{-8} g/cm²/d and for cesium and strontium about 5×10^{-7} g/cm²/d. The leach rate appeared to reach steady state after 40 days. The leach rates of cesium and strontium were found to increase by a factor of 3 if the temperature was raised from 25 to 50°C, by a factor of 10 if the temperature was raised from 25 to 70°C, by a factor of 33 if the temperature was raised from 25 to 100°C, and by a factor of 180 if the temperature was raised from 25 to 150°C. The leach rate of plutonium was found to increase slightly with time. The increase, however, does not make up for the loss due to radioactive decay. No crystallization has been observed by scanning electron microscope investigation on alpha-doped glasses irradiated at 4×10^{17} alpha disintegrations per gram of glass.

12.016

Bonniaud, R.A., et al., "The Behavior of Actinides in Alpha-Doped Glasses as Regards to the Long Term Disposal of High Level Radioactive Materials", Commissariat a l'Energie Atomique, Scientific Basis for Nuclear Waste Management, Vol. 2, Northrup, C.J.M., Jr., (ed), Plenum Press, New York (1980), pp 117-225.

The investigation of the chemical and mechanical properties of waste glasses which contain alpha-emitting actinide elements and modifications in those properties is being studied for property changes over time. This paper describes the fabrication of the doped glasses, using americium-241, curium-244, and plutonium as doping elements; the actinide distribution in the glass samples; the helium buildup in the samples; and the leaching characteristics of the test samples.

12.017

Boult, K.A., et al., "Leaching of Radioactive Waste Storage Glasses", Harwell Atomic Energy Research Establishment, AERE-R-9188, (August 1978), 40 p.

Results of leach tests on various glass compositions proposed for the vitrification of the highly radioactive waste from the reprocessing of Magnox fuel are reported. Tests were carried out using both the Soxhlet apparatus and static conditions. Variables tested included the solution composition, effect of devitrification, time, temperature, specimen irradiation, and the effect of a surrounding can.

LEACHING

III-30

12.018

Boult, K.A., et al., "The Leaching of Radioactive Waste Storage Glasses", Ceramics in Nuclear Waste Management, Chikalla, T.D., and Mendel, J.E., (eds), Cincinnati, Ohio, April 30 - May 2, 1979, CONF-790420, (1979), pp 248-255.

Results are reported of leach tests under various conditions on glass compositions proposed for the vitrification of the highly radioactive waste from the reprocessing of Magnox fuel. Among the variables studied are waste composition, glass devitrification, the source and pH of the water, temperature, and radiation dose to the glass.

12.019

Boulton, J., and Gibson, A.R., " First Annual Report on the Canadian Nuclear Fuel Waste Management Program, Whiteshell Nuclear Research Establishment", AECL-6443, (December 1979), 75 p.

Samples of CRNL nepheline syenite glass at 60°C were leached in flow-through tests to determine cesium-137 loss rates. Two of the four tests were performed on glass embedded in quartz sand.

12.020

Bower, J.R., (ed), "Idaho Chemical Programs Annual Technical Report, Fiscal Year 1974", Allied Chemical Corp., ICP-1063, (January 1975), 181 p.

Leach tests were performed using distilled water at 90°C on the high-fired alumina-base ceramic sintered samples. The leaching period was 250 hrs.

12.021

Bradley, D.J. "History of Prototype High-Level Waste Canister SS-9 While in Air and Water Storage", Battelle Pacific Northwest Laboratories, PNL-2278 (November 1977), 34 p.

Canister SS-9 was filled with high-level phosphate ceramic waste material in 1969. Following 1.2 yrs water storage at 50°C, 3.5 yrs hot air storage at about 500°C, and 10 months water storage at 50°C, the canister failed. It was concluded from metallography that failure was due to stress-assisted intergranular attack enhanced by metal sensitization during the hot air storage period, and a high chloride ion concentration in the canister storage water. Leach rates of phosphate ceramic are given based on cesium-137 from one to 26 days for 14 samples.

12.022

Bradley, D.J., "Leaching of Fully Radioactive High-Level Waste Glass", Battelle Pacific Northwest Laboratories, PNL-2664, (September 1978), 101 p.

The release rates of 10 radioisotopes from fully radioactive waste glass in deionized water for a period of 1.75 yrs were determined using the IAEA procedure. A zinc borosilicate glass frit was blended with waste calcine in a 3 to 1 weight ratio to make the waste glass. For cesium and strontium, good agreement exists between the leach rates for simulated and fully radioactive glass of the same composition. The release rate was found to be dependent on time, sampling frequency, and type of element.

12.023

Bradley, D.J., "Waste Isolation Safety Assessment Program. Leaching of Actinides and Technetium from Simulated High-Level Waste Glass", Battelle Pacific Northwest Laboratories, PNL-3152, (August 1979).

The interactions of glass-waste-solution were studied using leach tests. Release rates were determined for technetium, uranium, neptunium, plutonium, americium, curium, and silicon in solutions of WIPP "B" brine, NaCl (287 g/l), NaCl (1.76 g/l), CaCl₂ (1.66 g/l), NaHCO₃ (2.52 g/l), and deionized water. The combinations which yield the lowest leach rates are given. An extensive appendix provides the accumulated data of the leach tests.

12.024

Bradley, D.J., "Non-Flow Release Studies", Battelle Pacific Northwest Laboratories, Task 2 Workshop Waste Isolation Safety Assessment Program, PNL-SA-7992, (1979), pp 45-66.

Four glasses of similar composition were prepared incorporating technetium-99, curium-244, uranium-233, and americium-243, and neptunium-237 and plutonium-239, for the purpose of leach testing. The procedure for making the doped glass beads is given. The leach rate was found to drop by a factor of 10 in the first 10 days. There was also an order of magnitude difference in the leach rates between WIPP "B" brine and a bicarbonate solution. Results of the IAEA leach tests are also given.

12.025

Bradley, D.J., "Basic Research for Evaluating Nuclear Waste Form Performance", Battelle Pacific Northwest Laboratories, Nuclear Technology, Vol. 51, (December 1980), pp 111-122.

This paper presents an overview of the types of basic research needed to assess the waste form performance and engineered barriers for different repository environments to allow defensible safety assessment. To assess the performance of a given waste form, water is assumed to penetrate the formation containing the repository. The leach rate, the temperature dependence of flow rates, effects on radionuclide release, effect on leach rate of solution composition (from deionized water to saturated salt brine), and other variables are discussed.

12.026

Bradley, D.J., "Leaching of Fully Radioactive High-Level Waste Glass and Waste-Geologic Environment Interaction Studies", Battelle Pacific Northwest Laboratories, Radioactive Waste in Geologic Storage, Fried, S., (ed), American Chemical Society Symposium Series 100, Washington, D.C., (1979), pp 75-92.

This report describes the preparation and leach testing of fully radioactive zinc borosilicate glass prepared from power reactor waste. Leach tests were conducted on this fully radioactive glass to: (1) determine the release rates of radioactive material from the glass; (2) compare leach rates to those of nonradioactive waste of the same composition; and (3) study elements not included in simulated waste glasses. Release rates were determined for 10 radioisotopes from this glass in deionized water for a period of 1.75 yrs.

12.027

Braithwaite, J.W., and Johnstone, J.K., "Chemical Durability and Characterization of Nuclear Waste Forms in a Hydrothermal Environment", Sandia Laboratories, Ceramics in Nuclear Waste Management, Chikalla, T.D., and Mendel, J.E., (eds), Cincinnati, Ohio, April 30 - May 2, 1979, CONF-790420, (1979), pp 269-273.

The chemical durability of a simulated copper borosilicate waste glass and titanate waste ceramic has been studied in hydrothermal environments which could be encountered in a bedded salt waste repository. The major parameters investigated include matrix corrosion and cesium leach rates.

12.028

Brezneva, N.E., et al., "Vitrification of High Sodium-Aluminum Wastes: Composition Ranges and Properties", Institute of Physical Chemistry, USSR. Scientific Basis for Nuclear Waste Management, Vol. 1, McCarthy, G.J., (ed), Plenum Press, New York, (1979), pp 43-50.

Leaching rates are given for some phosphate and silicate glasses containing Na_2O , Al_2O_3 , P_2O_5 , CaO , B_2O_3 , and SiO_2 , expressed in $\text{g/cm}^2/\text{day}$. Leach rates of glasses containing cesium, strontium, cerium and ruthenium are also given. It is concluded that up to 50 weight percent waste oxides can be incorporated into the glass. Some relative advantages and disadvantages of borosilicate and phosphate glasses are discussed.

12.029

Brotzman, J.R., "Vitrification of High-Level Alumina Nuclear Waste", Exxon Nuclear Idaho Co., Inc., Scientific Basis for Nuclear Waste Management, Vol. 2, Northrup, C.J.M., Jr., (ed), Plenum Press, New York, (1980), pp 215-222.

Borophosphate glass compositions have been developed for the vitrification of a high alumina calcined defense waste. The effect of substituting SiO_2 and P_2O_5 for the B_2O_3 on the viscosity and leach resistance was measured. The effect of the alkali to borate ratio and the Li_2O to Na_2O ratio on the melt viscosity and leach resistance were also measured.

12.030

Burkholder, H.C., et al., "Waste Isolation Safety Assessment Program Technical Progress Report for FY'77", Battelle Pacific Northwest Laboratories, PNL-2642, (November 1978).

This document reports FY'77 progress by the Waste Isolation Safety Assessment Program (WISAP) to develop technology that can be used as part of a comprehensive assessment program to evaluate the effectiveness of nuclear waste repository systems in preventing adverse effects to present and future humans and their environs. The work was centered on four subject areas, the most important of which is the measurement of waste form leach rates.

12.031

Claiborne, H.C., and Gera, F., "Potential Containment Failure Mechanisms and Their Consequences at a Radioactive Waste Repository in Bedded Salt in New Mexico", Oak Ridge National Laboratory, ORNL-TM-4639, (October 1974), 80 p.

Maximum leach rates from one row of waste canisters for various times after repository closure are calculated. The waste is in glass form with a particulate diameter of 2 mm, a density of 3.2 g/m^3 , and a leach rate of $10^{-7} \text{ g/cm}^2/\text{day}$. Assuming leaching of 765 canisters and groundwater flow of 130 liters/yr/m of width, groundwater contamination and deposition of radioactive material on the ground when water is used for irrigation is calculated. Also calculated are the doses to humans from heavy metals when farmland is irrigated with contaminated groundwater.

12.032

Clark, D.E., et al., "Corrosion Behavior of Zinc Borosilicate Simulated Nuclear Waste Glass", University of Florida, Ceramics in Nuclear Waste

Management, Chikalla, T.D., and Mendel, J.E., (eds), Cincinnati, Ohio, April 30-May 2, 1979, CONF-790420, (1979), pp 256-262.

The corrosion behavior of a zinc borosilicate glass containing simulated nuclear wastes is evaluated in acidic, neutral, and basic solutions. Auger electron spectroscopy coupled with argon ion milling and infrared reflection spectroscopy coupled with sequential polishing are used to determine the depth of the leached layers. Leach rates based on solution analysis are calculated for several species including cesium and strontium. Scanning electron microscopy in conjunction with energy dispersive x-ray spectroscopy is used to monitor surface morphological changes that accompany glass corrosion.

12.033

Cohen, J.J., et al., "Lead Encapsulation of Nuclear Waste: A Risk Assessment", Lawrence Livermore Laboratory, Transactions of the American Nuclear Society, Vol. 34, (1980), pp 350-351.

The use of lead as encapsulating material for nuclear waste is compared to borosilicate glass using several parameters, i.e., toxicity, leach rate, corrosion, etc. Leaching experiments were conducted on lead shot and zinc-borosilicate glass doped with both neptunium and plutonium at 25°C in 0.03 molar NaHCO₃. Volume flow rates of bicarbonate water were 10 to 300 ml/day for the glass experiment and 100 ml/day for the lead experiment. The inherent toxicity and long-term safety aspects of using lead as an encapsulating material was also considered.

12.034

Cole, H.S., et al., "Properties of Vitrified ICPP Zirconia Calcine", Exxon Nuclear Idaho Co., Inc., ENICO-1028, (July 1980) 38 p.

Properties of borosilicate glass were evaluated with a zirconia calcine containing from 10 to 40 percent waste. Soxhlet, acid, and static distilled water leach rates were determined. Viscosity and crystallinity measurements were made in annealed and in air-quenched glasses.

12.035

Coles, D.G., et al., "Single Pass Leaching of Nuclear Melt Glass by Groundwaters", Lawrence Livermore Laboratories, Radioactive Waste in Geologic Storage, Fried, S., (ed), American Chemical Society Symposium Series 100, Washington, D.C., (1979), pp 93-114.

The leaching concept used in these experiments differs significantly from most of those used by previous investigators because it uses a dynamic flow-through system in which the solid sample is continually exposed to fresh leaching solution. Such a system is designed to simulate actual underground conditions. It is a nonequilibrium system in which the displacement from equilibrium can be varied by changing the flow rate. Glasses examined are close analogues of vitrified HLW. Leaching was carried out at 25°C, and groundwater obtained from a well at a western site was the leaching media. This interim report covers the results of the first 120 days of a planned 420 day experiment.

12.036

Coons, W.E., and Patera, E.S., "Reactions of Spent Fuel and Reprocessing Waste Forms with Water in the Presence of Basalt", Rockwell Hanford Operations, National Waste Terminal Storage Program Information Meeting, Columbus, Ohio, December 9-11, 1980, ONWI-212, (December 1980), pp 121-124.

The leach rates of simulated spent fuel, simulated supercalcine and simulated borosilicate glass are discussed. Some leach data at temperatures of 200 and 300°C are given.

12.037

Crandall, J.L., "U.S. Program for the Immobilization of High-Level Nuclear Wastes", Savannah River Laboratory, Transactions of the American Nuclear Society, Vol. 32, (1979), pp 101-104.

A large number of alternative waste forms have been proposed for waste immobilization. They need to be carefully assessed for processing feasibility, resistance to transport conditions, and resistance to repository conditions. Table II of this paper compares nine characteristics of thirteen different waste forms. Characteristics considered include: (1) development status; (2) process complexity; (3) process flexibility; (4) waste loading capacity; (5) dispersion and impact resistance; (6) long-term stability; (7) fire resistance; (8) leachability at 100°C; and (9) leachability at 350°C. Waste forms considered include: (1) calcine; (2) rich clay; (3) normal concrete; (4) hot-pressed concrete; (5) pelletized calcine; (6) glass; (7) clay ceramic; (8) super-calcine; (9) SYNROC; (10) glass ceramic; (11) pellets in metal matrix; (12) coated super-calcine in metal matrix; and (13) cermet.

12.038

De, A.K., et al., "Fixation of Fission Products in Glass Ceramics", Hahn-Meitner-Institut, Management of Radioactive Wastes from the Nuclear Fuel Cycle, Vol 2, (CONF-760310), STI/PUB/433, International Atomic Energy Agency, Vienna, (1976), pp 63-73.

The document is primarily directed towards the processing steps for manufacture of glass ceramics from borosilicate glass. The phases contained in the glass ceramic are identified. The leach rates from the parent borosilicate glass are compared to the leach rates of the glass ceramic. The impact resistance of glass ceramics is reported to be substantially improved over the impact resistance of glass. The coefficient of thermal expansion is smaller for glass ceramics than for glass, which indicates an increased thermal stability for glass ceramics.

12.039

De, A.K., et al., "Development of Glass Ceramics for the Incorporation of Fission Products", Hahn-Meitner-Institut, Bulletin of the American Ceramic Society, Vol. 55, No. 5, (1976), pp. 500-503.

Leach resistance of four borosilicate glasses was tested by boiling 2 grams of glass powder (grain size 0.0315 to 0.05 cm) for 1 hr in distilled water.

12.040

Delegard, C.H., et al., "Fixation of Radioactive Waste by Reaction With Clays: Progress Report", Atlantic Richfield Hanford Company, ARH-ST-124, (July 1975).

The fixation of radioactive waste by reaction with bentonite and kaolin clay is discussed. Data on leachability, mechanical strength and thermal and radiolytic stability are given. Clay calcination is also discussed.

12.041

Devell, L., and Hesbol, R., "Leachable Gap Activity", KBS Technical Reports 1-120, Report No. 109, (1977-1978).

In the event of groundwater penetration into fuel pins after ultimate disposal in crystalline rock, a certain fraction of the volatile fission products iodine-129 and cesium-135 may be leached from the fuel in a shorter period of time than the rest of the fission products. Their leach rates have been estimated from leach experiments.

12.042

Dukes, M.D., et al., "Multibarrier Storage of Savannah River Plant Waste", Savannah River Laboratory, Scientific Basis for Nuclear Waste Management, Vol. 2, Northrup, C.J.M., Jr., (ed), Plenum Press, New York, (1980), pp 231-238.

Borosilicate glass will perform well as part of a multibarrier system for SRP HLW. Storage temperatures can be kept less than 100°C with appropriate spacing of the reference canisters. Under dry conditions the waste package should remain intact for very long periods of time. Even if the package is breached and water contacts the glass, the radionuclide release rate will be very low. Cumulative leaching of typical SRP glass in brine is given. The initial leach rate was 8.4×10^{-6} g/cm²/day. Candidate canister material, Cor-Ten A and Type 304L stainless steel, heated with salt for 10,000 hrs showed no significant reaction with dry salt.

12.043

Eklund, E., and Forsyth, R., "Leaching of Irradiated UO₂ Fuel", KBS Technical Reports 1-120, Report No. 70, (1977-1978).

Leaching tests over a period of 105 days have been performed on 20 mm long sections of an irradiated fuel rod. Leaching was performed using two solutions, a synthetic Swedish groundwater and distilled water as reference, at 60°C without mechanical stirring.

12.044

Elliot, M.N., and Auty, D.B., "The Durability of Glass for the Disposal of Highly Radioactive Waste. Discussion of Method and Effect of Leaching Conditions", Glass Technology, Vol. 9, No. 1, (1968), pp 5-13.

III-41

A new technique is described for measuring low leach rates of hazardous fission products. This report first describes results of tests to check the accuracy and reproducibility of the new method. Secondly, the report discusses the effects of leach rate, leach fluid, and leach temperature on the rates of removal of cesium, strontium, cerium, and terbium from two borosilicate samples and one phosphate glass sample in distilled water and in sea water. Results showed little effect due to cooling rate of the glass or the leach fluid used. The cesium and strontium leach factors decreased by approximately an order of magnitude as the temperature was increased from 30 to 60°C. The borosilicate glasses were slightly more durable than the phosphate glass, the cesium leach factors being 1×10^{-6} to 4×10^{-6} and 10^{-5} to 10^{-6} g/cm²/day, respectively. Leach factors indicate that a leached silicate layer with fission product oxides formed at the glass surface.

12.045

Ewest, E., "Calculations of Radioactivity Release Due to Leaching of Vitrified High Level Waste", Hahn-Meitner-Institut, Scientific Basis for Nuclear Waste Management, Vol. 1, McCarthy, G.J., (ed), Plenum Press, New York, (1979), pp 161-168.

A model is presented for the calculation of radioactivity released from vitrified waste. An example is given for a borosilicate glass block which is assumed to be exposed to water for a million years.

12.046

Flynn, K.F., et al, "Effects of Temperature and Leaching Media on the Leach Rates of Nuclear Waste Materials", Argonne National Laboratory, Transactions of the American Nuclear Society, Vol. 32, (1979), pp 394-396.

LEACHING

III-42

Leach rates measurements were performed on PNL waste glass in five different leaching media. Leach rates for six representative isotopes are tabulated as a function of length of time for the leach test. The five leaching media were: (1) distilled water (25°C, pH 5.8), (2) tap water (25°C, pH 9.7), (3) carbonated saturated distilled water (25°C, pH 3.9), (4) WIPP brine--saturated distilled water (25°C, pH 7.6), and (5) WIPP brine--saturated distilled water (109°C, pH 7.6).

12.047

Flynn, K.F., "Leach Rate Characterization of Solid Radioactive Waste Forms", Argonne National Laboratory, Scientific Basis for Nuclear Waste Management, Vol. 1, McCarthy, G.J., (ed), Plenum Press, New York, (1979), pp 149-155.

Leach rates for a number of proposed radioactive waste forms were measured using a special neutron activation analysis technique that permitted determination of leach rates as low as 10^{-9} g/cm²/day. Problems associated with surface area determination, sample preparation, and data interpretation are discussed.

12.048

Flynn, K.F., et al, "Method for Determining Leach Rates of Simulated Radioactive Waste Forms", Argonne National Laboratory, Radioactive Waste in Geologic Storage, Fried, S., (ed), American Chemical Society Symposium Series 100, Washington, D.C., (1979), pp 115-128.

Measurement of leach rates of radioactive waste by a method that incorporates neutron activation is described. Sintered granules and lead matrix materials were the waste forms examined using the neutron activation method. Some preliminary leach rate data for strontium, barium, and some rare

earths over a seven day leaching period are presented. Other methods for leach rate measurements are reviewed. Some advantages and disadvantages of various leaching procedures are briefly discussed.

12.049

Flynn, K.F., et al., "Resistance of High Level Waste Materials to Dissolution in Aqueous Media", Argonne National Laboratory, Scientific Basis for Nuclear Waste Management, Vol. 2, Northrup, C.J.M., Jr., (ed), Plenum Press, New York, (1980), pp 103-108.

Results from the studies of the rate of destruction by aqueous media of proposed barrier materials as well as the rate of leaching from proposed matrix materials are reported. The effect of temperature on leach rate of the glass matrices studied is shown to be the predominant variable associated with the leaching medium.

12.050

Gelin, R., "Leaching of Spent Nuclear Fuel (Irradiated Uranium Oxide) Following Direct Deposition", KBS Technical Reports 1-120, Report No. 13, (1977-1978).

Leach rates for cesium, antimony, strontium, yttrium, plutonium, and curium from LWR spent fuel in water are reported. Relative leach rates of these elements from spent fuel are compared to those obtained for the same elements from borosilicate glass.

12.051

Grambow, B., and Lutze, W., "Chemical Stability of a Phosphate Glass Under Hydrothermal Conditions", Hahn-Meitner-Institut, Germany, Scientific Basis for Nuclear Waste Management, Vol. 2, Northrup, C.J.M., Jr., (ed), Plenum Press, New York, (1980), pp 109-117.

Phosphate glasses free of sodium may be a suitable nuclear waste form in terms of leach resistance at temperatures below 370°K. At higher temperatures, the leach resistance decreases rapidly.

12.052

Grover, J.R., and Elliot, M.N., "Long Term Examination of Glasses for the Final Process", Harwell Atomic Energy Research Establishment, Solidification and Long Term Storage of Highly Radioactive Wastes, Richland, Washington, February 14-18, 1966, CONF-660208, (April 1967), pp 585-605.

Glasses to be used for the safe storage of fission products must maintain their stability in spite of severe conditions of temperature and radiation. Activity can be released from glass in storage either by leaching or by volatilization. A sophisticated technique is described for measuring very low leach rates after glasses have been subjected to high radiation doses and a variety of thermal treatments.

12.053

Guber, W., et al, "Preparation and Characterization of an Improved High Level Radioactive Waste (HAW) Borosilicate Glass", Nuclear Research Center, Karlsruhe, Scientific Basis for Nuclear Waste Management, Vol. 1, McCarthy, G.J., (ed), Plenum Press, New York (1979), pp 37-42.

The criterion formulated for leach resistance is 10^{-5} g/cm²/day in distilled water, salt brine, or carnallite brine. Composition and properties of selected glass products are given. Investigations with an optical microscope, scanning electron microscope, microprobe, and X-ray diffractometer were made on the specimens annealed in a gradient furnace.

12.054

Halaszovich, S., et al, "FIPS--A Process for the Solidification of Fission Product Solutions Using a Drum Drier", Institut fuer Chemische Technologie, Management of Radioactive Wastes from the Nuclear Fuel Cycle, Vol. 1, International Atomic Energy Agency, Vienna, (CONF-760310), STI/PUB/433, (1976), pp 315-323.

Most of the document is concerned with the processing steps in FIPS. A short description of the final waste form is included which identifies the waste loading as 20 percent fission product oxides and the leach rate as 3 to 6×10^{-5} g/cm²/day in unspecified conditions.

12.055

Hall, A.R., et al., "Development and Radiation Stability of Glasses for Highly Radioactive Wastes", Harwell Atomic Energy Research Establishment, Management of Radioactive Wastes from the Nuclear Fuel Cycle, Vol. 2, International Atomic Energy Agency, Vienna, (CONF-760310), STI/PUB/433, (1976), pp 3-14.

The variations of formation temperature, crystallizing behavior, and leach resistance with composition changes for sodium-lithium borosilicate glasses suitable for vitrifying Magnox waste are discussed. Viscosities were measured between 400 and 1050°C. The principal phases which occur were identified as magnesium silicate, magnesium borate, and ceric oxide. The leach rate of polished discs in pure water at 100°C does not decrease with time if

the fragile siliceous layer that is observed to occur is taken into account. The effect of 100 years' equivalent to alpha- and beta-irradiation on glass properties is discussed. Samples 1 cm square and 1 mm thick were irradiated with electrons accelerated to 0.5 MeV in a Van de Graaff accelerator. Irradiation in an electron microscope caused minute bubbles in lithium-containing glasses above 200°C.

12.056

Heimerl, W., et al., "Research on Glasses for Fission Product Fixation, Summary Report, January 1968 - June 1971", Hahn-Meitner-Institut, HMI-B-109, (September 1971), 97 p.

Short- and long-term leaching was done on different glasses for confirmation of durability. For the short-term leach testing, the glass was powdered to a grain size of 0.3-0.5 mm and a sample of 1 g leached in 50 ml distilled water at 98°C for 1 hr. For long-term leach testing, much larger samples were used. The temperature ranged from 20 to 50°C and the pH from 4 to 5. The length of time was 100 days. Leach studies were also performed with Filtrolite glasses and their crystallization products.

12.057

Hench, L.L., et al., "Surface Leaching of Glasses and Glass-Ceramics", University of Florida, High-Level Radioactive Solid Waste Forms, Casey, L.A., (ed), Denver, Colorado, December 19-21, 1978, (CONF-781241), NUREG/CP-0005, (April 1979), pp 199-238.

An experimental approach is presented which enables one to predict the stability of glass and glass-ceramic waste forms for long periods of time. The method is based upon a theoretical correlation of changes in thickness of leached layers with given changes in ratio of surface area exposed to a

specific solution volume. The effect of crystallization on leaching is examined by comparing rates of attack on polycrystalline and glassy materials of equivalent compositions. Results of the variables on leaching are applied to understanding the behavior of a PNL zinc-borosilicate glass with simulated waste loadings. Reviewing the literature, leach rates at room temperatures vary from 10^{-4} to 10^{-8} g/cm²/day.

12.058

Hespe, E.D., (ed), "Leach Testing of Immobilized Radioactive Waste Solids, A Proposal for a Standard Method", Atomic Energy Review, Vol. 9, No. 1, (1971), pp 195-207.

In 1969 the International Atomic Energy Agency brought together a group experienced in the field to give advice on the standardization of tests for leaching of immobilized waste solids. Details are given for two leach rate test methods; one an intercomparison method, and one to approximate the behavior of an immobilized waste solid form under conditions of the storage environment. The article discusses the possibility of extrapolating the test data to a disposal program. The solidification process and the materials used are not elaborated.

12.059

Hill, L.R., et al., "Parametric Study of Glass Formulations", Rockwell Hanford Operations, RHO-SA-117, (August 1979), 7 p.

Using main-effects and first-order interactions, a model for predicting weight percent of the composition dissolved by water was constructed having an R^2 value of 0.973.

12.060

Hill, M.D., and Grimwood, P.D., "Preliminary Assessment of the Radiological Protection Aspects of Disposal of High-Level Waste in Geologic Formations", National Radiological Protection Board, NRPB-R69, (January 1978), 75 p.

This report studies the potential radiological consequences of disposing of vitrified HLW in geologic formations. Events which could lead to the release of radioactivity from a geologic repository are reviewed and ingress of groundwater is found to be the principal mechanism by which radioactivity may be transported back into the biosphere. Geologic containment failure mechanisms are discussed, including natural rapid events, natural geologic events, geologic events caused by the repository, and human actions. The quantity and the composition of the waste is discussed, along with a release model of the canister. The corrosive properties of the canister and the properties of the solidified waste are examined.

12.061

Holton, L.K., et al., "Alternatives for Vitrification of Existing Commercial High-Level Waste by Spray Calcination/In-Can Melting", Battelle Pacific Northwest Laboratories, Waste Management '80, Post, R.G., (ed), University of Arizona, Tucson, (1980), pp 365-380.

Simulated waste chemicals were combined with glass-forming chemicals in batches of 100 g. Physical characteristics of the laboratory pilot-plant glasses with silicon metal added as a reducing agent showed leaching to be 3.3 to 8 weight percent for Soxhlet at a pH of 4. Thermal expansion was 1.12×10^{-9} in/°C over the range of 100 to 400°C. An option experiment gave leach results of 2.5 to 9.1 weight percent for Soxhlet at a pH of 4. Thermal expansion was 9.93×10^{-6} in/°C over the same temperature range.

12.062

Houser, C., et al., "Characterization of Leached Surface Layers on Simulated High-Level Waste Glasses by Sputter-Induced Optical Emission", Pennsylvania State University, Scientific Basis for Nuclear Waste Management, Vol. 1, McCarthy, G.J., (ed), Plenum Press, New York, (1979), pp 131-139.

The leaching process in simulated waste encapsulant glasses was studied by measuring the compositional depth-profiles of hydrogen (from water); the glass framework formers silicon and boron; the alkalis, sodium and cesium; the alkaline earths, calcium and strontium; the transitional metals, molybdenum and iron; and the rare earths, lanthanum, cerium, and neodymium using the technique of sputter-induced optical emission. The leaching process of these glasses is highly complex. In addition to alkali/hydrogen exchange, there is breakdown of the glass framework, buildup of barrier layers on the surface, and formation of layered reaction zones of distinctly different chemistry, all within the outer micrometer of the glass.

12.063

Jacquet-Francillon, N., et al., "Glass as a Material for the Final Disposal of Fission Products", CEA Centre de Marcoule, Radiochemica Acta, Vol. 25, No. 3/4, (1978), pp 231-240.

This paper describes the desirable properties of borosilicate glass and the effects which degrade those properties. Leaching is identified as the primary mechanism for removal of radionuclides from the glass, and the effects of time, temperature, and pH on the leach rate are discussed. Temperature is identified as the cause of devitrification, and temperature limits to avoid devitrification in aluminoborosilicate glass and borosilicate glass are identified. It is reported that a dose of 1.2×10^{11} rads from high-energy electrons does not modify the properties of glass nor the structure of glass.

Also, stored energy in glass saturates at about 10^{18} alpha disintegrations per gram and the sudden release of stored energy would not raise the temperature of glass to above 300°C. Finally, the effect of helium generation on the mechanical strength of glass is discussed.

12.064

Kaernbraenslesaeckerhet, "Handling of Spent Nuclear Fuel and Final Storage of Vitrified High-Level Reprocessing Waste. 3. Facilities", INIS-MF-4461 (1978), 140 p.

Chapter 4, Reprocessing and Vitrification, includes a review of solubility of the vitrified waste (time dependence of leaching rate, influence of groundwater composition, influence of pH on resistance to leaching, influence of temperature, and leaching tests with French glass at Studsvik) and durability of the waste glass (resistance of French borosilicate glass to radiation, mechanical properties, thermal stability, etc.). Facilities described are designed for the handling and storage of vitrified HLW and have been designed for a total waste capacity corresponding to approximately 9,000 metric tons of uranium, which is the quantity which would be produced by 13 reactors operating over a period of 30 yrs.

12.065

Katayama, Y.B., "Leaching of Irradiated LWR Fuel Pellets in Deionized and Typical Ground Water", Battelle Pacific Northwest Laboratories, BNWL-2057 (July 1976), 78 p.

Results of leach test experiments on enriched, 5.81 weight percent uranium-235, UO_2 fuel pellet fragments removed from LWR fuel rods irradiated to an average burnup of 54,450 MW/MTU are presented. Leach tests were performed at 25°C in static Hanford groundwater, deionized water, and distilled water. The experimental period was from 20 to 21 weeks.

12.066

Katayama, Y.B., "Spent LWR Fuel Leach Tests", Battelle Pacific Northwest Laboratories, Task 2 Workshop Waste Isolation Safety Assessment Program, Bradley, D.J., (ed), PNL-2982, (April 1979).

This is a continuation of the deionized-water leach tests beyond the original standard one year period which has allowed the observation of an extended-term effect. This effect, the effect of spent fuel burnup, and the durability of spent fuel in sea brine solution comprise this report. Spent LWR fuels with burnups of 54.5, 28, and 9 MWd/kgU were leach-tested in deionized water at 25°C. Fuel burnup has no apparent effect on the calculated leach rates based upon the behavior of cesium-137, plutonium-239, and plutonium-240. A leach test on 54.5 MWd/kgU spent fuel in synthetic sea brine showed that the cesium leach rate is lower in sea brine than in deionized water. A rise in the leach rate was observed after approximately 600 days of cumulative leaching. During the rise, the leach rate for all the measured radionuclides became nearly equal. Evidence suggests that exposure of new surfaces to the leachate may cause the increase.

12.067

Katayama, Y.B., and Mendel, J.E., "Leaching of Irradiated Fuel Pellets in Deionized Water, Sea Brine, and Typical Ground Water", Battelle Pacific Northwest Laboratories, PNL-SA-6416, (1977).

The goal of this program was to generate leach rate data for irradiated fuels and to compare these data with those from other waste forms. The two major experimental parameters that were considered in the LWR fuel leaching studies were: worst case fuel and groundwater, as opposed to other reference leachants.

12.068

Katayama, Y.B., and Bradley, D.J., "Long-Term Leaching of Irradiated Spent Fuel", Battelle Pacific Northwest Laboratories, Scientific Basis for Nuclear Waste Management, Vol. 2., Northrup, C.J.M., Jr., (ed), Plenum Press, New York, (1980), pp 323-334.

Spent fuel with burnups of 9, 28, and 54 MWd/kgU were leach-tested at 25°C in deionized water in a Paige apparatus. No discernable differences in long term leach rates were observed relative to burnup. Deionized water gave the highest leach rates, and calcium chloride solution gave the lowest. Comparisons between spent fuel and borosilicate waste glass leach rates were made.

12.069

Kawamoto, T., (ed), "Chemical Durability of Glasses Suitable for the Storage of High Level Radioactive Wastes. II. On the Continuous Leach Method for the Low Temperature Melting Glasses of Borosilicate System", ORNL-TR-4632, (1975), 17 p.

To develop glasses suitable for the long-term storage of HLW, the leach rates of the glasses composed of 35 percent SiO_2 , 15 percent B_2O_3 , 25 percent Na_2O , 5-35 percent CaO , and 20 to 50 percent water oxides were investigated. The tests were carried out using the continuous leach method for a period of one week and for 280 days. Fresh and devitrified glass samples were studied.

12.070

Kelley, J.A., "Evaluation of Glass as a Matrix for Solidification of Savannah River Plant Waste", Savannah River Laboratory, DP-1382, (May 1975), 28 p.

III-53

Soxhlet leach tests of borosilicate glass mixes were done for sodium and strontium for 24 hrs at about 99°C. For comparison, Corning 7740 Pyrex glass and National Bureau of Standards glass 710 were leached by the same method. Another leach test for cesium-137 and strontium-85 was done over 99 days at 25°C. The properties of waste glass forms containing curium-244 and plutonium-238 are given.

12.071

Kelley, J.A., "Evaluation of Glass as a Matrix for Solidification of Savannah River Plant Waste", Savannah River Laboratory, DP-1397, (October 1975), 21 p.

Laboratory tests were performed on SRP waste tank sludge which was vitrified into borosilicate glass by melting at 1150°C. Leach rates are given for cesium-137, strontium-90, antimony-125, and alpha activity. Testing time was up to 100 days. Heating the glasses at 600°C for one month caused devitrification which increased leachability. Some of the glasses were leached in distilled water at 25°C. Comparisons are given of leachability of glass in heated and unheated conditions.

12.072

Kelley, J.A., et al., "Correlation of Radionuclide Leachabilities with Microstructures of Glasses Containing Savannah River Plant Waste", Savannah River Laboratory, DP-1411, (May 1976).

Leachabilities of cesium and strontium from the glass matrices were correlated with the microstructures of glasses containing waste sludge from two SRP waste tanks. The effects of devitrification of the glass on the leachability were examined and compared to the leachability of nondevitrified glass.

12.073

Kelley, J.A., et al., "Vitrification Studies for Savannah River Plant Radioactive Wastes", Savannah River Laboratory, DP-MS-76-76, (April 1977), 22 p.

Glasses containing 35 to 45 weight percent of simulated waste were irradiated to 10^{10} rads of gamma radiation from a cobalt-60 source. Curium-244 and plutonium-239 were incorporated into glasses to determine alpha radiation stability. The waste forms were found to have low leachability and were stable to radiation and thermal treatments.

12.074

Kenna, B.T., et al., "Long-Term Elevated Temperature Leaching of Solid Waste Forms", Scientific Basis for Nuclear Waste Management, Vol. 1, McCarthy, G.J., (ed), Plenum Press, New York, (1979), pp 157-160.

Long-term Soxhlet leaching of simulated waste glass and ceramic materials was initiated to determine leaching behavior of complex waste forms. A cyclic leaching pattern was found in all systems over a 20-month period. Maxima and minima were observed in the leaching rates of all components studied with the minima coinciding.

12.075

Kenna, B.T., and Murphy, K.D., "Mechanism for Elevated Temperature Leaching", Sandia Laboratories, Scientific Basis for Nuclear Waste Management, Vol. 2, Northrup, C.J.M., Jr., (ed), Plenum Press, New York, (1980), pp 191-198.

The examination of test samples that underwent long-term elevated temperature leaching and subsequent electron microprobe analysis is described. A cyclic leaching pattern was found in all systems over a 20-month period. It appears that the leaching of mobile ions by simple diffusional processes is modified by more complex chemical interactions. The release of immobile ions is primarily a function of their chemical interactions in the matrix which suggests that these ions may be complex species when released into solution. A mechanism is proposed which incorporates these ideas and the cyclic phenomenon observed.

12.076

Kirkbride, R.A., "Development of a Pelleted Waste Form for High-Level Alumina Wastes", Exxon Nuclear Idaho Co., Inc., ENICO-1051, (September 1980).

A formulation to pelletize simulated high-level ICPP alumina waste calcine was developed. The pellets are formed on a 41-centimeter-diameter disc pelletizer using 5 weight percent bentonite, 2 weight percent metakoalin, and 5 weight percent calcium hydroxide as a solid binder and a solution of 7 molar phosphoric acid and 4 molar nitric acid as a liquid binder. After drying and heat treatment at 800°C for 2 hours, the average crush strength of the pellets is 3.9 MPa, and the pellets have a leach resistance of 10^{-3} g/cm²/day based on the Soxhlet leaching for 40 hour at 95°C with distilled water.

12.077

Kulichenko, V.V., et al., "Glass-Crystalline Materials for Active Waste Incorporation", The State Committee for Atomic Energy, USSR, Ceramics in Nuclear Waste Management, Chikalla, T.D., and Mendel, J.E., (eds), Cincinnati, Ohio, April 30 - May 2, 1979, CONF-790420, (1979), pp 374-377.

This paper presents the USSR's efforts to develop glass-crystalline materials that have increased chemical and thermal stability with a reduction of waste volume. The primary material studied is a basalt based pyroxene (CaO-MgO-SiO_2). The co-melting of a pyroxene composition charge and calcined wastes in a weight ratio of 30:70 and 40:60 at a temperature of approximately 1350°C results in a fine crystal material of high strength and chemical stability, characterized by sodium and cesium leach rates of approximately 10^{-6} g/cm²/day and a strontium leach rate of 10^{-7} g/cm²/day with a depth of attack of 22 to 25 angstroms on contact with distilled water. As compared with glass materials, the use of pyroxene materials permits a two-fold decrease in volume of solidified materials and an eightfold increase in material loading per unit storage volume through an increase in allowable storage temperature.

12.078

Kupfer, M.J., et al., "Application of the Hanford Thermite Process to Increase Immobilization of In-Tank Solidified Waste", Atlantic Richfield Hanford Co., ARH-2458, (September 1972).

The leach rates of ITS-1 and ITS-2 were determined after 24 and 96 hrs. The specific leach rates of cesium and strontium were measured.

12.079

Lackey, W.J., et al., "Sol-Gel Technology Applied to Glass and Crystalline Ceramics", Oak Ridge National Laboratory, Waste Management '80, Post, R.G., (ed), University of Arizona, Tucson, (1980), pp 391-417.

SYNROC microspheres and pellets containing up to 25 percent simulated defense waste were prepared by the sol-gel process. The leach rates of SYNROC-B based on the isotopes plutonium-239, americium-241, and curium-244 are nearly 1.5 orders of magnitude lower than data reported for glass.

12.080

Lamb, K.M., "Development of a Metal Matrix for Incorporating High-Level Commercial Waste", Idaho National Engineering Laboratory, ICP-1144, (March 1978), 37 p.

Simulated calcined and pelleted waste were incorporated into aluminum and lead on a bench-scale model. Leach rates of the matrices were improved by pelletizing the calcine prior to metal incorporation. The Soxhlet leach rate at 95°C was 10^{-4} to 10^{-5} g/cm²/day and the percent weight loss per 100 hrs was 0.4 to 0.5.

12.081

Lamb, K.M., and Cole, H.S., "Development of a Pelleted Waste Form, High-Level ICPP Zirconia Wastes", Idaho National Engineering Laboratory, ICP-1185, (April 1979), 32 p.

Simulated high-level ICPP zirconia waste calcine is pelletized on a 41-cm diameter pelletizer using 5 percent bentonite, 2 percent metakaolin, and 2 percent boric acid for solids binders and 7 molar phosphoric and 4 molar nitric acid as the liquid binders. After heat treatment at 800°C for 2 hrs, the pellets are impact resistant and have a leach resistance of 10^4 g/cm²/day, based on Soxhlet leaching for 100 hrs at 95°C with distilled water.

12.082

Lanza, F., and Parnisari, E., "Evaluation of Long-Term Leaching of Borosilicate Glass in Pure Water", Commission of the European Communities, Ceramics in Nuclear Waste Management, Chikalla, T.D., and Mendel, J.E., (eds), Cincinnati, Ohio, April 30 - May 2, 1979, CONF-790420, (1979), pp 238-242.

In order to confirm the validity of the model of the dissolving sphere, a series of tests of leaching of borosilicate glasses has been performed. The leach rate is followed by measurements of weight loss and analysis of the surface composition. Accumulation at the surface of the less soluble cations is studied.

12.083

Larker, H.T., "Densification of Calcines and Direct Containment of Spent Nuclear Fuel in Ceramics by Hot Isostatic Pressing", ASEA, Sweden, (1979), 16 p.

A seventy-two hour Soxhlet test at approximately 100°C of crushed super calcine in a mullite matrix provided data on leach rates of Cs_2O , SrO , and MoO_3 . Also, corrosion rates of hot isostatically pressed aluminum oxide with a density of nearly 4 g/cm³ are given.

12.084

Lazarev, L.N., et al., "Use of Natural Aluminosilicates and Porous Ceramic Materials for the Inclusion of Radioactive Wastes", V.G. Khlopin Radium Institute, USSR, Ceramics in Nuclear Waste Management, Chikalla, T.D., and Mendel, J.E., (eds), Cincinnati, Ohio, April 30-May 2, 1979, CONF-790420, (1979), pp 197-182.

In laboratory-scale experiments on simulated liquid wastes, it has been shown that inorganic materials, such as ditomite and shamote, are a promising method for incorporation of liquid radioactive wastes into a solid matrix. The operations of solution absorption by porous materials, drying and calcination of salts in pores, and the subsequent conversion into glassy phosphate-silicate products seem to be promising from a technological point of view. The product is characterized by a sodium leaching rate of the order

of 10^{-5} g/cm²/day and exhibits good resistance to crystallization. The content of various oxides in the wastes can attain 15 to 20 weight percent.

12.085

Lin, K.H., et al., "Process for Immobilization of High-Level Solid Wastes by Glass and Ceramic Matrices", Oak Ridge National Laboratory, Ceramics in Nuclear Waste Management, Chikalla, J.D., and Mendel, J.E., (eds), Cincinnati, Ohio, April 30 - May 2, 1979, CONF-790420, (1979), pp 354-358.

Processes for the immobilization of reprocessed HTGR fuels, including waste hulls and fissile particles, are presented. Waste-matrix material was formed into cylinders at pressures of 35, 71, and 142 MPa and sintered at temperatures ranging from approximately 500 to 1025°C for approximately 1 hr. The cylinders that showed the most desirable properties (high mechanical strength, low surface area, and low leachability) contained 50 percent SiC, 25 percent soft glass, and 25 percent red clay. The effects of pressure of cylinder formation and sintering temperature on the leachability appear to be minor except for initial cesium leachability which is significantly reduced at higher pressures and temperatures.

12.086

Malow, G., et al., "Testing and Evaluation of the Properties of Various Potential Materials for Immobilizing High Activity Waste. First Annual Report (1977)", Hahn-Meitner-Institut, Commission of the European Communities, EUR-6213, (1979), 38 p.

Five Soxhlet leach tests were carried out on separate specimens of undoped glasses and on two specimens of the plutonium-238 doped glasses. Testing was done at 100°C and was completed for samples treated for 2 hrs, 1 day, 10 days, and 100 days. Grain titration experiments were used to

evaluate the water resistance of the glasses, and these results are compared with the Soxhlet results.

12.087

McCarthy, G.J., and, Grutzeck, M.W., "Preliminary Evaluation of the Characteristics of Nuclear Wastes Relevant to Geological Isolation in Basalt", Pennsylvania State University, RH0-C-12, (May 1978), 33 p.

The physical and chemical characteristics of radioactive waste which have bearing upon the concept of geologic storage in basalt are reviewed and discussed. Each class of waste form (spent unprocessed fuel, calcine, glass, and supercalcine) has its own unique characteristics which may or may not affect the concept of final storage of radioactive waste in basalt. These characteristics form the basis for scientific evaluation of the relative merits of each of the waste/basalt pairs. Most of the leachability measurements have been taken by the Paige (25°C), Soxhlet (95°C), or IAEA (25°C) distilled water methods and are not representative of leaching behavior under the hydrothermal conditions which are of concern in the immediate repository during the thermal period.

12.088

McCarthy, G.J., and Davidson, M.T., "Ceramic Nuclear Waste Forms: II. A Ceramic-Waste Composite Prepared by Hot Pressing", Pennsylvania State University, Bulletin of the American Ceramic Society, Vol. 55, No. 2, (1976), pp 190-194.

Leachability of different glasses and glass waste forms is given as percent weight loss resulting from 72 hour Soxhlet tests. This method, in common with all current leachability tests, does not allow the projection with certainty of the durability of even the most resistant waste forms having thousands of years contact with water. Weight losses given represent the combined effect of chemical dissolution and physical weathering.

12.089

McElroy, J.L., et al., "Waste Solidification Program Summary Report, Volume II, Evaluation of WSEP High-Level Waste Solidification Processes", Battelle Pacific Northwest Laboratories, BNWL-1667, (July 1972).

The fraction of cesium-137 that was leached from a WSEP pot calcine product is shown as a function of time (Figure 6.3). Also given are leach rates of RLG product at 25°C. Measurements were made on laboratory-nonradioactive samples for comparison. Further studies were done on spray calcine-melter products at 25°C for times of up to two years. Leach rates for the in-pot melting products are given at a temperature of 25°C for 36 weeks. Devitrification of the phosphate glass products results in a very significant increase in leachability.

12.090

McElroy, J.L., "Quarterly Progress Report, Research and Development Activities, Waste Fixation Program, January through March 1974", Battelle Pacific Northwest Laboratories, BNWL-1826, (April 1974).

This progress report primarily covers studies of the effects of composition of waste glass on the leachability, the leaching resistance of selected crystalline phases, and a mathematical model of leaching of glass by water. The secondary topics cover heat transfer in internally finned waste canisters, the thermal expansion of simulated PW-4b waste glass, and waste vaporization studies. Another group of topics covers impact testing of encapsulated waste simulant, cement composition as a variable in hot pressing, the strength of hot pressed specimens, and the strength of specimens heated for prolonged periods. Also covered are the gas generation from calcined waste due to radiolysis and the coating of waste calcine with pyrolytic carbon.

LEACHING

III-62

12.091

McElroy, J.L., "Quarterly Progress Report, Research and Development Activities, Waste Fixation Program: April to June, 1975", Battelle Pacific Northwest Laboratories, BNWL-1932, (September 1975), 96 p.

The report contains a section "Leach Rates of Melt 72-68 Containing Cm-244", by F. P. Roberts, pp 22-24. Seven specimens containing 1 weight percent curium oxide and one button containing no curium were subjected to a standard leach test. The doped specimens were stored at room temperature before 20-week leach testing. The leachant was analyzed for potassium by flame photometry and for curium-244 by total alpha counting.

12.092

McElroy, J.L., (ed), "Quarterly Progress Report Research and Development Activities - Waste Fixation Program, October through December 1975", Battelle Pacific Northwest Laboratories, BNWL-1994, (April 1976).

Soxhlet leach tests were made of glass samples in long-term storage at elevated temperature. The only time-dependent changes appear to be an increase in leach rate of samples stored at 700°C and a decrease in those stored at 900°C. Ninety-six super calcine discs, made by cold pressing followed by sintering, were tested to measure crush strength, density and leach rate. Discs formed at 45,500 psi and sintered at 1100°C display densities and leach rates near those of zinc borosilicate glass waste and contain approximately twice as much waste oxides.

12.093

McElroy, J.L., "Quarterly Progress Report, Research and Development Activities, Waste Fixation Program, July through September 1976", Battelle Pacific Northwest Laboratories, BNWL-2243, (August 1977).

III-63

This progress report covers the leach tests on irradiated LWR fuel pellets, on actual HLW glass samples, and on 72-68 glass stored at elevated temperatures for 2 months. An outline of standardized test procedures for impact strength and leach rates, thermal and mechanical shock, and thermal expansion are discussed. The thermal expansion tests consider the effects of incomplete annealing of 72-68 glass, and its effect on the thermal stress in both the glass and the canister. Alternate waste fixation processes were considered including coatings, supercalcine, metal matrix, glass marble, and other multibarrier forms.

12.094

McElroy, J.L., (ed), "R&D Activities Waste Fixation Program. Quarterly Progress Report, January-March 1977, Battelle Pacific Northwest Laboratories, PNL-2265-1, (November 1977), 66 p.

Computer modeling of canisters made of 304L and Inconel 601 for use in the full-scale spray calciner/in-can-melting system was performed to provide thermal, corrosion, and strength data. Thermal effects on leach rates of 76-68 glass were investigated. The acid-base durability, leach rates over long time periods, high temperature behavior of 72-68 and 76-68 glass were examined. Impact data are provided for 76-101, ICM-11, and 73-1 waste glasses. Tests on the mechanical strength of ICM-11 are described. Thermal shock data for 73-1 and ICM-11 are given. Neutron radiography tests on borosilicate glass are described. Leach data on irradiated LWR fuel pellets are given.

12.095

McElroy, J.L., "Quarterly Progress Report, Research and Development Activities Waste Fixation Program. April through June 1977", Battelle Pacific Northwest Laboratories, PNL-2265-2, (July 1978).

LEACHING

III-64

This progress report primarily discusses leaching and the effects of temperature on leaching. The long-term leach rates as a function of time and temperature are discussed for 76-68 and 72-68 glass compositions. Leach data for carbon steel, 304L stainless, Inconel 601, 72-68 and 76-68, and borosilicate glass are given, as well as for other materials. Soxhlet leach tests are performed on in-can-melting glass. A computer code was developed to calculate heat flow from chemical-vapor-deposition-coated supercalcine, glass-control supercalcine, and glass marble. Many materials, including aluminum, 316 stainless, lead and copper were examined as possible metal matrix material. Canister dimensions were discussed in regard to the glass composition characteristics.

12.096

McElroy, J.L., "Research and Development Activities, Waste Fixation Program, Quarterly Progress Report October-December, 1977", Battelle Pacific Northwest Laboratories, PNL-2765-4, (1979).

The objective of the Waste Fixation Program is to develop processes for converting HLW to solid forms that are demonstrated to be physically, chemically, and radiologically stable. This report describes progress on developing four multibarrier waste forms. Also the variation of leach rate with pH on four typical waste glasses are given over the pH range of 1 to 13.

12.097

McElroy, J.L., et al., "Research and Development Activities: High Level Waste Immobilization Program, Quarterly Progress Report, April-June 1979", Battelle Pacific Northwest Laboratories, PNL-3050-2, (December 1979).

Two sections of this report provide information and data on leach tests. Various ceramics (99.8 percent Al_2O_3 , 99 percent Al_2O_3 , ZrO_2 , mullite, graphite, vitreous silica, pyroceram 9617, marcor glass, and ceramic) have been statically and dynamically leach tested in demineralized water at 90, 100, 150, and 250°C. Dry tests of the above materials were conducted by packing crushed salt or basalt in sealed silica ampules and holding at 375°C for three to six months. The chemical durability of the ORNL cermet waste form was investigated using the Soxhlet, pH 4, pH 9, IAEA, and autoclave leach tests. The leach rates for cesium and ruthenium are given for various test conditions.

12.098

McVay, G.L., and Buckwalter, C.O., "The Nature of Glass Leaching", Battelle Pacific Northwest Laboratories, Nuclear Technology, Vol. 51, (1980), pp 123-129.

The nature of glass leaching and possible models of glass-water interaction and their limitations are discussed. Leaching in glasses containing radioactive waste and parameters such as flow rate, solubility, temperature, the leach solution, pH, and time that affect leach rates must be incorporated into the model. Tables and figures show the effect of glass composition and gamma radiation on the leach rate.

12.099

Mendel, J.E., and McElroy, J.L., "Waste Solidification Program, Vol. 10, Evaluation of Solidified Waste Products", Battelle Pacific Northwest Laboratories, BNWL-1666, (July 1972), 134 p.

The leachability of phosphate ceramics and glasses and also borosilicate glass at 25°C was studied. A comparison of leachability of sodium and samarium in WSEP solidified waste products is given.

12.100

Mendel, J.E., "A Review of Leaching Test Methods and the Leachability of Various Solids Media Containing Radioactive Wastes", Battelle Pacific Northwest Laboratories, BNWL-1765, (July 1973), 50 p.

A review of experimental techniques which have been employed to measure the leachability of solidified radioactive wastes in the United States and other countries is made, and factors which affect leachability are described. There is a need for a rapid comparison test and a more comprehensive performance test wherein the actual exposure environment is simulated.

12.101

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

The annual report gives a description of leach rates, melting temperatures, and compositions of a number of different waste glasses. Data are provided on the actual compositions of the glass. Radiation dose rate data for HLW and HLW glass are provided. Temperature effects on the HLW and the canister are described. Data on the thermal power versus time are given. Descriptions of the canister design, closure, inspection, decontamination, and the repository are given. Data on the impact strength of the glass, the thermal shock of the glass, and corrosion of 304L stainless steel with 72-68 glass are given. Results of leach rate tests for both reference glass and actual radioactive samples are given. The acid-base, Soxhlet, Paige and IAEA leach test methods are described.

12.102

Mendel, J.E., et al., "Physical Properties of Waste Glass", Battelle Pacific Northwest Laboratories, High-Level Radioactive Solid Waste, Casey, L.A., (ed), Denver, Colorado, December 19-21, 1978, (CONF-781241), NUREG/CP-0005, (April 1979), pp 37-79.

Hydrothermal leaching of HLW borosilicate glass in salt brine at 350°C is discussed. A typical HLW package is illustrated giving heat generation rate, cumulative alpha dose/gm, and dose rate 1 ft from the surface for one to 100,000 years. Methods for keeping the waste dry and controlling the temperature are reviewed.

12.103

Mendel, J.E., (ed), "Materials Characterization Center Workshop on Leaching of Radioactive Waste Forms, Summary Report", Battelle Pacific Northwest Laboratories, PNL-3318, (April 1980).

This report is a summary of a Materials Characterization Center (MCC) workshop on the leaching of radioactive waste forms. The workshop was divided into five working groups. The first group, Waste Management Criteria, assessed the ability of the MCC leach test development plans to provide the data required to comply with the applicable criteria currently proposed by the NRC, NAS, EPA, WIPP, and DOE/ONWI. The second group, Scoping Tests, assessed the ability of the MCC leach tests to satisfy the needs of the waste form developer for simple, rapid, screening tests. The third group, Data Base for Modeling, examined why the large amount of collected leach data is not being used by mathematical modelers and to recommend methods to establish compatibility between the experimental leach test data base and mathematical models. The fourth group, Waste Controlling Mechanisms, examined what type of information is needed to obtain leach data that fully account for the effects of

basic leach controlling mechanisms. The fifth group, Data Reliability, examined the factors involved in providing confidence in the accuracy and reproducibility of leach test data and recommended techniques that can be used by the MCC to document leach test reliability. Five leach tests are proposed and described which are intended to sufficiently characterize the leaching characteristics of a waste form for isolation purposes.

12.104

Mendel, J.E., et al., "Physical Properties of Glass for Immobilization of High Level Radioactive Waste", Battelle Pacific Northwest Laboratories, Nuclear and Chemical Waste Management, Vol. 1, No. 1, (1980), pp 17-28.

A general overview of the suitability of borosilicate glass for the immobilization of radioactive waste is presented. The ranges of chemical composition for commercial HLW, defense HLW, and incinerator ash are presented, as are the heat generation rate, cumulative alpha dose rate, and biological dose rate 1 foot from a typical HLW canister. Thermal stability (devitrification) and its effect on leach rate as a function of storage temperature is discussed for glass types 72-68 and 76-68. A summary of radiation effects on waste includes stored energy, density change, leach rate, and transmutation. Leach rates are presented for temperatures between 25 and 350°C in deionized water and salt brine.

12.105

Merritt, W.F., "High-Level Waste Glass: Field Leach Test", Chalk River Nuclear Laboratories, Nuclear Technology, Vol. 32, (January 1977), pp 88-91.

Glass blocks combining 85 percent nepheline syenite and 15 percent lime with the nuclear waste solution from fuel reprocessing were prepared in a ceramic crucible. The resulting gel was dried, denitrated at 900°C and melted

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at 1350°C to form a glass. Volatile components (mainly ruthenium and cesium) were absorbed on a heated bed of fire bricks and iron oxides. Field leach tests were conducted on the glass blocks which were buried below the water table in the waste management area of CRNL.

12.106

Merritt, W.F., "The Leaching of Radioactivity from Highly Radioactive Glass Blocks Buried Below the Water Table: Fifteen Years of Results", Chalk River Nuclear Laboratories, Management of Radioactive Wastes from the Nuclear Fuel Cycle, Vol. 2, (CONF-760310), STI/PUB-433, International Atomic Energy Agency, Vienna, (1976), pp 27-35.

Results from two test burials of HLW incorporated into nepheline syenite glass are reported. The leach rates are given for strontium-90 leaving the glass blocks and of the glass blocks themselves over a 15 year test period.

12.107

Mukerji, J., "Disposal of Radioactive Wastes by Vitrification", Central Glass and Ceramic Research Inst., Science Reporter (New Delhi), Vol. 16, No. 6, (June 1979), pp 371-378.

Disposal experts believe the best way to permanently dispose of HLW will be to immobilize the fission products by incorporation in a solid matrix, preferably glass. Advantages of use of a glass matrix are given: (1) glass is a universal solvent and can accommodate various elements in its structure; (2) the fission products become part of the structural block of glass; (3) fixation in glass is essentially irreversible; and (4) glass has very high leach resistance. The glass composition developed at Central Glass and

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Ceramic Research Institute and used at Bhabha Atomic Research Center, India, by pot melting was suitable for processing and for corrosion characteristics, and the plant at Tarapur will use this operation.

12.108

Norris, A.E., "Fission-Product Release", Los Alamos Scientific Laboratory, National Waste Terminal Storage Program Information Meeting, Columbus, Ohio, October 30-November 1, 1979, ONWI-62, pp 42-44.

Leaching experiments have been performed on spent fuel. Four figures of data are given which record the leach rates of strontium-90, antimony-125, cesium-137, curium-244, cerium-144, and americium-241. The leach experiments were conducted at temperatures of 25 and 70°C for approximately 100 days.

12.109

Paige, B.E., "Leachability of Alumina Calcine Produced in the Idaho Waste Calcining Aluminum Waste", Phillips Petroleum Co., IDO-14672, (February 1966), 32 p.

Alumina-phosphate glass was prepared from full-level radioactive calcine produced in the Waste Calcining Facility at the Idaho Chemical Processing Plant. The glass was prepared from 25 weight percent alumina calcine, 40.5 percent P_2O_5 , and 18.6 percent Na_2O (added as $NaH_2PO_4 \cdot H_2O$) and 15.9 percent PbO . Leaching data obtained for fission products and the glass components are compared for the glass in distilled water at 25 and 94°C. Devitrification by heating for 30 days at 450°C was found to have no effect on the leaching rate. Relative leaching rates are given for aluminum, cesium, and strontium at 25°C. Approximately 95 percent of the cesium and 35 percent of the strontium was leached in seven weeks.

12.110

Paige, B.E., "Leachability of Alumina Calcine Produced in the Idaho Waste Calcining Facility", Phillips Petroleum Co., IN-1011, (July 1966), 26 p.

Leaching rates for cesium, strontium, cerium, and aluminum from amorphous alumina calcine produced in the Waste Calcining Facility of the Idaho Chemical Processing Plant have been determined. Rates were obtained for cesium, strontium, and aluminum with distilled water at 25°C and 80 to 90°C with dilute nitric acid at 25°C. The amount of calcine dissolved in water is relatively low; therefore, the high leaching rates of cesium and strontium make recovery of these fission products from calcine potentially feasible.

12.111

Pines, E., (ed), "Report to the American Physical Society by the Study Group on Nuclear Fuel Cycles and Waste Management", Reviews of Modern Physics, Vol. 50, No. 1, Part 2, (January 1978), S1-S185.

The difficulty of extrapolation of leachability tests is illustrated by listing some of the sources of complexity and uncertainty that arise. The section on waste form durability appears on pages S129-S130.

12.112

Platt, A.M., "Research and Development Activities-Fixation of Radioactive Residues, Quarterly Progress Report January-March 1965", Battelle Pacific Northwest Laboratories, BNWL-76, (April 1965), 43 p.

The leachability of high-level phosphate glass was measured using double distilled water and pH 7.8 groundwater. Solubility measurements on the phosphate mixtures of synthetic fission product and process oxides containing 3, 10, and 50 percent fission products were also completed.

12.113

Platt, A.M., "Research and Development Activities-Fixation of Radioactive Residues, Quarterly Progress Report April-June 1965", Battelle Pacific Northwest Laboratories, BNWL-139, (September 1965), 36 p.

The leachability of phosphate glass produced from Purex waste in the Hot Cell Glass Experiment over a seven month period was measured. During this period the devitrification rate of the waste glass was also examined.

12.114

Platt, A.M., et al., "Nuclear Waste Management, Quarterly Progress Report, October through December 1979", Battelle Pacific Northwest Laboratories, PNL-3000-4, (April 1980).

The report describes the leaching test done on three HLW glasses, Supercalcine and SYNROC at 250°C. Corrosion of container material in 250°C Hanford groundwater is described. The use of polymers to inhibit corrosion is discussed. Samples of the canister materials were coated with polymers, irradiated with gamma rays, and then subjected to tensile, impact, and elongation tests. Tests were done on seven alternate waste forms to determine which showed the best overall properties. Alpha radiation had no effect on the integrity of the waste form. Waste package interactions, leaching, and mechanical strength were discussed. The water absorption of Bentonite clay is discussed. Electrochemical oxidation, leaching, and solubility of the waste form is covered.

12.115

Platt, A.M., and McElroy, J.L., "Management of High-Level Nuclear Wastes", Battelle Pacific Northwest Laboratories, PNL-SA-7072, (September 1979).

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This report presents an overview of the present domestic nuclear policies and programs. The United States options for encapsulating spent fuel are described, including canisters with inert gas fill as the baseline option. Also described are the metal matrix, sand fill, glass, ceramic materials, and multiple barrier encapsulation of spent fuel. The composition of 76-68 borosilicate waste glass is compared to pyrex glass. The time, temperature, and radiation effects on mechanical strength and leachability are discussed. Corrosion resistance of the waste form and of lead and titanium is discussed.

12.116

Plodinec, M.J., et al., "Evaluation of Glass as a Matrix for Solidifying Savannah River Plant Waste: Properties of Glasses Containing Li_2O ", Savannah River Laboratory, DP-1498, (February 1979).

Properties of waste forms made of borosilicate glass containing Li_2O combined with both simulated and actual SRP waste sludges were studied. Frit 21 is presently the leading candidate for vitrifying SRP waste in a proposed process for long-term waste management. The effects of temperature and devitrification on the leach rate were correlated with the leach rate of waste forms at room temperature.

12.117

Plodinec, M.J., "Improved Glass Compositions for Immobilization of SRP Waste", Savannah River Laboratory, Scientific Basis for Nuclear Waste Management, Vol. 2, Northrup, C.J.M., Jr., (ed), Plenum Press, New York, (1980), pp 223-230.

Improved glass compositions for the immobilization of radioactive waste produced from reprocessing nuclear materials for defense programs at Savannah River Plant have been developed. Studies of the leachability and

solubility of various elements and element groups were performed by adjusting the amounts of B_2O_3 , TiO_2 , ZrO_2 , La_2O_3 , MgO , and CaO . Further tests were then performed on improved frits. These tests included leach rate at pH 4.4 and pH 11.5, waste solubility at $1150^\circ C$, melting rate, and tendency to devitrify.

12.118

Rankin, W.N., and Kelley, J.A., "Microstructures and Leachability of Vitrified Radioactive Wastes", Savannah River Laboratory, DP-MS-77-69, (1977).

To determine leachabilities of heated and unheated glasses, two types of tests were performed. The first consisted of exposing the glass to water at ambient temperature for 84 to 100 days to measure leach rates of cesium and strontium. The second type of test was an "accelerated" test to correlate microstructure and leachability. Specimens of vitrified waste were examined by metallography and scanning electron microscopy before and after immersion in boiling water for 24 to 96 hrs.

12.119

Rankin, W.N., and Kelley, J.A., "Microstructures and Leachability of Vitrified Radioactive Wastes", Savannah River Laboratory, Nuclear Technology, Vol. 41, (1978), pp 373-380.

This article discusses the leachability of vitrified radioactive wastes correlated with the microstructure before and after devitrification. Metal oxides precipitate in the glass matrix of some compositions of vitrified waste, and these precipitates containing part of the cesium and strontium nuclides are more resistant to leaching by water than the as-cast glass. Leachability of as-cast glasses was from 10^{-8} to 10^{-7} g/cm²/day. Prior heating to $600^\circ C$ for 1 month increased leachabilities to 10^{-6} g/cm²/day.

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Devitrified glass can have up to 100 times greater leachability than as-cast glass. The metal oxide precipitates are less leachable in water than any phase of the devitrified matrix.

12.120

Reeve, K.D., et al., "Progress in SYNROC Research and Development at Lucas Heights", Australian Atomic Energy Commission Research Establishment, Waste Management '80, Post, R.G., (ed), University of Arizona, Tucson, (1980), pp 437-450.

The superior leach resistance of simulated radwaste-doped SYNROC compared with borosilicate waste glass, particularly under hydrothermal conditions is confirmed. The specimens contained 10 weight percent simulated waste. Soxhlet tests in which the waste was contacted with refluxing water at 99°C for 72 hours and hydrothermal tests using deionized water are discussed.

12.121

Ringwood, A.E., et al., "Immobilization of High-Level Wastes in SYNROC Titanate Ceramic", Australian National University, Ceramics in Nuclear Waste Management, Chikalla, T.D., and Mendal, J.E., (eds), Cincinnati, Ohio, April 30 - May 2, 1979, (CONF-790420), (May 1979), pp 174-178.

Accelerated leaching tests in water and saline solutions show that SYNROC is unaffected after 24 hrs at 900°C and 5000 bars, whereas borosilicate glasses decompose within hours at 350°C and 1000 bars. The document also indicates that SYNROC may be particularly well suited for immobilization of United States military waste.

12.122

Ringwood, A.E., et al., "The SYNROC Process: A Geochemical Approach to Nuclear Waste Immobilization", Australian National University, Geochemical Journal, Vol. 13, No. 4, (August 1979), pp 141-165.

Accelerated leaching tests on SYNROC and glasses are described in this paper. Samples had a diameter of 2 to 5 mm. These were sealed in silver tubes, together with 12 times their volume of water or 10 wt percent NaCl solution. Temperatures ranged from 350 to 400°C at 1,000 bars for 24 hrs. Two variants of SYNROC were tested at 600°C for 24 hrs. A series of experiments was also carried out in sealed platinum tubes with internally heated high pressure apparatus at 900 and 1,000°C at 5kb for 24 hrs.

12.123

Ross, W.A., "Properties and Characteristics of High-Level Waste Glass", Battelle Pacific Northwest Laboratories, Properties and Characteristics of High-Level Waste Glass, January 4-5, 1977, Washington, D.C., (CONF-770159), (BNWL-SA-6146), (January 1977), 25 p.

Corrosion rates in mm/yr, with melt temperatures of 900 to 1150°C, of stainless steel by a 72-68 zinc borosilicate are given. Bulk leach rates, based on cesium, of the same glass for 1 to 150 days are reported.

12.124

Ross, W.A., "Development of Glass Formulations Containing High-Level Nuclear Wastes", Battelle Pacific Northwest Laboratories, PNL-2481, (February 1978), 28 p.

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PNL type 76-375 glass was leach tested to determine the effect of waste oxide content on the leachability of the final product. The waste oxide content, which was varied, was K_2O , B_2O_3 , TiO_2 , CaO , and ZnO . The leach tests were performed at a pH ranging from 4 to 9.

12.125

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

Two waste glass compositions (PW-7c and PW-9) were characterized, and the results are presented. The study dealt primarily with leaching effects, but impact and strength tests, thermal conductivity tests, density measurement, glass composition, air volatility, and accelerated radiation effects are also presented.

12.126

Ross, W.A., "Characteristics of High-Level Radioactive Waste Glasses", Battelle Pacific Northwest Laboratories, Transactions of the American Nuclear Society, Vol. 28, (June 1978), pp 357-358.

Because of its properties, glass is considered an excellent material for solidification of radioactive waste. The mechanisms of dispersal of radioactivity are leaching with radionuclide migration, dispersal of particles by air, and vaporization. These mechanisms are discussed in light of the properties of glass. Leaching is increased if the glass form is exposed to high temperatures (650 to 900°C). Dispersal by air may be from mechanical impact, thermal shock, or radiation effects on the glass. Volatilization would most probably occur during the incorporation of wastes into the glass. The use of canisters to contain the waste glass further reduces the possibility of radioactive releases to the environment.

12.127

Roy, D.M., "Cement Based Nuclear Waste Solidification Forms", Pennsylvania State University, High-Level Radioactive Solid Waste Forms, Casey, L.A. (ed), Denver, Colorado, December 19-21, 1978, (CONF-781241), NUREG/CP-0005, (April 1979), pp 479-515.

Previous and current work on the physical, mechanical, and thermal properties; stability; and solubility of waste forms relevant to their immobilization of radionuclides is discussed. Leachability rates for cesium-137, strontium-90, and alpha emitters are given for general and specific radioactive concrete waste forms.

12.128

Roy, D.M., et al., "Low Temperature Ceramic Waste Form: Characterization of Monazite Cement Composites", Pennsylvania State University, Scientific Basis for Nuclear Waste Management, Vol. 2, Northrup, C.J.M., Jr., (ed), Plenum Press, New York, (1980), pp 343-350.

Experiments were conducted on the solidification of monazite, in which neodymium was used to simulate rare earths and actinides in a cementitious matrix. No interactions across grain boundaries were observed, but some exchange might happen with the minor amount of calcium phosphate present in the cement with prolonged curing at elevated temperature and pressure. No detectable neodymium or phosphorous was found in the solution analyses for the experiments after a severe accelerated leach test at 473°K for 28 days.

12.129

Rusin, J.M., et al., "Characterization and Evaluation of Multibarrier Nuclear Waste Forms", Battelle Pacific Northwest Laboratories, Ceramics in Nuclear

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Waste Management, Chikalla, T.D., and Mendel, J.E., (eds), Cincinnati, Ohio, April 30 - May 2, 1979, (PNL-SA-7599), CONF-790420, (1979) pp 66-72.

Four multibarrier concepts using coatings and metal matrices were developed and demonstrated on a one-liter scale. The effects of volatility, leachability, and impact resistance on product integrity are discussed. Results of radiation stability tests are also included.

12.130

Sansel, E.G., and Berreth, J.R., "Preparation and Characterization of Sintered Glass-Ceramics from Calcined Simulated High-Level Waste", Idaho National Engineering Laboratory, Nuclear Technology, Vol. 33, (April 1977), pp 68-75.

This paper reports the development and characterization of sintered glass-ceramics as a monolithic waste form for disposal of HLW. Sintered glass-ceramics have durability and properties similar to ceramics and glass-ceramics. Compacted mixtures of calcine are sintered with aluminosilicate fluxes at temperatures ranging from 965 to 1070°C. Products are resistant to impact and thermal shock and exhibit leach rates of 10^{-5} to 10^{-9} g/cm²/day. Waste volumes as low as 0.040 m³/MTU reprocessed are given. Storage centerline temperatures up to 800°C appear feasible.

12.131

Scheetz, B.E., et al., "Hydrothermal Interaction of Simulated Nuclear Waste Glass in the Presence of Basalts", Pennsylvania State University, Scientific Basis for Nuclear Waste Management, Vol. 2, Northrup, C.J.M., Jr., (ed), Plenum Press, New York, (1980), pp 207-214.

This paper discusses the effect of the presence or absence of basalt on the solution properties of cesium, barium, calcium, strontium, sodium, and silicon contained in 76-68 waste glass at up to 300°C and 300 bars pressure. The principal result is that basalt has a marked influence on the hydrothermal alteration behavior of the glass waste form.

12.132

Schneider, K.J., (ed), "Research and Development Activities Fixation of Radioactive Residues Quarterly Progress Report February-April 1971", Battelle Pacific Northwest Laboratories, BNWL-1578, (May 1971), 22 p.

This progress report is the forty-ninth in a series presenting research and development activities in the field of radioactive wastes. Nonradioactive leach rate measurements were made on borosilicate glasses at 100°C and 600°C. Glasses studied included those made by the rising level and in-pot melting processes. The differential leach rates of sodium and strontium from the glasses were measured.

12.133

Schuman, R.P., "Leach Testing of Idaho Chemical Processing Plant Final Waste Forms", Exxon Nuclear Idaho Co., Inc., ENICO-1064, (October 1980), 39 p.

Pellets and highly durable glasses prepared from nonradioactive-simulated HLW calcines have been leach tested, patterned on the IAEA standard test and the proposed Materials Characterization Center tests. Most tests are made with static distilled water at 25, 70, 95, and 350°C and in refluxing distilled water, Soxhlet, at 95°C. Leach rates are determined by analyzing the leach rate by instrumental activation analysis or spectrochemical analysis

and from weight loss. Leach tests have been run on glass using cast and core drilled cylinders, broken pieces, and coarse ground material.

12.134

Schumann, R.P., "Determination of Nuclear Waste-Form Leach Rates by Activation Analysis", Exxon Nuclear Idaho Co., Inc., Transactions of the American Nuclear Society, Vol. 33, (1979), pp 422-423.

The leach rates from zirconia calcine and lanthanide-actinide calcine have been determined using neutron activation analysis. Test results are tabulated for the two calcines from two separate tests on each calcine: static distilled water at 25°C and distilled water in a Soxhlet extractor at 95°C.

12.135

Seitz, M.G., et al., "Leach-Migration Experiments to Determine the Mobility of Radionuclides in Geologic Media", Argonne National Laboratory, Transactions of the American Nuclear Society, Vol. 32, (1979), pp 385-386.

In previous experimental work to evaluate the safety of a waste repository, the movement of radionuclides from a repository has been studied using radionuclide leaching and radionuclide migration experiments. These tests are usually conducted separately. In fact, migration behavior is dependent on solution composition which is altered by leaching. Therefore, this present study presents results of experiments which consider these two processes simultaneously. Results from several experiments in the leaching of simulated HLW glass by warm groundwater are presented.

12.136

Shade, J.W., "An Approach to the Prediction of Waste-Form Stability", Atlantic Richfield Hanford Co., ARH-ST-105, (October 1974).

Calculated activity diagrams are suggested as an aid in predicting the long-term stability of aluminosilicate types of nuclear waste forms. Possible disposal sites can be characterized in terms of mineral-field equilibrium and weathering reaction of a given waste form estimated for that site. Chemical thermodynamic calculations of hypothetical solubility reactions for crystalline feldspar and glasses of the same composition are given for comparison.

12.137

Simmons, J.H. et al., "Chemical Durability of Nuclear Waste Glasses", Catholic University of America, Ceramics in Nuclear Waste Management, Chikalla, T.D., and Mendel, J.E., (eds), Cincinnati, Ohio, April 30 - May 2, 1979, CONF-790420, (1979), pp 263-268.

Test methods are presented for the measurement of chemical durability of glass media proposed for nuclear waste fixation. The release rates of individual components are measured to determine matrix dissolution rates, possible transport of components through the matrix, and corrosion mechanisms. Measurement on a model glass, a borosilicate glass, and a high-silica glass fixation medium are reported. The extrapolation of short-term laboratory tests to long-term storage is discussed, and the essential parameters for accurate long-term predictions are determined and evaluated.

12.138

Simmons, J.H., "Fixation of Radioactive Waste in High Silica Glasses", Catholic University of America, Nature, Vol. 278, (April 19, 1979), pp 729-731.

A multibarrier vitrification process carried out below 900°C accomplishes the fixation of HLW as oxides of a high-silica matrix. These high-silica glasses have good chemical stability and a lower leaching rate than other fixation materials. The physical properties required of fixation materials, chemical properties of various glasses, and fixation in different types of glass are discussed. Cesium oxide as a dopant and the problems it poses are described. The excellent chemical stability, high thermal and structural stability, and the possibility of a multibarrier effect of this high-silica glass matrix over other glass encapsulation processes are shown.

12.139

Steindler, M.J., et al., "Chemical Engineering Division, Fuel Cycle Programs Report January-September 1977", Argonne National Laboratory, ANL-78-11, (1977), 168 p.

Results of numerous different fuel cycle studies for the period January-September, 1977 are presented. Of interest are data on the leach rates of three different waste forms: (1) spray calcine; 10 weight percent silica, 10 weight percent glass frit, (2) spray calcine; 15 weight percent glass frit, and (3) several metal-matrix materials.

12.140

Steindler, M.J., et al., "Chemical Engineering Division, Fuel Cycle Programs Progress Report January-March 1978", Argonne National Laboratory, ANL-78-68.

Results of numerous different fuel cycle studies during the period January-March, 1978 are reported. Of greatest interest to the waste package will be leach rate data on metal-encapsulated waste forms on pp 153-159 and corrosion studies on matrix metal or canister materials by brines under hydrothermal conditions reported on pp 159-161.

12.141

Strathdee, G.G., et al., "Development of Aluminosilicate and Borosilicate Glasses as Matrices for Candu High-Level Waste", Battelle Pacific Northwest Laboratories, Ceramics In Nuclear Waste Management, Chikalla, T.D., and Mendel, J.E., (eds), Cincinnati, Ohio, April 30 - May 2, 1979, CONF-790420, (1979), pp 243-247.

Two radioactive nepheline syenite glass blocks were recovered after 18 and 20 yrs of in-ground leaching experiments at Chalk River Nuclear Laboratories. Virtually no change has occurred to the glass surfaces exposed to flowing groundwater. Gamma spectrometry, scanning electron microscopy, X-ray photoelectron spectroscopy, and alpha particle-track autoradiography were used for analyzing the interior of the blocks.

12.142

Strathdee, G.G., et al., "Fuel and Waste Immobilization in Canada Related to WISAP Task 2", Task 2 Workshop Waste Isolation Safety Assessment Program, Bradley, D.J., (ed), PNL-SA-7992, (1979), pp 181-204.

Two static leaching experiments of CANDU fuel using Zircaloy-sheathed sections of the Pickering reactor fuel (burnup 610 GJ/kgU) were performed for 900 days. The leach rates of all analyzed radionuclides decreased with time. Results for cesium-137, strontium-90, cerium-144, and plutonium are figured for both air-saturated distilled and tap water. Irradiated-fuel leach rates

at 150°C for cesium-137 are given for less than 100 days. The standard leach tests normally used to provide comparisons of glass product durabilities are expected to yield leach rate values which are conservative compared to the leach rates expected in an actual repository environment.

12.143

Strathdee, G.G., (ed), "Second Annual Report on the Evaluation of Immobilized High Level Waste Products", Atomic Energy of Canada, Ltd., WNRE-466, (December 1979).

Borosilicate and aluminosilicate glasses are under consideration in Canada for solidification of waste from reprocessing of CANDU UO₂ or advanced thorium cycle fuel. The projected compositions of various waste forms are documented. The effect of phase separation, devitrification, and anhydrous crystallization of glasses is presented. Physical properties such as glass viscosity, thermal conductivity, and electrical and thermal conductivity are also briefly discussed.

The majority of the report discusses the leaching of radioactive and simulated waste products. The results of three tests; the draft international standards organization test for radioactive waste solidification products, a flow through test, and an in-situ test are discussed at length. Results are presented for leach rate tests at temperatures up to 150°C with different samples under varying conditions. A comparison is made between measured and model predictions for strontium-90 concentration in water for the in-situ test.

12.144

Svensson, B.M., and Dahl, L., "Leaching of Al₂O₃ Under Simulated Repository Conditions", KBS Technical Reports 1-120, Report No. 116, (1977-1978).

Aluminum material has been leached at 90°C: (1) in simulated ground-water at pH 8.5; (2) embedded in bentonite and silica sand saturated with the same water; and (3) in simulated groundwater at pH 6 and 10. Leaching periods varied from 30 to 300 days.

12.145

Svensson, B.M., and Blomqvist, G., "Leaching of Al_2O_3 in Twice-Distilled Water", KBS Technical Reports 1-120, Report No. 117, (1977-1978).

Al_2O_3 material was leached at 90°C in pure double-distilled water and in double-distilled water buffered to pH 9.3 with sodium carbonate. The corrosion rate of Al_2O_3 was calculated from the aluminum content of the leaching solution.

12.146

Terai, R., et al., "Characterization of Borosilicate Glasses Containing Simulated High-Level Radioactive Wastes from PNC", Government Industrial Research Inst., Ceramics in Nuclear Waste Management, Chikalla, T.D., and Mendel, J.E., (eds), Cincinnati, Ohio, April 30 - May 2, 1979, CONF-790420, (1979), pp 62-65, (also in Bulletin of the American Ceramic Society, Vol. 58, No. 3, (1979), p 320).

The chemical composition and physical properties of eight typical borosilicate glasses containing simulated HLW are presented. Leachability was determined by the Soxhlet leach test and by the continuous leaching method.

12.147

Terai, R., (ed), "Chemical Durability of Glasses Suitable for the Storage of High Level Radioactive Wastes. I. On the Mechanism of Dissolution of the

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Glasses of Borax-Alumina-Silica System in Water", Government Industrial Research Inst., ORNL-TR-4626, (1975), 22 p.

The leach rates of various borosilicate glasses were measured. Tests were carried out using three procedures for a period of one week. The selective leaching of the boron and sodium as opposed to the silica and aluminum components of the glass was also studied.

12.148

Terai, R., (ed), "Volatilization of Low Temperature Borosilicate Glasses for High-Level Radioactive Wastes at Elevated Temperatures", Government Industrial Research Inst., ORNL-TR-4629, (1976), 28 p.

The leach rates of glasses spanning a range of chemical compositions in the borax-alumina-silica system were studied. The selective leaching of certain chemical components of the waste glass is discussed.

12.149

Turcotte, R.P., "Release Model Development", Battelle Pacific Northwest Laboratories, Task 2 Workshop Waste Isolation Safety Assessment Program, Bradley, D.J., (ed), PNL-SA-7992, (1979), pp 73-81.

Leach rates are given for cesium, rubidium, strontium, and molybdenum from solid and powdered samples of 76-68 glass at 350°C using deionized saltwater or salt brine. Tests were conducted from one to 100 days.

12.150

Turcotte, R.P., et al., "Devitrification of Nuclear Waste Glasses", Scientific Basis for Nuclear Waste Management, Vol. 2, Northrup, C.J.M., Jr., (ed), Plenum Press, New York, (1980), pp 141-146.

Devitrification studies of waste glasses are reported to determine the processing conditions through which significant crystallization can be avoided. The compositions of various glasses examined have been reported elsewhere as follows: glass 72-68, glass 76-68, glass 77-260, glass SRL-C-411, glass SRL-T-211. Crystallization trends are established by isothermal and constant cooling rate experiments, the latter results reported here. Figures of the in growth of ZnSiO_4 in glass 72-68 at 750°, 800° and 850°C vs. time, and percent crystallinity as a function of temperature are given. Overall, the effect of devitrification on leach rate appears to be minimal.

12.151

Upson, U.L., "Observed Properties of Some Solidified High Level Wastes and Their Stability Under Simulated Storage Conditions", Solidification and Long-Term Storage of Highly Radioactive Waste, Richland, Washington, February 14-18, 1966, CONF-660208, (April 1967), pp 642-659.

Selected calcines produced by spray and pot calcination of high-level purex and redox process wastes were examined for density, leachability and visual appearance, and sealed pots containing calcines were monitored for pressure buildup. A sample of phosphate glass produced from the high-level Purex waste was examined for mechanical stability and for leachability over a one year period. Volume reduction factors and leach rate data are given.

12.152

Vandergraaf, T.T., et al., "Leaching of Irradiated Candu UO_2 Fuel", Whiteshell Nuclear Research Establishment, Scientific Basis for Nuclear Waste Management, Vol. 2, Northrup, C.J.M., Jr., (ed), Plenum Press, New York, (1980), pp 335-342.

Irradiated fuel, leached at room temperature with distilled water and with slightly chlorinated river water, releases about 4 percent of its cesium inventory over a comparatively short period of a few days. The actinides and rare earths are released more slowly. The matrix dissolves at about 2×10^{-6} g $UO_2/cm^2/day$.

12.153

Veal, B.W., et al., "X-Ray Photoelectron Spectroscopy Studies of Silicate Glasses: Implications to Bonding and Leaching", Argonne National Laboratory, Nuclear Technology, Vol. 51, (1980), pp 136-142.

Because of the 1,000 or more years nuclear waste products must be immobilized, it is necessary to the waste system design to investigate the bonding and leaching mechanisms of complex silicate glasses that may be used. This paper discusses the use of X-ray photoelectron spectroscopy to study bonding properties and leaching effects of glasses and provide an easy measurement of relatively elemental concentrations on sample surfaces. Monitoring oxygen-to-silicon ratios before and after leaching can test dissolution models. The discussion considers the synergistic effect of the addition of CaO to sodium silicate glasses and the leaching of silicate glasses containing uranium. The conclusion is that uranium is leached away in the U^{6+} state at a faster rate than the dissolution rate of silica, and that for UO_2 glasses, U^{4+} to U^{6+} conversion occurs, but at a slower rate than U^{6+} in UO_3 glasses.

12.154

Wald, J.W., and Westsik, J.H., Jr., "Devitrification and Leaching Effects in HLW Glass--Comparison of Simulated and Fully Radioactive Waste Glass", Battelle Pacific Northwest Laboratories, Ceramics in Nuclear Waste Management, Chikalla, T.D., and Mendel, J.E., (eds), Cincinnati, Ohio, April 30 - May 2, 1979, CONF-790420, (1979), pp 277-283.

Devitrification and leaching analyses of four waste glasses were investigated to compare nonradioactive compositions to compositions made using fully radioactive waste calcine. Leach tests were carried out for different durations and at two different temperatures, 25 and 75°C. Also the effect of glass composition on leach rate was studied.

12.155

Walker, C.T., et al., "Chemical Compatibility of HLW Borosilicate Glasses with Actinides, Solubilities, and Segregation Processes", Nuclear Research Center, Karlsruhe, (EUR-5750E), KFK-2552, (November 1978).

The leach rate of actinides from borosilicate glasses in salt brine at 23 and 115°C has been measured.

12.156

Walton, F.B., and Merritt, W.F., "Long Term Extrapolation of Laboratory Glass Leaching Data for the Prediction of Fission Product Release Under Actual Groundwater Conditions", Whiteshell Nuclear Research Establishment, Scientific Basis for Nuclear Waste Management, Vol. 2, Northrup, C.J.M., Jr., (ed), Plenum Press, New York, (1980), pp 155-166.

Release and migration of strontium-90 and cesium-137 from nepheline syenite-based glass buried since 1960 below the water table in sandy soil at CRNL have been predicted by models using laboratory glass leaching data. Glass leach rates of 8×10^{-14} kg/m²/s after exposure to groundwater for 17 years are observed.

12.157

Weed, H.C., et al., "Leaching Characteristics of Actinides from Simulated Reactor Waste Glass", Lawrence Livermore Laboratory, Scientific Basis for Nuclear Waste Management, Vol. 1, McCarthy, G.J., (ed), Plenum Press, New York, (1979), pp 141-147.

Two methods for measuring the leach rates of simulated HLW glass are compared. One is a modification of the standard IAEA method, and the other is a one-pass method in which fresh leachate solution is pumped over the sample at a controlled flow rate and temperature. For a time up to 3 days, there is close agreement between results from the two methods at 25°C.

12.158

Weed, H.C., et al., "Leaching Characteristics of Actinides from Simulated Reactor Waste Glass, Part 2", Scientific Basis for Nuclear Waste Management, Vol. 2, Northrup, C.J.M., Jr., (ed), Plenum Press, New York (1980), pp 167-174.

Leach rates of neptunium-237 and plutonium-239 were investigated with a single-pass leaching system. The experimental design uses several combinations of solution composition and flow rate and two temperatures, 25°C and 75°C. At 25°C leach rates decreased with time using distilled H₂O, NaHCO₃ and WIPP brine leachant. Leach rates are approximately independent of flow

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rate at room temperature, but increase with flow rates at high temperature. Release rates of neptunium-237 increase with temperature, but those of plutonium-239 decrease or do not change with temperature.

12.159

Westsik, J.H., Jr., and Turcotte, R.P., "Hydrothermal Glass Reactions in Salt Brine", Battelle Pacific Northwest Laboratories, Scientific Basis for Nuclear Waste Management, Vol. 1, McCarthy, G.J., (ed), Plenum Press, New York, (1979), pp 341-344.

A simulated HLW glass, a supercalcine, and some common ceramic and metallic solids were exposed to hydrothermal conditions at 350 and 250°C from three days to three weeks. Leachability results are given for six tests of Glass 76-68 and supercalcine SPC-4 at 350°C with brine and deionized water solutions. Results are also tabulated for autoclave leach tests at 250°C of various glasses, granite, UO₂ fuel pellet, and alumina, primarily in brine solution.

12.160

Westsik, J.H., Jr., and Turcotte, R.P., "Hydrothermal Reactions of Nuclear Waste Solids--A Preliminary Study", Battelle Pacific Northwest Laboratories, PNL-2759, (September 1978), 37 p.

Glass samples (76-68) and deionized water were reacted together with combinations of salt and canister material. The salt for the brine came from the 2621 ft level of the site being considered for WIPP. Stainless steel (4.2 g of 100 mesh 304 SS) was added in two tests to determine the influence of a canister material on the hydrothermal-glass reactions. Each test was run at 350°C in an autoclave vessel. The equilibrium operating pressure ranged

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from 1750 to 2510 psi. The glass cylinder was characterized using metallography and scanning electron microscope with X-ray fluorescence capabilities. Leach tests showed percent of cesium, rubidium, strontium, molybdenum, silicon, boron, uranium, and zinc in solution. Results of scouting autoclave leach tests (samples immersed in simulated WIPP brine for 72 hr at 250°C and 1000 psi) are given for various glasses, granite, UO₂ fuel pellet, SC-2, and steels. Effects of temperature on HLW glass leach rates are reported.

12.161

Westsik, J.H., Jr., "Autoclave Release Studies", Battelle Pacific Northwest Laboratories, Task 2 Workshop Waste Isolation Safety Assessment Program, Bradley, D.J., (ed), PNL-SA-7992, (1979), pp 67-71.

Small, 1/3 g glass beads were sealed in gold capsules containing deionized water, a 252 g/l sodium bicarbonate solution, or with WIPP "B" brine. The samples were held at 150 or 250°C and 2000 psi for times of 2, 4, 8, 16, or 32 days. Replication was achieved by running separate tests with undoped beads and with beads doped with technitium-99, curium-244, uranium-233, americium-245, or plutonium-239, and neptunium-237. Weight lost data are given as a function of time. Weight loss in the salt brine is about 2-1/2 times lower at 150°C and 4 times lower at 250°C than in the lower ionic strength solutions. Cesium concentrations in solution are shown as a function of time.

12.162

Westsik, J.H., Jr., et al., "Temperature Dependence for Hydrothermal Reactions of Waste Glasses and Ceramics", Scientific Basis for Nuclear Waste Management, Vol. 2, Northrup, C.J.M., Jr., (ed), Plenum Press, New York, (1980), pp 239-248.

This paper discusses the results of a study to determine the effect of temperature on the leaching behavior of several HLW forms. This study shows that simulated HLW glasses, super calcine and SYNROC exhibit similar leaching behaviors in hydrothermal environments. The ceramics show a smaller temperature dependence than do the glasses. At temperatures above 250°C, the ceramics show lower releases, but below 250°C, where actual repository temperatures are expected to be, releases from waste glasses are lowest.

12.163

Wiley, J.R., and LeRoy, J.H., "Long-Term Leach Rates of Glasses Containing Actual Waste", Savannah River Laboratory, Ceramics in Nuclear Waste Management, Chikalla, T.D., and Mendel, J.E., (eds), Cincinnati, Ohio, April 30 - May 2, 1979, (DP-MS-78-82), CONF-790420, (1979), pp 284-288.

Leach rates of borosilicate glasses that contained actual SRP waste were measured. Leaching was done by water and by buffered solutions of pH 4, 7, and 9. Rates were then determined from the amount of cesium-137, strontium-90, and plutonium released into the leach solutions. The test time varied from 150 to 200 days. The cumulative fractions leached were fit to a mathematical model that included leaching by diffusion and glass dissolution.

12.164

Wiley, J.R., "Leach Rates of High Activity Waste for Borosilicate Glass", Savannah River Laboratory, DP-MS-77-80, (1977).

Borosilicate glasses containing actual, SRP HLW were leached in static, distilled water. Leach rates of strontium-90, cesium-137, and plutonium were determined. Mechanisms of radionuclide leaching from the glass were also examined.

12.165

Wiley, J.R., et al., "Glass as a Matrix for SRP High-Level Defense Waste", Savannah River Laboratory, DPST-79-294, (January 1980), 53 p.

Testing of borosilicate glass formulations for solidifying SRP wastes began in 1973. This report summarizes its development and current activities, especially in leaching and radiation effects. Leach tests were done over a 200-day period at pH of 4, 7, 9, and in distilled water. Soxhlet and brine leaching data are reviewed.

12.166

Wood, B.J., "Estimation of Waste Package Performance Requirements for a Nuclear Waste Repository in Basalt", Rockwell Hanford Operations, National Waste Terminal Storage Program Information Meeting, Columbus, Ohio, December 9-11, 1980, ONWI-212, (December 1980), pp 335-338.

Equations are given for determining maximum permissible release rates in the repository which would keep radionuclide concentrations at the biosphere below the recommended concentration limits at all times. The radionuclides requiring release rates less than 10^{-3} /yr to keep concentrations at the biosphere below recommended concentration values are listed for three different release scenarios. The relative advantages and disadvantages of a 1,000 year as opposed to a 100 year waste package are compared.

12.167

Yen-Bower, E.L., et al., "An Approach Towards Long Term Prediction of Stability of Nuclear Waste Forms", University of Florida, Ceramics in Nuclear Waste Management, Chikalla, T.D., and Mendel, J.E., (eds), Cincinnati, Ohio, April 30 - May 2, 1979, CONF-790420, (1979), pp 41-46.

Durability of glasses which have been proposed as encapsulants for nuclear wastes is discussed in this paper. Short term laboratory experiments were conducted to make predictions concerning the long-term storage stability of the glasses. The experimental method for predicting long-term stability of nuclear waste encapsulants uses various ratios of surface area to the solution volume exposed to the encapsulant. A graphical technique, involving leaching indices and exposure times, illustrates how predictions can be made based on laboratory experiments, field trials, and museum artifacts.

12.168

Zemlyanukhin, V.I., et al., "Clay Phosphate Ceramics and Vitromets: Alternatives to Monolithic High Level Waste Glass Products", V.G. Khlopin Radium Institute, USSR, Scientific Basis for Nuclear Waste Management, Vol. 1, McCarthy, G.J., (ed), Plenum Press, New York, (1979), pp 195-200.

Tabulated are data on temperature dependency of open porosity and room temperature sodium leachability of clay-phosphate ceramics. It is shown that with increasing temperature (from 700 to 950°C), both the open porosity and the sodium leachability decrease. Leachability rates are given in g/cm²/day.

see also

13.006	13.010	13.011
13.014	13.015	13.017
14.001	14.012	14.013
14.014	14.017	14.018
14.019	14.026	14.027
14.030	14.031	14.033
14.045	15.002	15.007
15.012	16.001	16.002
16.009	16.012	16.018

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16.020	16.021	16.022
16.024	16.025	16.026
16.027	16.030	16.031
16.034	16.035	16.037
16.038	16.040	16.041
16.042	16.043	16.044
16.045	16.046	16.047
16.048	16.049	16.050
16.051	16.052	16.055
16.056	16.058	16.059
16.061	16.064	16.065
16.066	16.067	16.070
16.072	16.073	17.003
18.011	18.014	18.016
21.004	21.009	21.011
21.018	21.019	21.022
21.023	22.008	22.015
23.009		

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13. Material Compatibility13.001

Ahlstroem, P.E., "Ceramic and Pure-Metal Canisters in Buffer Materials", Kaernbraenslesaeckerhet, High Level Radioactive Solid Waste Forms, Casey, L.A., (ed), Denver, Colorado, December 19-21, 1978, (CONF-781241), NUREG/CP-0005, (April, 1979), pp 285-316.

Studies have been made on lead-titanium for HLW canisters and copper for spent fuel canisters. The disposal is planned in hard crystalline rock where the waste canisters are embedded in a buffer material (clay). For the lead-titanium canister a mixture of quartz sand and bentonite is proposed, whereas for the copper canister a highly compressed bentonite is preferred. Ceramic canisters of alumina have a very high chemical and mechanical resistance over very long periods of time.

13.002

Anderson, R.N., and Selvaduray, G.S., "The Effect of Pressure on Interface Interactions Between Solids", San Jose State University, High-Level Radioactive Solid Waste Forms, Casey, L.A., (ed), Denver, Colorado, December 19-21, 1978, (CONF-781241), NUREG/CP-0005, (April 1979), pp 317-339.

Reactions occurring at the waste/canister/rock interface, characteristics of the reaction products, kinetics associated with the rate of formation of these products, diffusional transport kinetics across the interfaces, effects of pressure, and effects on the operation of a repository are addressed. Studies were done in the laboratory under pressures equivalent to the rock overburden of a 2000 ft deep repository.

13.003

Cameron, D.J., and Strathdee, G.G., "Materials Aspects of Nuclear Waste Disposal in Canada", *Ceramics in Nuclear Waste Management*, Chikalla, T.D., and Mendel, J.E., (eds), Cincinnati, Ohio, April 30 - May 2, 1979, CONF-790420, (1979), pp 4-8.

An overview of the waste disposal program in Canada from the perspective of material design considerations is presented. Specifics about the CANDU waste form (dimensions, fission product weight percent etc.), initial specifications for the reference package (containers dimensions, waste glass volume/mass, decay heat), engineered containment design considerations (specifically corrosion resistant materials under study), and buffer/backfill materials are discussed. The waste forms under consideration are borosilicate glasses and container materials discussed include 316L stainless steel, Inconel 625, and titanium Grade 2. For all design considerations presented, a total system approach to material selection is being undertaken.

13.004

Ewing, R.C., and Krumhansl, J.S., "Rad. Waste-Form and Rock Interactions: A Discussion", *University of New Mexico, Transactions of the American Nuclear Society*, Vol. 33, (1980), p 280.

This paper reports some of the laboratory and field experiments currently under way to study waste form/rock interactions. Laboratory experimental results are reported on the effect of groundwater and basalt reactions on waste glass. Results are also reported on field tests in Conasuaga shale using a sample of PNL 75-32 waste glass simulant at 100°C. The effects of temperature, water, and radiation on the long-term stability of glass are discussed. Radiation effects are reported based on studies of metamict minerals.

13.005

Feinstein, J.G., et al., "Procedures, Barriers, State Variables, and Processes Important to Near-Field Analysis", NUS Corp., Transactions of the American Nuclear Society, Vo. 34, (1980), pp 403-404.

Boolean algebra techniques are used to develop a procedural matrix for identifying the barriers, state variables, and processes that should be considered in near-field waste-rock interactions.

13.006

Healy, T.V., "Fuel Reprocessing Solvent Tributyl Phosphate, Its Degradation, Clean-Up and Disposal", Harwell Atomic Energy Research Establishment, Management of Radioactive Wastes from the Nuclear Fuel Cycle, Vol. 1, (CONF-760310), STI/PUB-433, International Atomic Energy Agency, Vienna, (1976), pp 201-214.

Solvent degradation studies at Harwell are described. The results may give some insight as to the behavior and persistence of certain radionuclide-complexing materials that might be present in HLW. The discussion relates to the ONWI criteria concerning excluded and undesirable materials in the waste package.

13.007

Koplik, C.M., et al., "Information Base for Waste Repository Design, Vol. 3, Waste/Rock Interactions", Analytic Sciences Corp., NUREG/CR-0495, (1979).

This report summarizes an extensive data base relevant to the disposal of HLW and spent fuel. Factors influencing the thermal effects of the waste on the package and on the repository such as type and age of waste, thermal output per canister, canister spacing and thermal load, and repository ventilation are discussed. Near- and far-field temperatures are estimated. Dose rates are calculated and radiolytic effects in the repository are summarized. Data on the corrosion rates of waste packages in the presence of brines are also presented.

13.008

Krogness, J.C., et al., "Survey Test of Canister, Geology, and Fuel Cladding Material Interactions", Hanford Engineering Development Laboratory, Transactions of the American Nuclear Society, Vol. 34, (1980), pp 426-427.

This paper summarizes the in-progress and planned tests at the Nevada Test Site covering the interactions between package, geology, and waste form materials. Tests will include various types of bedrock (basalt, granite, argillite, welded tuff), several canister materials (austenitic stainless steels, low-carbon steels, nickel base alloys, brass), and three types of capsule interior fill gases (helium, air, air-water). Tests are also planned to separate thermal from radiation effects on the canister.

13.009

McCarthy, G.J., et al., "Crystal Chemistry of the Synthetic Minerals in Current Supercalcine-Ceramics", Pennsylvania State University, Ceramics in Nuclear Waste Management, Chikalla, T.D., and Mendel, J.E., (eds), Cincinnati, Ohio, April 30 - May 2, 1979, CONF-790420, (1979), pp 315-320.

The investigation of the crystal chemistry of the synthetic mineral phase assemblages in the current supercalcine ceramics is presented in this

paper. Of particular interest is the compatibility studies of all phases of the glass at temperatures as high as 800°C. Also discussed is the low leachability, low waste volatilization during processing, and the high waste loading of the ceramics.

13.010

McCarthy, G.J., and Scheetz, B.E., "High-Level Waste Basalt Interactions, Annual Progress Report for Period February 1, 1977 to September 30, 1977", Rockwell Hanford Operations, RHO-BWI-C-2, (May 1978).

This report presents experimental results of hydrothermal interactions of spent fuel, glass, calcine, and stainless steel with basalt under three potential repository conditions. The three repository conditions studied were; dry low pressure, open repository in a retrievable mode of operations, dry sealed repository that has been backfilled and sealed, and a wet sealed repository where after a time groundwater could fill the repository. Mixture of the waste glass and basalt were exposed to a variety of hydrothermal conditions for durations of 30 days. Test variable extremes were 600°C and 400 bars of pressure. A variety of techniques were employed to study the waste glass-basalt interactions and resultant changes such as leaching rate.

13.011

Mendel, J.E., "Glass: An Available Material for the Immobilization of Nuclear Waste", Battelle Pacific Northwest Laboratories, Transactions of the American Nuclear Society, Vol. 33, (November 1979), p 278.

Various aspects of using a glass as a waste form are discussed. The major topics are waste acceptance flexibility, process adaptability to radioactive service, canister interactions, thermal stability, radiation stability, and leach behavior.

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13.012

Rankin, W.N., "Compatibility Testing of Vitrified Waste Forms", Savannah River Laboratory, DP-MS-77-115, (March 1978).

An experimental program to evaluate candidate metals for use in the fabrication of canisters for long-term storage of vitrified radioactive wastes is described. The long-term compatibility of the candidate metal both with the contained vitrified radioactive waste and with the external environments expected in the final disposal location is determined. These tests involve heating combinations of waste forms and canister metals in intimate contact for up to 50,000 hrs to accelerate any reactions that may occur.

13.013

Rusin, J.M., "Development of Multibarrier Nuclear Waste Forms", Battelle Pacific Northwest Laboratory, PNL-SA-7292, (March 1979), 14 p.

Two options were developed for the inner core of the multibarrier concept: supercalcine pellets and glass marbles. The process development and one-liter laboratory-scale demonstration of four multibarrier concepts are described.

13.014

Scheffler, K., et al., "Evaluation of the Borosilicate Glass Matrix for the Immobilization of Actinide Waste Concentrations", Nuclear Research Center, Karlsruhe, Treatment, Conditioning and Storage of Solid Alpha-Bearing Waste and Cladding Hulls, Paris, CONF-771208, (December 1977), pp 269-278.

The experiments described examine the suitability of borosilicate glass as a matrix for the solidification of high-actinide bearing wastes. Accelerated experiments were performed on curium-242 doped HLW glass specimens

to determine long term radiation stability of borosilicate glasses against alpha-emitters. The effects of radiation on the mechanical strength, structure, and homogeneity of the glass were studied. Also borosilicate glass leach rates were measured as a function of solution composition and temperature.

13.015

Smith, M.J., et al., "Engineered Barrier Development for a Nuclear Waste Repository in Basalt: An Integration of Current Knowledge", Rockwell Hanford Operations, RH0-BWI-ST-7, (May 1980).

This is a large document which is a compilation of data and interpretive studies of the engineered barriers program of the Basalt Waste Isolation Project. It includes studies on barrier system designs, emplacement and isolation techniques, and chemical reactions expected in a nuclear waste repository located in the basalts underlying the Hanford Site.

13.016

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

An initial approach to the methodology of selecting materials for a spent fuel package, including the stabilizer, canister, overpack, liner, liner coating, and backfill material, is presented. The function of each package barrier is discussed, as well as the advantages and disadvantages of various proposed barrier materials in satisfying the barrier requirements. Four preliminary "candidate barrier system concepts" are developed within the framework of a set of performance criteria assumed for the analysis.

Conformance to the ONWI Criteria. Some, but not all, of the performance criteria developed for the study are in conformance to the ONWI packaging criteria. The purpose of specifying criteria in the study, however, was only to establish a framework for evaluating the performance of materials. The criteria were not intended as suggestions for final packaging criteria.

13.017

Wurm, K.J., and O'Hara, F.A., "Spent-Fuel Processing and Packaging Options", Battelle Columbus Laboratories, Transactions of the American Nuclear Society, Vol. 32, (1979), pp 382-383.

This paper discusses the design possibilities available for a radioactive waste package for spent fuel. Some areas discussed include: the form of the spent fuel, material possibilities for a canister and stabilizer, canister geometry, and liner considerations. The advantages and disadvantages of many of the individual options are presented with regards to total system parameter interrelationship, e.g., the canister wall thickness can be based on corrosion rate (desired lifetime), or accident handling stresses.

see also

11.003	11.009	11.013
11.019	12.033	12.123
14.019	15.001	16.006
16.061	17.002	17.004
18.007	19.006	22.008

14. Thermal Effects

14.001

Anonymous, "New Problems Arise for Nuclear Waste Storage", Chemical and Engineering News, Vol. 56, No. 24, (1978), p 28.

The deleterious effects of heat in an underground geologic repository are summarized. Effects of heat on groundwater, rock properties, and on the leach rate of various waste forms are of major concern.

14.002

Auty, D.B., and Grover, J.R., "Thermal Conductivity of a Borosilicate Glass Suitable for Fission Product Wastes", Harwell Atomic Energy Research Establishment, AERE-R-4685, (December, 1964), 24 p.

The thermal conductivity of a borosilicate glass incorporating mock fission product wastes has been measured by a radial heat flow method. The temperature drop across the glass/steel interface is small and hardly significant when calculating the axial temperature in cylinders of glass in storage.

14.003

Blackburn, L.D., et al., "Maximum Allowable Temperature for Storage of Spent Nuclear Reactor Fuel--An Interim Report", Hanford Engineering Development Laboratory, HEDL-TME 78-37, (May 1978).

A maximum allowable cladding temperature of 380°C is recommended on an interim basis for extended dry storage of spent nuclear reactor fuel. This

conclusion was reached after studying the following cladding failure or degradation mechanism: mechanical overload, stress rupture, stress corrosion cracking, delayed hydrogen cracking and fatigue-cracking propagation, internal hydriding, and oxidation. A temperature limit for mechanical overload was established at 396°C for PWR spent fuel and 427°C for BWR spent fuel. The limits for stress rupture vary from 361°C for 1-year storage to 321°C for 100-year storage of PWR fuel. Higher limits are obtained for BWR fuel. Preliminary estimates on a limit for stress corrosion cracking range from 290 to 350°C. Additional experiments are recommended in the area of stress corrosion cracking.

14.004

Blomqvist, G., "Preliminary Temperature Calculations for the Final Storage of Radioactive Waste in Rock", KBS Technical Reports 1-120, Report No. 15, (1977-1978).

Near-field temperatures are predicted for different waste disposal scenarios. Important variables include: (1) elapsed time between generation and disposal; (2) geometry of canister; (3) emplacement configuration; (4) distance between tunnels; and (5) thermal conductivity of waste, backfill, and rock materials.

14.005

Blomqvist, G., "Calculation of Temperatures in a Single-Level Final Repository in Rock for Vitrified Radioactive Waste", KBS Technical Reports 1-120, Report No. 45, (1977-1978).

Near-field temperatures for a one-level, horizontal area having the dimensions of 1 x 1 km with a 10 percent bentonite/90 percent quartz backfill material are calculated.

14.006

Bolmgren, C.R., et al., "Temperature Regime Associated with Dry Storage of Spent LWR Fuel", Westinghouse Advanced Energy Systems, Transactions of the American Nuclear Society, Vol. 34, (1980), pp 427-428.

This paper includes a description and results of temperature measurements which have been made along an instrumented fuel assembly and in the soil around the canister in a dry well storage hole at the EMAD facility. The testing was performed with 1 and 2 kw heaters, and also with a PWR fuel assembly with a decay heat rate of 1.1 kw. All measured temperatures are plotted on a single graph of temperature vs. power level along with predicted values for the maximum center fuel rod temperature.

14.007

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

This report summarizes the observations made during an in-situ test of the disposal of high-activity waste in an underground salt mine. Topics discussed that will be of interest to the waste packager include: (1) calculations of fuel assembly temperatures, both in the shipping cask and in the salt mine; (2) calculation of radiation dose levels; and (3) radiolytic gas production. In addition, a hazards analysis was conducted, and this includes information on resistance of the cask to fire, shielding against ionizing radiation, criticality prevention, removal of decay heat, evaluation of shipping hazards, and evaluation of accidents during handling.

14.008

Callahan, G.D., et al., "Heat Transfer Analysis of the Waste-Container Sleeve/Salt Configuration, Technical Memorandum Report (RSI-0023)", RE/SPEC, Inc., ORNL/SUB-4269/7, (March 1975), 68 p.

The heat transfer analysis of the waste-container sleeve/salt configuration is examined using two-dimensional finite element computer models. The CAN-1 and CAN-2 models were used to determine the magnitudes of induced thermal stresses in the steel sleeve that surrounds the waste container.

14.009

Davis, W., Jr., "Temperature Profiles within Cylinders Containing Internal Heat Sources and Materials of Temperature-Dependent Thermal Conductivities, Description of Fast Computer Programs as Applied to Solidified Radioactive Wastes", Oak Ridge National Laboratory, ORNL-4345, (January 1969), 32 p.

The safety and economic aspects of producing and storing radioactive wastes as solids, usually in vessels having a cylindrical geometry, require that the maximum internal temperatures to be expected as a result of decay heat be known. This report presents a computer program (STORE), which was written to permit more rapid calculation of temperature profiles within cylinders containing homogeneously distributed heat sources and materials whose thermal conductivities can be expressed as a tabular function of temperature.

14.010

Deane, J.S., and Hollis, A.A., "Practical Aspects of Heat Transfer in Radioactive-Waste Repository Design", Harwell Atomic Energy Research Establishment, AERE-R-9343, (January 1979), 19 p.

III-111

An assessment has been made of the effect of high near-field temperature on the practical features associated with the disposal of HLW in a repository constructed within a granite formation. The effect of thermal conductivity, waste encapsulation, the use of a backfill material, and reduction in the axial spacing between canisters on groundwater temperature have been modeled on a computer.

14.011

Ensminger, D.A., et al., "Presealing Risk Analysis", Analytic Sciences Corp., UCRL-15168, (January 1980), 85 p.

The report describes work done in FY 1978. Heat loss mechanisms and heating rates for glass waste are given from one to ten years. Maximum temperatures for two canister spacings (1.5 and 3 ft) are given. The effect of side radiation on maximum canister wall temperature for 2, 5, and 10 years is given. Human factor accidents are discussed. Geometry for determining radiation view factors to space from regions on the inner pair of concentric cylinders is figured, and equations are given in Appendix B for radiation view factors. Appendix C is a list of specialists contacted for sources of human factors.

14.012

Ewest, E., and Levi, H.W., "Evaluation of Products for the Solidification of High-Level Radioactive Waste From Commercial Reprocessing in the Federal Republic of Germany", Hahn-Meitner-Institut, Management of Radioactive Waste from the Nuclear Fuel Cycle, Vol. 1, (CONF-760310), STI/PUB/433, International Atomic Energy Agency, Vienna, (1976), pp 15-25.

This paper outlines the efforts of the Federal Republic of Germany to provide adequate disposal of HLW. The properties of the solidification product are discussed. Examinations of the irradiation stability, thermal stability, mechanical stability, and chemical stability are performed. Chemical stability primarily concerns the leachability of the waste in solid form. The mechanical stability concerns both the ability of the waste to maintain its structure, so that the surface area will not increase, thus increasing the leaching process, and its ability to not be broken into small particles and thus spread by air. The stability of the various solidification products is compared with the simplicity of manufacture. General policy requirements for disposal are discussed.

14.013

Gera, F., et al., "Considerations in the Long-Term Management of High-Level Radioactive Wastes", Centro Nazionale Energia Nucleare, ORNL-4762, (February 1972), 145 p.

The report covers many characteristics of solidified HLW, particularly leachability and heat generation. The leach rates were determined using water at room temperature and are presented in tabular form. Leachability is also related to solubility and temperature effects. The heat generation rates assume that the solid waste is in the form of glass and is contained in a stainless or mild steel canister. The heat generation rate controls the dimensions of the cylinders as does the thermal conductivity of the glass. Heat generation tables are included which show the different rates of heat generation as related to the chemical composition and time after reactor discharge. The gas generation or radiolysis of the chemical structure is discussed. The report also discusses the corrosive properties of container materials in certain geologic formations.

III-113

14.014

Guber, W., et al., "Development and Characterization of the Glass-Ceramic VCP 15 for Immobilization of High-Level-Liquid Radioactive Waste", Nuclear Research Center, Karlsruhe, Ceramics in Nuclear Waste Management, Chikalla, T.D., and Mendel, J.E., (eds), Cincinnati, Ohio, April 30 - May 2, 1979, CONF-790420, (1979), pp 188-192.

Characteristics of the glass ceramic VCP 15 are identified and compared to a KFK borosilicate glass product. It is reported that most of the properties of the glass-ceramics are better than those of the glass, especially thermal properties. The leach rates are also slightly better. The most important advantage of the glass ceramics is their greater thermodynamic stability and consequently improved product stability for long-term storage.

14.015

Hoskins, A.P., et al., "Heat Transfer Considerations in the Canister Storage of High-Level Solidified Wastes", Allied Chemical Corp., ICP-1090, (April 1976).

Heat transfer within solidified wastes, and from storage canisters to air and water cooling media is considered in this report as a function of waste form, permissible heat generation rate, and canister configuration. The number of storage canisters required annually for a 1500 MTU/year reprocessing plant was calculated for various waste forms and canister configurations. Waste forms studied were stabilized (nitrate and water removed) calcines, glass, sintered glass-ceramics, and calcine metal matrices. Solid cylinders, cylinders with external longitudinal fins, and unfinned annular cylinders were the canister configurations studied. Heat generation rates were calculated based on maximum permissible product temperature and canister wall temperature. Heat generation rates were correlated with waste decay time and product waste concentration to determine the number of canisters required annually.

14.016

Jansen, G., and Kaser, J.D., "Improved Heat Transfer from Radioactive Waste Canisters", Battelle Pacific Northwest Laboratories, Transactions of the American Nuclear Society, Vol. 17, (1973), p 327.

The radioactive nuclide content that can be tolerated in a canister full of solid waste can be increased by lowering the thermal resistance of the average path for heat flowing to the surroundings from the waste. Several types of canister designs and their resulting heat transfer characteristics are presented. Cost-benefit analyses are also performed.

14.017

Jardine, L.J., and Steindler, M.J., "A Review of Metal-Matrix Encapsulation of Solidified Radioactive High-Level Waste", Argonne National Laboratory, ANL-78-19, (May 1978).

Literature describing previous and current work on the encapsulation of solidified HLW forms in a metal matrix are reviewed. Encapsulation of either stabilized calcine pellets or glass beads in alloys by casting techniques was concluded to be the most developed and direct approach to fabricating solid metal-matrix waste forms.

Steady-state heat transfer properties of waste canisters in air and water environments were calculated for four reference waste forms: calcine, glass monoliths, metal-encapsulated calcine, and metal-encapsulated glass beads. A set of criteria for the maximum allowable canister centerline and surface temperatures and heat generation rates per canister at the time of shipment to a federal repository was developed. Also the shortest time after reactor discharge that canisters could be filled and the subsequent "interim" storage times prior to shipment to a federal repository for various canister

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diameters and waste ages were determined. A reference conceptual flowsheet based on existing or developing technology for encapsulation of stabilized calcine pellets is discussed.

14.018

Jardine, L.J., and Steindler, M.J., "Metal Matrix Encapsulation of Waste", Argonne National Laboratory, Waste Management and Fuel Cycles '78. Post, R.G., and Wacks, W.E., (eds), University of Arizona, Tucson, (1978), pp 388-397.

This paper introduces the metal-encapsulation concept giving a process flow sheet for encapsulation of wastes from a 5 Mg/day reprocessing plant. Probable product and process attributes of metal-encapsulated waste forms are compared to those of glass monolith forms. Thermal conductivities are larger than those of glass monolith forms, which allows shorter waste aging. Leach rates for reasonably corrosion-resistant metal-encapsulated forms such as lead are comparable to those for nondevitrified borosilicate glass. Dispersibility risks from mechanical impacts in transportation and handling accidents are addressed.

14.019

Jenks, G.H., "Maximum Acceptable Temperatures of Wastes and Containers During Retrievable Geologic Storage", Office of Waste Isolation, Y/OWI/TM-42, (August 1977).

Estimates of the maximum acceptable temperatures of waste and containers during retrievable geologic storage are given. The estimates are made by examining the properties of the disposal package which are temperature dependent. The effects of temperature on material properties, solubility, leaching, corrosion, and mechanical stress are discussed in relation to the retrieval of the waste. The concept of retrievability is defined, and the

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period of retrievability is assumed. Acceptable temperature guidelines are determined for varying types of waste and container materials.

Conformance to the ONWI Criteria. This was an early attempt to specify maximum allowable temperatures, and the limits identified in this report are not applicable to the ONWI criteria.

14.020

Knutsson, S., "Thermoconductivity Experiments with Buffer Material of Bentonite/Pitesilt", KBS Technical Reports 1-120, Report No. 28, (1977-1978).

The thermal conductivity of the 10 percent bentonite/90 percent quartz Swedish backfill material is discussed. Theoretical relationships associated with the various heat transfer mechanisms in moist granular materials are presented. Also included are data from experimental measurements of the thermal conductivity of the buffer material as a function of water content.

14.021

Knutsson, S., "Thermal Conductivity Tests on Buffer Material of Compacted Bentonite", KBS Technical Reports 1-120, Report No. 72, (1977-1978).

Data are reported on the thermal conductivity of the highly compacted potential backfill material-10 percent bentonite/90 percent quartz. Results of both theoretical calculations and experimental analyses are presented.

III-117

14.022

Larson, D.E., "High-Level Waste Canister Thermal Characterization for the Sealed Storage Cask Concept", Atlantic Richfield Hanford Co., ARH-LD-104, (December 1974), 66 p.

The study was done to determine the compatibility of various commercial HLW and canister characteristics. The parameters included calcine and glass, heat generation rate (0.25 to 5 kw/hr/ft³), radiation generation rate (0 to 10 yr old reference waste), canister diameter (6 to 24 in.), and length (10 to 15 ft). Shielding requirements and storage unit weights were determined for the various filled canisters.

14.023

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

Comparisons in temperature increases incurred from hypothetical storage of 133 MW of 10-yr old spent fuel or high-level waste in underground salt formations have been made using the HEATING5 computer code. The comparisons are based on far-field homogenized models that cover areas of 65 and 25 sq miles for spent fuel and HLW, respectively, and near-field unit-cell models covering respective areas of 610 ft² and 400 ft². Preliminary calculations based on heat loads of 150 kw/acre and 3.5 kw/canister indicate the temperature increases expected and amount of thermal energy stored in the salt after 500 years for both spent fuel and HLW. Detailed far-field comparisons of transient temperature increases have been made based on these same loadings. Comparisons were made between a canister containing HLW with an initial heat production rate of 2.1 kw and a canister containing a spent fuel assembly producing 0.55 kw.

14.024

Lowry, W.E., et al., "The Effects of Annular Air Gaps Surrounding an Emplaced Nuclear Waste Canister in Deep Geologic Storage", Lawrence Livermore Laboratory, American Society for Metallurgical Engineers, Winter Meeting, Chicago, Illinois, November 16-21, 1980.

Annular air space surrounding an emplaced nuclear waste canister in deep geologic storage will have significant effects on the long-term performance of the waste forms. Addressed specifically in this analysis is the influence of a gap on the thermal response of the waste package. Three dimensional numerical modeling predicts temperature effects for a series of parameter variations, including the influence of gap size, surface emissivities, initial thermal power generation rate of the canister, and the presence/absence of a sleeve. Particular emphasis is placed on determining the effects these variables have on the canister surface temperature.

14.025

McCann, R.A., "HYDRA-I: A Three Dimensional Finite Difference Code for Calculating the Thermohydraulic Performance of a Fuel Assembly Contained Within a Canister", Battelle Pacific Northwest Laboratories, PNL-3367, (December 1980), 71 p.

This report describes a finite difference computer code which simulates the three-dimensional performance of a spent fuel assembly contained within a cylindrical canister. The code accounts for the coupled heat transfer modes of conduction, convection, and radiation and permits spatially varying boundary conditions, thermophysical properties, and power generation rates. The appendix summarizes approximately two dozen different cases that have been examined.

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14.026

McCarthy, G.J., et al., "Hydrothermal Reactivity of Simulated Nuclear Waste Forms and Water-Catalyzed Waste-Rock Interactions", Pennsylvania State University, Scientific Basis for Nuclear Waste Management, Vol. 1, McCarthy, G.J., (ed), Plenum Press, New York, (1979), pp 329-340.

Experiments were performed in sealed gold capsules contained in autoclaves or cold-seal pressure vessels. The pressure was uniformly set at 300 bars, a value that corresponds to the pressure in a backfilled repository at a depth close to 1000 m. The temperature covered in the experiments was 200 to 300°C. The simulated spent fuel pellets, a mixture of 96.5 weight percent UO_2 and 3.5 weight percent "reduced fission products", was meant to resemble LWR fuel with a burnup of approximately 33,000 MWd/MTU.

14.027

McCarthy, G.J., et al., "Hydrothermal Stability of Spent Fuel and High-Level Waste Ceramics in the Geologic Repository Environment", Pennsylvania State University, Ceramics in Nuclear Waste Management, Chikalla, T.D., and Mendel, J.E., (eds), Cincinnati, Ohio, April 30 - May 2, CONF-790420, (1979), pp 274-276.

The stability of four simulated waste forms (spent fuel, a reference glass, a supercalcine ceramic, and a coated ceramic) under hydrothermal conditions (200 to 300°C; 400°C for coated ceramic, 300 bars) with solutions alone and in contact with basalt and shale are summarized.

14.028

McCarthy, G.J., et al., "Simulated High-Level Waste--Basalt Interaction Experiments, First Interim Progress Report", Pennsylvania State University, RHO-BWI-C-12, (March 1978).

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Reprocessed waste forms, calcine and glass, have been found to be relatively more reactive in hydrothermal experiments than the more crystalline waste forms, spent unprocessed fuel and supercalcine. Experiments with cesium compounds have shown that pollucite forms and, as a result, fixes 86 to 99 percent of the cesium.

14.029

McCarthy, G.J., et al., "Hydrothermal Stability of Simulated Waste Glass", Pennsylvania State University, RHO-BWI-SA-15, (December 1978), 57 p.

Simulated HLW glass PNL 76-68 was tested for its stability in contact with pressurized water at 300°C in a closed system for a few weeks. The solutions were analyzed for various elements.

14.030

McCarthy, G.J., "High-Level Waste Ceramics: Materials Considerations, Process Simulation and Product Characterization", Pennsylvania State University, Nuclear Technology, Vol. 32, (January 1977), pp 92-105.

The various processes for manufacturing supercalcines are described and the phases which result are identified. Supercalcines are reported to have good thermal stability with a maximum weight loss of 7.5 percent when fired at 1200°C for 24 hr. The bulk leachability of 9 different types of supercalcines is reported to be between 6×10^{-5} and 5×10^{-6} g/cm²/day using the Soxhlet leach test.

III-121

14.031

McCarthy, G.J., et al., "Interactions Between Nuclear Wastes and Surrounding Rock", Pennsylvania State University, Nature, Vol. 273, (May 1978), pp 216-217.

This study shows that contact with pressurized water at 200 to 400°C rapidly alters borosilicate glass, and promotes interactions between the waste and both shale and basalt repository wall rocks. Uranyl silicate is formed by hydrothermal alteration of the glass and cesium and sodium from calcine reaction with aluminosilicates in basalt and shale to form the cesium-sodium aluminosilicate, pollucite. Several hydrothermal experiments that confirm the waste-rock interactions are discussed. The formation of sodium-pollucite is noted as potentially desirable in a waste-rock interaction occurring if the container is breached during the thermal period. The cesium in the waste calcine is effectively fixed and in equilibrium with certain natural hydrothermal environments.

14.032

McElroy, J.L., et al., "Recent Calcination and Vitrification Process Accomplishments", Battelle Pacific Northwest Laboratory, Management of Radioactive Wastes from the Nuclear Fuel Cycle, Vol. 1, (CONF-760310), STI/PUB/433, International Atomic Energy Agency, Vienna, (1976), pp 283-302.

The corrosion of 304L stainless steel canisters during melting processes is minimal. The contraction of the canister is greater than that of the waste. The compressive stress on the glass is beneficial, but the canister may yield under the tensile stress.

14.033

Mendel, J.E., et al., "Thermal and Radiation Effects on Borosilicate Waste Glasses", Battelle Pacific Northwest Laboratories, Management of Radioactive Wastes from the Nuclear Fuel Cycle, Vol. 2, (CONF-760310), STI/PUB/433, International Atomic Energy Agency, Vienna, (1976), pp 49-61.

The devitrification behavior of zinc borosilicate glass at various temperatures from 300 to 1000°C for various times is given. Leach rate, porosity, and relative concentration data are presented for specimens held at temperatures from 300 to 900°C for 2 months. For the radiation effects studied, curium-244 was substituted for uranium in the zinc borosilicate glass. Data are presented on stored energy and density increase as a function of cumulative alpha dose. Seven curium doped glass specimens were leached in distilled water at 25°C to determine the effect of alpha radiation on leachability.

14.034

Pariseau, W.G., "Thermoelastic/Plastic Analysis of Waste-Container Sleeve: I. Initial Estimates of Loading on the Sleeve, Technical Memorandum Report (R&I-0010)", RE/SPEC, Inc., ORNL/SUB-3706/6, (May 1974).

This report presents the results of first estimates of waste-container sleeve loading as obtained by axially-symmetric thermo-elastic/plastic analysis. The computer code SALT-4/T simulates a sequence of operations that includes: (1) excavation of a typical repository room and drillhole in the room floor; (2) emplacement of the sleeve and the waste container; and (3) subsequent heating over a 10-yr period. Stress changes caused by the excavation are added to the initial gravity stresses in order to obtain the post-mining stress field. The waste container and sleeve are simulated as a single solid cylinder, with a diameter of 2 feet, a length of 10 feet, and an effective modulus of elasticity of 8,000,000 psi.

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14.035

Pariseau, W.G., "Thermoelastic/Plastic Analysis of Waste-Container Sleeve: IV. Air Gap Influence on Hole Closure, Technical Memorandum Report (RSI-0019)", RE/SPEC, Inc., ORNL/SUB-4269/3, (March 1975).

This report presents the results of an axially symmetric thermo-elastic/plastic finite element analysis of the closure of an air-gap between the borehole wall and the waste-container sleeve assembly positioned in the lower 10 ft of an 18-ft-deep borehole centrally located in a typical 18-ft-wide by 17-ft-high room adjacent to a 60-ft-wide pillar. The objective of the analysis performed with SALT-4/T Model is to obtain an estimate of the maximum air-gap closure expected over a ten year period of heating. The effect of air-gap size on temperature, field history, and retrievability is determined.

14.036

Pusch, R., "Influence of Cementation on the Deformation Properties of Bentonite/Quartz Buffer Substance", University of Lulea, KBS Technical Reports 1-120, Report No. 14, (1977-1978).

The nature of silica (SiO_2) dissolution and precipitation in the 10 percent bentonite/90 percent quartz backfill material is investigated. Temperature along with pH and the ionic composition of percolating groundwater are important variables. Silica precipitation is of concern because it can lead to embrittlement of the backfill.

14.037

Ramspott, L.D., and Ballou, L.B., "Test Storage of Spent Reactor Fuel in the Climax Granite at the Nevada Test Site", Lawrence Livermore Laboratory, Waste

Management '80, Post, R.G., (ed), University of Arizona, Tucson, (1980), pp 515-529.

This generic test is located 420 m below the surface in the Climax granitic stock. Although no effects are deemed likely on the granite, the combination of heat and radiation could cause significant deleterious effects on backfill materials of the bentonite-zeolite types currently under consideration. The temperature history of the rock surface at fuel-assembly midpoint in the center hole of a 17-canister spent fuel storage array is calculated.

14.038

Rankin, W.N., "Canister Compatibility with Carlsbad Salt", Savannah River Laboratory, Scientific Basis for Nuclear Waste Management, Vol. 2, Northrup, C.J.M., Jr., (ed), Plenum Press, New York, (1980), pp 395-402.

No significant reaction was found when candidate canister alloys were heated with salt from Carlsbad, New Mexico, for up to 5000 hours in sealed capsules and for up to 10,000 hours in unsealed capsules at temperatures (80 to 225°C) that bracket the maximum temperature calculated for reference waste containers at 20-foot spacings in salt. The canister materials tested were 304L stainless steel and low-carbon steel. At extreme temperature conditions (600°C) tests indicate no significant attack of Type 304L stainless steel; however, there was extensive attack of low-carbon steel by volatile products of the salt released by decrepitation.

14.039

Scheetz, B.E., et al., "Simulated High-Level Waste-Basalt Interaction Experiments, Annual Progress Report, October 1-September 30, 1978", Pennsylvania State University, RH0-BWI-C-33, (November 1978).

III-125

The objectives were to determine the nature of waste-basalt interactions and to begin to establish an understanding of their relevance to radionuclide release rates. The experiments suggest that the observed reactivity of calcine and glass at 300°C would be anticipated at lower temperatures but only after longer times of hydrothermal treatment. Analysis of the solution phase in the closed system experiments revealed that nearly all of the boron, about 70 percent of the molybdenum, and about 50 percent of the sodium in the original glass were dissolved and were present in the aqueous phase.

14.040

Simonen, F.A., and Friley, J.R., "Stress Analysis of Glass Canister Interaction: A Study of Residual Stresses and Fracturing", Battelle Pacific Northwest Laboratories, Ceramics in Nuclear Waste Management, Chikalla T.D., and Mendel, J.E., (eds), Cincinnati, Ohio, April 30 - May 2, 1979, CONF-790420, (1979), pp 333-337.

This paper discusses the simulation of residual stresses and cracking in canisters filled with vitrified nuclear waste using finite element computer calculations. Cooling rates, heat generation, and thermal expansion coefficients significantly affect stress levels. Glass behavior within the softening temperature range is taken to follow the instant freezing concept of Bartenev. Computer simulation descriptions of solidification and cooling processes which predict time histories of stress in the glass canister system are discussed.

14.041

Slate, S.C., et al., "Stresses and Cracking in High-Level Waste Glass", Battelle Pacific Northwest Laboratories, High-Level Radioactive Solid Waste Forms, Casey, L.A., (ed), Denver, Colorado, December 19-21, 1978, (CONF-781241), NUREG/CP-0005, (April 1979), pp 393-455.

Mechanical durability, specifically cracking as it relates to cast waste glass in steel storage canisters is discussed. The main causes of cracking during filling of continuous-melter canisters are temperature gradients from uneven filling and thermal shock from the wall. The fact that cracking does not have a significant impact on safety in the solidification facility or during transport is discussed. Residual stresses in canisters are given in Appendix B of this paper.

14.042

Steindler, M.J., et al., "Chemical Engineering Division Fuel Cycle Programs October-December 1967", Argonne National Laboratory, ANL-77-36, (1977), 79 p.

Progress on several different fuel cycle studies for the quarter, October-December 1976, is presented. Of interest are calculations of projected surface temperatures as a function of total HLW canister heat generation rates for different canister diameters.

14.043

Sutherland, S.H., and Allen, G.C., Jr., "High-Level Defense Waste and Spent-Fuel Characterization for Geologic Waste Repositories", Sandia Laboratories, Transactions of the American Nuclear Society, Vol. 30, (1978), pp 283-284, (also CONF-781105-23 and SAND-78-1289C).

The thermal characteristics of both BWR and PWR spent fuel are reported. Decay heat rates are expressed for the first 10^6 years after removal from the reactor.

14.044

Tarandi, T., "Temperature Calculations for Spent Fuel", KBS Technical Reports 1-120, Report No. 46, (1977-1978).

The temperatures and temperature gradients in a repository containing BWR spent fuel are calculated. The results are also applicable for PWR spent fuel if the amount of fuel is reduced so that the effect per canister is the same. Final calculations are based on conservative assumptions.

14.045

Tennant, M.H., "Temperatures Generated by Underground Storage of Defense Waste Canisters", Savannah River Laboratory, Scientific Basis for Nuclear Waste Management, Vol. 2, Northrup, C.J.M., Jr., (ed), Plenum Press, New York, (1980), pp 825-840.

Temperatures have been calculated for canisters containing vitrified radioactive waste buried in a generic salt repository. Each canister generates approximately 1 kw of radionuclide heat at burial. For conditions conducive to leaching, i.e., moisture in contact with glass, canister spacing is the dominant parameter in determining glass surface temperatures. For dry salt in contact with the storage container, the size and content of the annulus surrounding the storage container can be an important parameter. The repository causes a temperature rise in an increasing volume of surrounding geology over a period of years. The placement of waste canisters in an underground repository will take into account temperature restrictions placed on the waste glass, containers, and surrounding geology. In conjunction with ONWI, SRP plans additional studies to set storage temperature restrictions based on leach rates and canister compatibility with salt, water, and salt brine. Determination of the limiting thermal restrictions will also require consideration of phenomena such as devitrification, decrepitation, salt brine migration, and mine stability.

14.046

Turcotte, R.P., and Wald, J.W., "Devitrification Behavior in a Zinc Borosilicate Nuclear Waste Glass", Pacific Northwest Laboratories, PNL-2247, (March 1978).

This report discusses the effect of temperature on the thermally induced changes of a simulated nuclear waste glass. The results are used to predict the behavior of the glass during long-term storage and suggest that canister centerline temperatures be limited to less than 375°C for retention of desirable glass properties.

Conformance to the ONWI Criteria. The ONWI specification regarding maximum canister centerline temperature has not yet been established.

14.047

Walsh, E.L., and Bathke, E.A., "Thermal Calculations in Shale", Kaman Sciences Corp., Y/OWI/SUB-76/16502, (September 1976), 40 p.

Thermal calculations are presented for underground nuclear waste disposal assuming shale as the site material. Average thermal properties used were for horizontally bedded shale with waste disposal at a depth of 2000 ft below the surface. Temperatures expected as a function of time after disposal are presented. Effects of ventilation are also discussed.

14.048

Wang, J.S.Y., et al., "A Study of Regional Temperature and Thermohydrological Effects of an Underground Repository for Nuclear Wastes in Hard Rock", Lawrence Berkeley Laboratory, LBL-8271, (1979).

III-129

Heat released by the radioactive decay of nuclear wastes in an underground repository causes a long-term thermal disturbance in the surrounding rock mass. Some of these disturbances are discussed. Based on a power density of 10 w/m^2 of spent fuel assemblies ten years after discharge from a reactor, the maximum temperature increase and thermal gradient in a repository in granite is calculated.

see also

11.002	11.009	11.010
11.018	11.021	12.010
12.012	12.015	12.017
12.018	12.021	12.022
12.036	12.037	12.038
12.040	12.043	12.049
12.052	12.061	12.063
12.064	12.074	12.077
12.083	12.087	12.090
12.092	12.093	12.094
12.095	12.099	12.101
12.102	12.104	12.107
12.115	12.125	12.126
12.127	12.129	12.138
12.143	12.146	12.154
12.160	12.162	13.003
13.004	13.008	13.010
13.011	13.012	13.015
13.107	15.001	15.006
15.012	16.001	16.006
16.020	16.021	16.022
16.023	16.025	16.026
16.027	16.030	16.033
16.034	16.035	16.037
16.038	16.040	16.041
16.042	16.044	16.045

THERMAL EFFECTS

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16.046	16.049	16.050
16.053	16.056	16.058
16.061	16.064	16.065
16.072	16.073	17.003
18.004	18.005	18.010
18.012	18.017	19.006
20.001	20.008	21.006
21.008	22.002	22.004
22.005	22.010	22.011
22.012		

15. Mechanical Properties

15.001

Angerman, C.L., and Rankin, W.N., "Durability of Containers for Storing Solidified Radioactive Wastes", Savannah River Laboratory, (CONF-770303-1), DP-MS-76-66, (1976), 30 p.

Although geared towards storage rather than disposal of solidified HLW, some of the concerns discussed in the report are closely related to the concerns for the container intended for geologic disposal. Various accidents involving the container are analyzed including subjection to fire and impact. The compatibility of various materials is discussed.

15.002

Anonymous, "HIPed Alumina Container for Nuclear Waste", Nuclear Engineering International, Vol. 22, (August 1977), p 38.

A nontechnical discussion of the Hot Isostatic Pressing Process for the fabrication of ceramic waste forms is presented. It is reported that the resulting waste forms have very high leach resistance and mechanical strength.

15.003

Bunnell, L.R., "Tests for Determining Impact Resistance and Strength of Glass Used for Nuclear Waste Disposal", Battelle Pacific Northwest Laboratory, PNL-2954, (May 1979).

Tests are described for determining the impact resistance and static tensile strength of glasses containing simulated or actual nuclear wastes.

This report describes the development and use of these tests to rank different glasses, to assess effects of devitrification, and to examine the effect of impact energy on resulting surface area.

15.004

Callahan, G.D., et al., "Thermo/Viscoelastic Analysis of the Waste-Container Sleeve: I. Time-Dependent Salt Loading on the Sleeve and Closure of the Drillhole", RE/SPEC, Inc., ORNL/SUB-4269/4, (October 1975).

Canister retrievability and encasement-sleeve structural integrity analyses are presented. Estimates of the radial sleeve loading and drillhole closure resulting from thermo/viscoelastic behavior of the sleeve were conducted using finite element model computer programs. Two cases were examined, namely the case of fixing the drillhole surface against radial displacement and the case of allowing the drillhole to displace radially without restraint.

15.005

Gnirk, P.F., "Elastic Solutions for Selected Circular Symmetric Problems as Applied to the Design of the Waste Container Sleeve", RE/SPEC Inc., ORNL/SUB-3706/4, (September 1973).

The relationships for the radial stress component in a steel waste container are presented for three limiting conditions of the steel-sleeve-salt configuration. The three limiting conditions for which results are presented are: two hollow salt/steel cylinders with stress free inner boundary, hollow salt cylinder with inner solid steel cylinder, and hollow salt cylinder with zero radial displacement at inner wall. Results using the derived relationships for these conditions are tabulated for horizontal field stress of 1333 and 2000 psi.

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15.006

Gnirk, P.F.. et al., "Thermoelastic Analysis of Waste-Container Sleeve, Technical Memorandum Report RSI-0008", RE/SPEC Inc., ORNL/SUB-37065, (January 1974).

This report contains the results of a thermoelastic analysis on a steel waste container sleeve in a salt repository. Finite element calculations were performed to determine the radial temperature and stress distributions in both the waste container and the salt which is in the near vicinity of the sleeve. Temperature profile results are shown graphically for a steel sleeve within an initial internal heating at a 5 kw power level and a decay constant of 0.023105. Radial stress profiles are presented both in graphical and tabular form after 5 yrs storage.

15.007

McElroy, J.L., "Quarterly Progress Report, Research and Development Activities Waste Fixation Program. October through December 1976", Battelle Pacific Northwest Laboratory, PNL-2264, (November 1977).

This progress report discusses the High-Level Waste Fixation Program at PNL. Canister design for the in-can melting process is discussed. Canister material properties of concern are the corrosion and high-temperature strength properties. Waste glass materials, including 76-68, PW-8a-3, 76-101, and 73-1, are examined for thermal effects, leachability, and impact strength. Tests on alternate waste forms, such as disc pellets, glass marble, and chemical-vapor-deposition coating are performed to determine the thermal properties of the glass compositions. Multibarrier waste forms, particularly supercalcine in stainless steel and glass in metal matrix, are analyzed for impact strength.

15.008

Nilsson, F., "Viewpoints on the Mechanical Reliability of a Canister for Nuclear Waste", KBS Technical Reports 1-120, Report No. 66, (1977-1978).

Resistance to stress for three different canister types (copper, alumina, and lead-titanium) is discussed. Susceptibility to fracture and creep are reported.

15.009

Pariseau, W.G., "Thermoelastic/Plastic Analysis of Waste Container Sleeve: II. Influence of Large Displacements on Sleeve Loading, Technical Memorandum Report (RSI-0017)", RE/SPEC, Inc., ORNL/SUB-4269/1, (March 1975).

Modification of the thermoelastic/plastic finite element program to account for large displacements possibly associated with the development of an extensive plastic zone about a radioactive waste container emplaced in a typical repository room (SALT-4/T model) has been completed. Comparisons of radial stresses acting on the waste container and borehole wall displacements computed by the modified and conventional analyses techniques reveal little difference between the two results over a 10 yr heating period for salt strengths 25 percent of original values.

15.010

Pariseau, W.G., "Thermoelastic/Plastic Analysis of Waste Container Sleeve: III. Influence of Salt Strength on Sleeve Loading, Technical Memorandum Report (RSI, 0018)", RE/SPEC, Inc., ORNL/SUB-4269/2, (March 1975).

Loading estimates on the sleeve of the waste container are obtained by axially symmetric thermoelastic/plastic analysis using salt tensile

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strengths that are considerably less than the compressive strengths. Computer plots of principal stresses and displacement fields at various stages of the excavation and heating simulation aid in the visualization of repository concept mechanics and show the possible need for additional mesh refinement for more precise stress information.

15.011

Pusch, R., "Required Physical and Mechanical Properties of Buffer Masses", University of Lulea, KBS Technical Reports 1-120, Report No. 33, (1977-1978).

The physical, physico-chemical, and mechanical properties of the proposed bentonite/quartz backfill are discussed. The rock supporting power of different bentonite/quartz ratios is also reported.

15.012

Rusin, J.M., et al., "Multibarrier Waste Forms. Part II: Characterization and Evaluation", Battelle Pacific Northwest Laboratory, PNL-2668-2, (August 1979).

This report discusses the characteristics of four multibarrier waste forms and compares them to reference calcine and glass. The four waste forms discussed are: simulated waste-glass marbles encapsulated in a vacuum-cast P-10Sn matrix; uncoated sintered supercalcine pellets encapsulated in a vacuum-cast Al-12Si matrix; glass-coated sintered supercalcine pellets encapsulated in a vacuum-cast Al-12Si matrix; and PyC/Al₂O₃ CVD-coated supercalcine encapsulated in a gravity-sintered copper matrix. The four multibarrier waste forms were evaluated for thermal stability, mechanical strength (impact resistance), and leach resistance.

15.013

Simonen, F.A., and Slate, S.C., "Stress Analysis of High-Level Waste Canisters: Methods, Applications, and Design Data", Battelle Pacific Northwest Laboratories, PNL-3036, (October 1979).

An overview of stress analysis methods, structural design procedures, and design data is presented for canisters used to package solidified wastes, particularly borosilicate glass. Waste processing canister materials, fabrication, and inspection methods, and performance testing are summarized. Sources of stress in canisters are lifting and handling loads, internal pressure, high-temperature filling operations, transient heating and cooling, differential thermal expansions of canisters and glass, and impact loadings from low-probability accidents. Results of case studies that illustrate applicable methods of stress analyses are presented for these sources of stress. Specialized criteria for minimum wall thicknesses to sustain filling stresses are proposed in this report. Results of a test program to measure the creep strength of candidate canister materials are described. Consequences of residual stresses are reviewed, and stress-corrosion cracking is identified as the mode of canister failure affected by residual stresses. Canister closure design is covered in detail, particularly the welding and inspection of the final closure seal-weld.

15.014

Simonen, F.A., and Slate, S.C., "Creep Analysis of Canisters for Radioactive Waste Vitrification", Battelle Pacific Northwest Laboratories, Transactions of the American Nuclear Society, Vol. 27, (1977), pp 259-260.

Canisters to be used for vitrification of radioactive wastes by in-can melting must be able to sustain stress during processing at temperatures in the range of the melting temperatures of the alloys used for the canister. Studies to establish a base of creep strength data and to

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verify the ability of stress analysis to predict creep deformations for the canister are being conducted. Inconel 601 and 304L stainless steel were selected for tests of rupture strength and creep curves. Isochronous stress-strain curves were found to be suitable for estimates of creep deformation in the canisters to be used for the in-can melting process.

15.015

Tignac, L.L., "Closure Weld Development for Spent Fuel Canisters; Interim Report", Rockwell Hanford Operations, RHO-LD-95, (June 1979).

The welding processes evaluated were gas metal arc welding, pulsed-current gas metal arc welding and high-energy laser processes. A full-scale remote welding demonstration effort using the gas metal arc welding process was initiated and a canister handling mechanism to facilitate remote welding was designed and fabricated.

15.016

Wallace, R.M., and Kelley, J.A., "An Impact Test for Solid Waste Forms", Savannah River Laboratory, DP-1400, (March 1976).

Samples of concretes and glasses being considered for incorporation of radioactive waste sludge were subjected to impact tests to determine the relationship between energy of the impact and the resulting increase in surface area of the damaged sample. The particulate fraction was screened to size, and the final particulates were examined with a Coulter counter. The particle size distribution was from 2000 microns down to 0.6 microns.

MECHANICAL PROPERTIES

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see also

12.008	12.021	12.033
12.038	12.040	12.063
12.064	12.076	12.077
12.085	12.090	12.093
12.094	12.111	12.114
12.115	12.125	12.126
12.129	12.151	12.165
13.014	14.012	14.017
14.021	16.008	16.020
16.023	16.027	16.036
16.040	16.041	16.044
16.045	16.046	16.048
16.049	16.050	16.051
16.052	16.056	16.060
16.072	18.009	18.012
18.014	21.008	21.018
21.022	21.023	22.002
22.004	22.005	22.008
22.012		

16. Solid16.001

Aaron, W.S., et al., "Development and Characterization of Cermet Forms for Radioactive Waste", Oak Ridge National Laboratory, Scientific Basis for Nuclear Waste Management, Vol. 2, Northrup, C.J.M., Jr., (ed), Plenum Press, New York, (1980), pp 315-322.

Cermets designed to isolate HLW in a solid form are a composite consisting of various ceramic phase particles uniformly dispersed in and micro-encapsulated by an iron-nickel base alloy matrix. Typical property ranges for the cermets are presented in tables. Among these are waste loading of 2 to 75 weight percent, thermal shock of 900°C to room temperature over 3 cycles, and Soxhlet leach test rate of 7.1×10^{-6} g/cm²/day, and 0.024 percent cesium loss over 72 hours.

16.002

Bancroft, A.R., "The Incorporation of Fission Products into Glass for Disposal", Chalk River Nuclear Laboratories, Canadian Journal of Chemical Engineering, Vol. 38, (1960), pp 19-24.

Nepheline syenite with up to 30 weight percent of lime is used as the glass to incorporate fission product wastes from nuclear reactors. A nitric acid solution of the fission products is added to the pelletized glass base. Nitrates are decomposed at 130 to 600°C and the mixture melted to form a glass at a maximum temperature of 1350°C. Ruthenium and cesium are volatilized and must be removed from the off-gas stream and incorporated into the glass. Leaching tests were conducted.

16.003

Battelle Pacific Northwest Laboratories, Materials Characterization Center, "Nuclear Waste Materials Handbook", DOE/TIC-11400, (to be published September 1981).

This document has not yet been published but is included in this bibliography due to its expected importance to the waste package designers. The document will contain up to date information on materials which play a part in geologic waste disposal. As test procedures, materials data, and supporting documentation are approved by the Materials Review Board, the information will be added to the handbook. The handbook will therefore be frequently updated. The first volume of the handbook will contain procedures and material data on waste forms and will be available about September 1981. Subsequent volumes will contain data and test procedures on other engineered barriers.

16.004

Bauer, A.A., et al., "The Influence of System Consideration on Waste Form Design", Office of Nuclear Waste Isolation, Ceramics in Nuclear Waste Management, Chikalla, T.D., and Mendel, J.E., (eds), Cincinnati, Ohio, April 30 - May 2, 1979, CONF-790420, (1979), pp 27-32.

This paper presents the objectives of the U.S. Waste Management Program for the disposal of radioactive waste in a manner that provides an acceptable risk to man. The design of waste forms is constrained by waste management system considerations imposed during generation, treatment, packaging, transportation, storage, and isolation. In the isolation phase, the waste form provides one of the barriers to release in a multibarrier system that includes the natural geologic and hydrologic barriers as well as other engineered barriers. A table of the advantages and disadvantages of various waste form types is presented.

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16.005

Bechtel National, Inc., "An Assessment of LWR Fuel Disposal Options", ONWI-39, Vols. 1-3, (July 1979).

This report is the final product of the Spent Fuel Disposal Study (SFDS). The report: (1) identifies several spent fuel waste forms as well as HLW and other waste forms from fuel reprocessing; (2) defines facility and system design features to handle, transport, process, encapsulate, and dispose of these waste forms; and (3) provides a comparative assessment of each of the spent fuel waste alternatives with respect to technical feasibility, safeguards, criticality, radiological impact, retrievability, and economics. Vol. 1 of this three volume series contains the summary and conclusions and provides the reader with a general overview of the study. Vol. 2 contains detailed evaluations and risk analyses of the study cases in each of the six assessment areas, and describes and evaluates a number of study variations. A large amount of information pertinent to the package designer is contained in the report. Vol. 3 (Appendices) provides a comprehensive tabulation of the bases and assumptions used in the study as well as preconceptual design descriptions and cost estimates of the facilities and transportation systems necessary to implement the various study cases.

16.006

Blomeke, J.O., et al., "Spent Fuel as a Waste Form", Oak Ridge National Laboratory, Waste Management and Fuel Cycles '78, Post, R.G., and Wacks, M.E., (eds), University of Arizona, Tucson, (1978), pp 608-621.

Spent fuel assemblies as solid waste forms should be encapsulated in a canister as a barrier to the spread of contamination in case of accident. The spent fuel assemblies would be placed in geologic isolation as in salt formations, or, as in Sweden, placed in steel containers, wound into a spiral configuration, and placed in a shaped aluminum oxide container sealed with an

aluminum oxide cover. The containers are then placed in granite caverns and filled with a mixture of bentonite and quartz. Advantages and disadvantages of using spent fuel assemblies as the solid waste form are discussed.

16.007

Bonniaud, R.A, et al., "Large Scale Production of Glass for High-Level Radioactive Waste", Commissariat a l'Energie Atomique, Nuclear and Chemical Waste Management, Vol. 1, No. 1, (1980), pp 3-16.

A description of the Marcoule continuous vitrification process is presented. Of interest to package designers are two tables. The first table lists nuclide composition for three different fuel types. The second table lists the chemical composition for three different Marcoule glass waste types.

16.008

Chapman, C.C., et al., "Vitrification of Hanford Wastes in a Joule-Heated Ceramic Melter and Evaluation of Resultant Canistered Product", Battelle Pacific Northwest Laboratories, PNL-2904, (August 1979), 55 p.

Results of a test run that produced borosilicate glass from simulated radioactive wastes are presented. The wastes were vitrified in an electric melter, and the glass product was cast in metal canisters. These glass castings were subsequently evaluated to determine chemical composition, density, leachability, and surface area both for the "as-cast" condition and after a drop onto a massive pad. Illustrations of glass leachability versus glass volume are included.

16.009

Chick, L.A., et al., "Statistically Designed Study of a Nuclear Waste Glass System", Battelle Pacific Northwest Laboratories, Scientific Basis for Nuclear Waste Management, Vol. 2, Northrup, C.J.M., Jr., (ed), Plenum Press, New York, (1980), pp 175-182.

The statistical experimental design and approximation modeling techniques employed in this study have been demonstrated to be effective in yielding empirical understanding and predictive capability of the effect of composition on properties in complex glasses. There is a general agreement between the component effects predicted by modeling of independent sets of data, which lends credence to the results.

16.010

Clelland, D.W., et al., "A Review of European High-Level Waste Solidification Technology", British Nuclear Fuels, Ltd., Management of Wastes from the LWR Fuel Cycle, CONF-760701, Denver, Colorado, (July 1976), pp 137-165.

Progress toward the development and demonstration of industrial-scale solidification process in Belgium, France, Germany, and the United Kingdom is reviewed, including such aspects as the historical background of research and development work, overall objectives, processes under investigation, and the present research and development program.

16.011

Dickey, B.R., et al., "High-Level Waste Solidification: Applicability of Fluidized-Bed Calcination to Commercial Wastes", Allied Chemical Corp., Nuclear Technology, Vol. 30, (December 1974), pp 371-382.

The paper describes the calcination process. A table is given which compares the average size, bulk density, gross porosity, and thermal conductivity of calcined solids from the fluidized bed, radiant-heat spray, and rotary ball kiln processes.

16.012

Dosch, R.G., "Ceramic Forms for Nuclear Waste", Sandia Laboratories, Radioactive Waste in Geologic Storage, Fried, S., (ed), American Chemical Society Symposium Series 100, Washington, D.C., (1979), pp 129-148.

The preparation, composition, structure, and leaching characteristics of a crystalline, ceramic radioactive waste form are discussed, and where applicable, compared with vitrified waste forms. The baseline product is a pelletized ceramic waste form with a density range of 4.2 to 4.5 g/cm³, containing approximately 25 weight percent of fission waste oxides. Although the pellets are visually homogeneous, microstructural analyses show them to be a very complex material consisting of an assemblage of crystalline phases along with amorphous phases due to zeolite and silicon additions. The matrix material is rutile, TiO₂, in which at least thirteen other crystalline phases have been identified.

16.013

Frischat, G.H., "Self-Diffusion Processes in Oxide Glasses", Technical University of Clausthal, Nuclear Technology, Vol. 51, (1980), pp 130-135.

This paper discusses self-diffusion mechanisms in oxide glasses, both single alkali and mixed alkali glass systems. Self-diffusion processes can be measured using radioactive or stable isotopes and are described. The process of self-diffusion is of fundamental importance in understanding more complex chemical processes in glasses or leaching processes between glasses and

aqueous solutions. Some of the important features of self-diffusion in glasses especially with respect to the fixation process are discussed. A recommendation is made that future self-diffusion studies of oxide glasses include the effects of thermal gradients and radiation damage.

16.014

Ghattas, N., et al., "Some Studies on Nuclide Losses. Part 1. Vitrification of High Activity Waste", Harwell Atomic Energy Research Establishment, Journal of Applied Chemistry and Biotechnology, Vol. 28, No. 8, (1978), pp 523-530.

Data concerning the percentage loss of radioactive nuclides of interest during the fundamental steps of vitrification of a simulated HLW solution and the high efficiency of the calcinate filter used are reported.

16.015

Ghattas, N.K., et al., "Some Studies of Nuclide Losses. Part 2. Cold Finger", Harwell Atomic Energy Establishment, Journal of Applied Chemistry and Biotechnology, Vol. 28, No. 12, (1978), pp 847-853.

The vaporization rates of cesium, tellerium, ruthenium, strontium, and cerium during vitrification of simulated HLW are reported. Vaporization rates as a function of temperature are also discussed.

16.016

Guber, W., (ed), "Thermodynamically Stable Product for Permanent Storage and Disposal of Highly Radioactive Liquid Wastes", Nuclear Research Center, Karlsruhe, U.S. Patent 4,097,401, filed July 30, 1975, (June 1978).

A description of a solid thermodynamically stable microcrystalline glass ceramic for permanent disposal and storage of highly radioactive wastes in a manner harmless to the environment is discussed. The composition of the components of the solid are given.

16.017

Headley, T.J., et al., "Microstructural Characterization of Glass and Ceramic Simulated Waste Forms", Sandia Laboratories, Scientific Basis for Nuclear Waste Management, Vol. 2., Northrup, C.J.M., Jr., (ed), Plenum Press, New York, (1980), pp 249-254.

The characterization of three nonradioactive glass samples, which simulate three Hanford process waste forms, and two samples of iodine sodalite, which simulate the fixation of radioactive iodine, are presented. These samples were intended only to demonstrate the potential for fixation of waste in a high integrity solid form, and not necessarily representative of the final products to be used for immobilization of Hanford defense waste.

16.018

Heimerl, W., "Solidification of HLW Solutions with the PAMELA Process" DWK, Ceramics in Nuclear Waste Management, Chikalla, T.D., and Mendel, J.E., (eds), Cincinnati, Ohio, April 30 - May 2, 1979, CONF-790420, (May 1979), pp 97-101.

A revised concept for the PAMELA process for the solidification of HLW solution is described. Some new practices, such as the future use of borosilicate glass instead of phosphate glass, and manufacture of glass blocks as well as blocks of glass beads in metal matrices, are described. Leach rate data on these solids in brines at elevated temperatures and pressures are presented.

16.019

Holcomb, W.F., "A Survey of the Available Methods of Solidification for Radioactive Wastes", U.S. Environmental Protection Agency, ORP/TAD-78-2, (November 1978).

The numerous solidification techniques and related matrix materials that are presently being offered or proposed for incorporating radionuclides into an immobilizing material are reviewed. The HLW solidification methods reviewed are fluidized-bed calcination, spray calcination, pot calcination, rotary kiln calcination, cementation, in-can melting, and continuous melting. Other waste forms mentioned but not discussed in any detail are supercalcine, sintered calcines, metal matrices, glass/ceramics, coated pellets, and aqueous silicates.

16.020

International Atomic Energy Agency, "Characteristics of Solidified High-Level Waste Products", Vienna, (1979).

The following HLW characteristics are discussed: waste solution compositions; solidification process product types and compositions including calcines, glasses and various ceramics; the physical state of the above mentioned products; thermal effects on the products including phase separation, crystallization and devitrification; and product physical properties such as formation temperature, viscosity, electrical conductivity, density, thermal conductivity, thermal expansion coefficient, mechanical strength, and homogeneity. Leaching methods, mechanisms, and results of leaching tests are also presented. The effects of radiation discussed include volume changes, stored energy, helium generation leach rate of irradiated glasses, and changes in mechanical strength. Data from the United States, United Kingdom, France, Italy, Japan, Federal Republic of Germany, India and Belgium are presented where appropriate.

16.021

Johnson, K.D.B., and Marples, J.A.C., "Glasses and Ceramics for Immobilization of Radioactive Wastes for Disposal", Harwell Atomic Energy Research Establishment, AERE-R-9417 (May 1979), 29 p.

The UK Research Programme on Radioactive Waste Management includes the development of processes for the conversion of high-level liquid reprocessing wastes from thermal and fast reactors to borosilicate glasses. The properties of these glasses and their behaviour under storage and disposal conditions have been examined. Subjects discussed include leach rate of Fingal glasses, effects of heat on devitrification rates, and combined effects of heat and radiation on waste form durability.

16.022

Johnson, K.D.B., "The UK Program-Glasses and Ceramics for Immobilization of Radioactive Waste for Disposal", Harwell Atomic Energy Research Establishment, Ceramics in Nuclear Waste Management, Chikalla, T.D., and Mendel, J.E., (eds), Cincinnati, Ohio, April 30 - May 2, 1979, CONF-790420, (1979), pp 17-24.

This report outlines the UK Research Program on Radioactive Waste Management for conversion of liquid HLW to borosilicate glasses. The properties of these glasses and their behavior under storage and disposal conditions have been examined. Illustrations provide information about leaching and human exposure to isotopes, and bulk leachability of various wastes.

16.023

Lakatos, T., "Durability of Borosilicate Glass", KBS Technical Reports 1-120, Report No. 44, (1977-1978).

The properties of different borosilicate glasses made by the French, English, Germans, and Americans are presented. Properties of interest include resistance to radiation, mechanical characteristics, and resistance to thermal devitrification.

16.024

Larker, H.T., and Wanland, P., "The Direct Disposal of Spent Nuclear Fuel in Alumina Containers", ASEA, Sweden, (August 1978), 28 p.

The direct disposal of spent nuclear fuel according to the method proposed by the ASEA involves the containment of nuclear fuel rods in alumina canisters. Experiments and calculations have been performed to estimate the effective life of an alumina canister of one decimeter thickness in the deep repository environment.

16.025

Laude, F., "Glass as the First Barrier for Long-Term Storage of Highly Active Radioactive Wastes", Commissariat a l'Energie Atomique, Risk Analysis and Geologic Modeling, Ispra, Italy, May 23-27, 1977, BNWL-TR-274, 34 p.

Glass is a homogeneous inorganic material which is capable of incorporating most oxides. To be able to evaluate the risks that activity will spread in the environment as soon as blocks of glass containing radioactive waste are buried, it is of paramount importance to know the characteristics of these glasses as well as the influence of time on their physicochemical properties. This report presents data on the chemical (leaching), thermal, and radiolytic stability of various glasses.

16.026

Levi, H.W., "Survey of Nuclear Waste Forms", Hahn-Meitner-Institut, Scientific Basis for Nuclear Waste Management, Vol. 2, Northrup, C.J.M., Jr., (ed), Plenum Press, New York, (1980), pp 21-30.

This paper gives an overview of the waste management program in The Federal Republic of Germany. Described in this report are the policies and plans for characterization of the waste form and its management. Of particular interest are the use of multiple barriers, a review of waste forms, the chemical, thermal, and irradiation stabilities of the waste form, and a waste form suitability assessment, all of which are similar in context to present ONWI plans for waste disposal.

16.027

Lutze, W., et al., "Characterization of Glass and Glass Ceramic Nuclear Waste Forms", Hahn-Meitner-Institut, Scientific Basis for Nuclear Waste Management, Vol. 1, McCarthy G.J., (ed), Plenum Press, New York, (1979), pp 69-81.

The criteria for the evaluation of solid waste forms are their thermal, chemical, mechanical, and radiation stability. Qualitative criteria for the selection of a suitable solidified glass and glass ceramic waste form are discussed. Of particular interest is a table in which data are compiled on the composition and physical properties of phosphate and borosilicate glasses and Celsian and Fresnoite-type glass ceramics. Data on melting point, nucleation temperature, crystallization temperature, heat conductivity, coefficient of thermal expansion, leach rates, impacted block grain sizes, and stored energy are provided.

16.028

Lutze, W., "Glassy and Crystalline High-Level Nuclear Waste Forms--An Attempt at Critical Evaluation", Hahn-Meitner-Institut, Ceramics in Nuclear Waste Management, Chikalla, T.D., and Mendel, J.E., (eds), Cincinnati, Ohio, April 30 - May 2, 1979, CONF-790420, (1979), pp 47-51.

This paper discusses the advantages and disadvantages of glass as opposed to crystalline nuclear waste forms. An evaluation is given in terms of technological simplicity and the safety-relevant product properties, i.e., thermal, chemical, mechanical and radiation stability. Also, simulations of hydrothermal waste/rock interactions were performed using autoclave leach tests.

16.029

Macedo, P.B., et al., "Porous Glass Matrix Method for Encapsulating High-Level Nuclear Wastes", Catholic University of America, Ceramics in Nuclear Waste Management, Chikalla, T.D., and Mendel, J.E., (eds), Cincinnati, Ohio, April 30 - May 2, 1979, CONF-790420, (1979), pp 321-326.

A waste form is described which consists of a high-silica glass core containing the waste elements in its structure and a protective layer of high-silica glass covering the core which is free of waste elements. It is reported that a higher coefficient of thermal expansion and a lower transition temperature in the core than in the protective layer lead to mechanical strengthening and good resistance to stress corrosion cracking in the composite due to the development of a high residual compressive stress on the surface of the glass composite. The chemical durability was found to be extremely high and was tested by immersing powdered glass samples in distilled water at 45 and 70°C for a variety of times up to 10 days. The rate of silica release into the distilled water was measured and is reported as being representative of the release rate of most waste elements in the glass.

16.030

Malow, G., "An Investigation of Crystallization in Glasses Containing Fission Products", Hahn-Meitner-Institut, Ceramics in Nuclear Waste Management, Chikalla, T.D., and Mendel, J.E., (eds), Cincinnati, Ohio, April 30-May 2, 1979, CONF-790420, (1979), pp 203-209.

Five potential solidification products for HLW, four borosilicate glasses and one Celsian glass ceramic, have been investigated in terms of devitrification. The effect of temperature on crystallization rates was studied, and the leach rates of devitrified glasses were measured.

16.031

McElroy, J.L., "Quarterly Progress Report, Research and Development Activities Waste Fixation Program April through June 1974", Battelle Pacific Northwest Laboratories, BNWL-1841, (July 1974), 102 p.

This progress report is the seventh in a series that presents research and development activities in the field of radioactive waste fixation. A study has begun to determine the long-term effects of alpha radiation on solidified HLW curium-244 as the radiation source. Enough curium-244 is being added to the waste to simulate 100 year storage in a 1-year test. •The study is measuring changes in leach rates, stored energy, helium-buildup effects, and physical properties.

16.032

McElroy, J.L., "Quarterly Progress Report, Research and Development Activities, Waste Fixation Program October through December 1974", Battelle Pacific Northwest Laboratories, BNWL-1893, (February 1975), 83 p.

This progress report is the ninth in a series that presents research and development activities in radioactive waste fixation. A series of laboratory experiments have been completed showing the buildup of stored energy due to alpha particle emission in synthetic HLW incorporated in borosilicate glass. Also, using an electric manometer system, helium diffusivity in simulated waste glass is being examined as a function of time and temperature.

16.033

McElroy, J.L., "Quarterly Progress Report, Research and Development Activities Waste Fixation Program, July through September 1975, Pacific Northwest Laboratory", Battelle Pacific Northwest Laboratories, BNWL-1949, (January 1975), 11 p.

This progress report is the twelfth in a series that presents research and development activities in radioactive waste fixation. Numerous individual research projects are described. Those of most interest here include a thermal analysis of canister design concepts including effects of type of storage, use of fines, emissivity, and canister size and cracks in glass and a devitrification study on waste-containing glasses.

16.034

McElroy, J.L., et al., "Waste Solidification Technology, U.S.A.", Battelle Pacific Northwest Laboratories, Management of Wastes from the LWR Fuel Cycle, Denver, Colorado, CONF-760701, (1976), pp 166-189.

Processes for solidifying HLW to calcine, glass, or several other alternative solid forms are discussed. Major emphasis is placed on borosilicate glass containing 25 to 35 weight percent waste oxide. The thermal and radiation stability of glass and some leach rate data are briefly presented.

16.035

McElroy, J.L., and Burns, R.E., "Nuclear Waste Management Status and Recent Accomplishments", Battelle Pacific Northwest Laboratories, EPRI-NP-1087, (May 1979).

This report is a review of HLW and spent fuel disposal. Vitrification in various glasses with different properties is discussed. Other solidification materials discussed are supercalcine and SYNROC. A Swedish project of packaging spent fuel is mentioned which has a copper canister holding 500 fuel rods. The canister is filled with lead and has a copper lid. When emplaced, the canister is surrounded with bentonite and sand.

16.036

Mecham, W.J., et al., "Characterization of Impact Fracture of Brittle Solid Waste Forms", Argonne National laboratory, Scientific Basis for Nuclear Waste Management, Vol. 2., Northrup, C.J.M., Jr., (ed), Plenum Press, New York, (1980), pp 307-314.

This paper presents a methodology for analyzing particle size distributions obtained in impact testing of brittle waste form materials. The methodology includes (a) a linear two-parameter lognormal correlation of the weight fraction less than any given size, (b) a surface-energy constant to predict the anticipated increase of surface area from the known energy absorbed in the impact of the brittle material, and (c) a mathematical function of the two lognormal parameters to determine the total surface area in terms of a dimensionless shape factor.

16.037

Mendel, J.E., et al., "High-Level Radioactive Waste Management Research and Development Program at Battelle PNL", Battelle Pacific Northwest Laboratories, (CONF-740428-2), BNWL-SA-4844, (1974), 30 p.

This is a background paper on Battelle Pacific Northwest Laboratories' research and development waste management program as of 1974. The authors discuss various characteristics of glass/borosilicate glass as a waste form. There are also results of canister testing under the WSEP program (Waste Solidification Engineering Prototypes) at PNL.

16.038

Mendel, J.E., et al., "High-Level Radioactive Waste Management Research and Development Program at Battelle PNL", Battelle Pacific Northwest Laboratories, High-Level Radioactive Waste Management, Campbell, M.H., (ed), Advances in Chemistry Series No. 153, American Chemical Society, Washington, D.C., (1976), pp 93-107.

Battelle Pacific Northwest Laboratory is working to perfect methods for immobilizing HLW from reprocessing fuels. The process converts HLW to silicate glass forms in a two-step calcination-melting process. The process consists of drying commercial HLW, melting it in glass-forming chemicals, and casting the molten glass into stainless steel canisters. Typical silicate glass compositions and properties during processing and during storage are discussed. Formation of a quasi-glass-ceramic with many of the advantages of a true glass-ceramic may be possible in HLW solidification. Thermal and radiation stability of borosilicate waste glass was also investigated, noting leach rates. The "Russian Doll" approach using various layers of different inert materials each with a specific property is mentioned.

16.039

National Academy of Sciences, "Solidification of High-Level Radioactive Wastes, Final Report", National Research Council, NUREG/CR-0895, (June 1979).

This document reviews potential solid waste forms for HLW, as well as spent fuel assemblies as a waste form. The report concludes that various solid forms could meet standards as stringent as those currently applied to the handling, storage and transportation of spent fuel assemblies; that many solid waste forms are likely to be satisfactory in an appropriately designed system; that differences in potential health hazards to the public resulting from the use of various solid forms and disposal options are likely to be small, so cost, reliability, and health hazards to operating personnel will be major considerations in choosing among the options; and that responsibility for all radioactive waste management operations should be centralized.

16.040

Neumann, W., and Kofler, O., "Coating of Waste Containing Ceramic Granules", Oesterreichische Studien gesellschaft fuer Atomenergie, Ceramics in Nuclear Waste Management, Chikalla, T.D., and Mendel, J.E., (eds), Cincinnati, Ohio April 30 - May 2, 1979, CONF-790420, (1979), pp 150-154.

Simulated HLW granules produced by fluidized-bed calcination are overcoated with pyrocarbon and nickel in laboratory-scale experiments. The coated granules have properties in conformance with many of the proposed ONWI criteria. The mechanical strength, particle size, temperature resistance, thermal conductivity, porosity, and leach resistance of the resulting solids are discussed.

16.041

Oguino, N., et al., "Solidification of HLLW by Glass-Ceramic Process", Power Reactor and Nuclear Fuel Development Corp., Ceramics in Nuclear Waste Management, Chikalla, T.D., and Mendel, J.E., (eds), Cincinnati, Ohio, April 30 - May 2, 1979, CONF-790420, (1979), pp 143-149.

The composition of glass-ceramics for the solidification of HLW were studied. Physico-chemical properties of glass-ceramics such as leach rates, thermal stability, and mechanical strength, were measured and compared to comparable properties for borosilicate glasses.

16.042

Ooka, K., et al., "Solidification of HLLW Into Sintered Ceramics", Toshiba Corp., Ceramics in Nuclear Waste Management, Chikalla, T.D., and Mendel, J.E., (eds), Cincinnati, Ohio, April 30 - May 2, 1979, CONF-790420, (1979), pp 155-159.

Simulated HLW is solidified into sintered ceramics by normal sintering or hot-pressing with addition of some oxides. Resulting solids are evaluated in terms of leach rate, thermal characteristics, and mechanical strength.

16.043

Ringwood, A.E., et al, "Immobilization of High-Level Nuclear Reactor Wastes in SYNROC", Australian National University, Nature, Vol. 278, No. 5701, (1979), pp. 219-223.

Solidification of HLW into a synthetic rock, SYNROC, is proposed. The chemical and mineralogical composition of SYNROC is reported. Various

physico-chemical properties of SYNROC, such as leach rates, waste loading capacity, and susceptibility to radiation damage, in addition to processing costs, are discussed.

16.044

Ross, W.A., (ed), "Glass Waste Forms for Radioactive Waste Containment", Battelle Pacific Northwest Laboratories, 20th Annual AIChE Meeting, New York, (CONF-771102-28), BNWL-SA-6382, (November 1977), 30 p.

This report presents data on the leachability, thermal stability, and rate of devitrification, mechanical stability, and radiation resistance of various waste glasses.

16.045

Ross, W.A., "Properties and Characteristics of High-Level Waste Glass", Battelle Pacific Northwest Laboratories, Ceramic and Glass Radioactive Waste Forms, Readey, D.W., and Cooley, C.R., (eds), CONF-770102, (January 1977), pp 49-64.

Three important characteristics can be noted which have contributed to the choice of glass as the first line waste form. These are its leachability, its ability to incorporate a wide variety of elements, and its size to surface area ratio. The important properties and characteristics of a waste form are discussed in this paper along with the chemical composition of water and various glasses. Also, anticipated thermal environments for waste glasses from processing, interim storage, transportation, and final disposal are presented. This study also reviews the volatility, bulk properties, and physical strength of waste glasses.

16.046

Ross, W.A., and Mendel, J.E., (eds), "Development and Characterization of Solidified Forms for High-Level Wastes: 1978. Annual Report", Battelle Pacific Northwest Laboratories, PNL-3060, (December 1979).

This report covers several topics including glass and sintered ceramic compositions; pyrolytic carbon and corundum coatings, alternative waste forms to glass such as metal matrix, supercalcine, and ORNL cermet, cesium and strontium leaching from glass in deionized water at 25°C; hydro-thermal studies of various glass compositions at 350°C; radiation effects on glass and crystalline waste solids, the effects of transmutation on the properties of several glasses and crystalline compositions, and thermal and mechanical shock of canisters.

16.047

Ross, W.A., et al., "Processes for Production of Alternative Waste Forms", Battelle Pacific Northwest Laboratories, Waste Management '79, Post, R.G., (ed), University of Arizona, Tucson, (1979), pp 335-348.

During the past 20 years, numerous waste forms and processes have been proposed for the solidification of HLW. At least five factors must be considered in selecting the waste form and process method: (1) processing stability; (2) waste loading; (3) canister size and stability; (4) waste form inertness and stability; and (5) processing complexity. This paper describes various waste form processes and operations, and a single system is proposed for making comparisons of the relative importance of the above factors.

16.048

Roy, D.M., and Gouda, G.R., "High-Level Radioactive Waste Incorporation into (Special) Cements", Pennsylvania State University, Nuclear Technology, Vol. 40, (September 1978), pp 214-219.

This feasibility study demonstrated that strong, durable, relatively impermeable cylinders may be prepared by hot pressing at 150 to 250°C a combination of cements with radioactive materials. Calcium aluminate cements appeared to have properties superior to portland cements. Also, high strength cement cylinders were prepared with a 20 weight percent waste solid in a separate core and were very resistant to leaching by water near its boiling point.

16.049

Rusin, J.M., et al., "Multibarrier Waste Forms, Part I: Development", Battelle Pacific Northwest Laboratories, PNL-2668-1, (1978).

This report describes research and development activities resulting in the demonstration of several different multibarrier waste forms for non-radioactive simulated waste compositions. Multibarrier waste forms include up to three barriers to isolate radionuclides from the environment. These barriers are a solid waste inner core, an impervious coating, and a metal matrix. The process for two inner core materials, sintered supercalcline and glass marbles, have been demonstrated. The coating barrier provides enhanced leach, impact, and oxidation resistance as well as thermal protection during encapsulation in the metal matrix. The purpose of the metal matrix is to improve impact resistance, protect the inner core from any adverse effects, provide radiation shielding, and increase thermal conductivity. Several different combinations of multibarrier waste forms and their physico-chemical properties are discussed.

16.050

Rusin, J.M., "A Review of High-Level Waste Form Properties", Battelle Pacific Northwest Laboratories, PNL-3035, (December 1980).

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Waste form options for the immobilization of liquid HLW are reviewed. A summary table is provided listing the various properties of various waste form options. It is concluded that one must consider the total waste management system in the final decision on waste form selection. Proposed waste forms have properties falling within a relatively narrow range, and several waste form options and multibarrier waste management systems are potentially available when one considers all of the possible combinations.

16.051

Rusin, J.M., and McElroy, J.L. "A Review of High-Level Waste-Form Properties", Battelle Pacific Northwest Laboratory, Transactions of the American Nuclear Society, Vol. 33, (1980), p 418.

This is a brief review paper of several glass and alternate waste forms. The waste forms discussed include: sintered ceramics, concrete, glass-ceramics, hot-pressed ceramics, and metal matrices. The ranges for leach rate and impact resistance are reported, as is the effect on leach rate of using pyrolytic carbon and alumina-coated samples.

16.052

Rusin, J.M., et al., "Alternative Waste Forms--A Comparative Study", Battelle Pacific Northwest Laboratories, Scientific Basis for Nuclear Waste Management, Vol. 2, Northrup, C.J.M., Jr., (ed), Plenum Press, New York, (1980), pp 255-264.

This paper provides a comparative study of sintered ceramics, glass ceramics, glass, and concrete on the bases of leaching, impact testing and compressive testing.

16.053

Scheetz, B.E., and McCarthy, G.J., "Synthetic Mineral Nuclear Waste Ceramics", Pennsylvania State University, Transactions of the American Nuclear Society, Vol. 33, (November 1979), p 278.

This is a brief review paper on recent work in synthetic mineral nuclear waste ceramics. Two broad types of synthetic material are specifically covered: supercalcine ceramics (SPC-2, 4 type) and Synroc (Type B). A table is included which lists the synthetic mineral structure type and nominal composition along with the hazardous radwaste elements incorporated in mineral types.

16.054

Schneider, K.J., "Status of Technology in the United States for Solidification of Highly Radioactive Liquid Wastes", Battelle Pacific Northwest Laboratories, BNWL-820, (October 1968), 63 p.

The status of technology from research and development in the United States on the solidification of highly radioactive wastes from the processing of spent fuel is discussed. A summary of individual technologies is presented for the four primary processes under development: spray calcination, phosphate glass solidification, pot calcination, and fluidized bed calcination. Technical incentives for the solidification of highly radioactive wastes before disposal are presented, and characteristics of the various solid products are summarized.

16.055

Schulz, W.W., and Dressen, A.L., "Immobilization of Spent Cladding Hulls by Conversion to Silicate Forms", Atlantic Richfield Hanford Co., ARH-SA-193S, (October 1974).

III-163

This is a brief report which discusses Zircaloy and stainless steel spent fuel cladding hulls which have been converted to silicate and/or aluminosilicate forms. Tabulated values are given of the various oxides formed as a function of the density, weight percent, state of the converted form (crystalline or glass), and the bulk leach rate.

16.056

Smith, T.H., and Ross, W.A., "Impact Testing of Vitreous Simulated High-Level Waste in Canisters", Battelle Pacific Northwest Laboratories, BNWL-1903, (May 1975), 115 p.

This work was performed in support of a risk analysis of a conceptual management system for HLW. Three basic driving forces are generally associated with the accidental release of radioisotopes: mechanical forces, thermal forces, and dissolution processes. Because no data for impact breakup characteristics of encapsulated waste glass were located, this test program was begun. The objectives, in order of priority, were to estimate: (1) the quantity of respirable glass fines produced, (2) the increase in glass surface area, and (3) the impact resistance of the filled canisters.

16.057

Solomah, A.G., et al., "HLW Fixation in Sintered SYNROC-B. Ceramics-- Compositions and Phase Characterization", North Carolina State University, Transactions of the American Nuclear Society, Vol. 34, (1980), pp 197-199.

This paper reports several physical properties of sintered SYNROC-B as a waste form for HLW. Results reported include: phase composition using X-ray diffraction analysis, location of simulated radwaste cations in the crystal lattice, and density measurements to indicate porosity. Also reported is the effect of HLW loading (5, 10, and 15 percent) on the composition of modified SYNROC-B.

16.058

Sombret, C. G., et al., "Status of the French AVM Vitrification Facility", Commissariat a l'Energie Atomique, Waste Management '79, Post, R.G., (ed), University of Arizona, Tucson, (1979), pp 263-295.

Some chemical and physical characteristics of Marcoule glasses are presented, including waste loading capacity, viscosity at 1100°C, thermal conductivity at 100°C, and leach rate in tap water at room temperature.

16.059

Steindler, M.J., et al., "Chemical Engineering Division Fuel Cycle Program, Quarterly Progress Report, October-December 1977", Argonne National Laboratory, ANL-78-34, (1978), 110 p.

The status of numerous fuel cycle studies is reported. Pertinent is a review and evaluation of the encapsulation of HLW in metal matrix waste forms. Corrosion and leach rates of simulated waste forms are being measured, and a model has been proposed to describe the reaction between solidified HLW and metals. Metal matrix corrosion studies were performed at elevated temperatures and pressure and in the presence of brines. Leachabilities of simulated waste forms encapsulated in a metal matrix are reported for 25 to 100°C.

16.060

Steindler, M.J., et al., "Fuel Cycle Programs Quarterly Progress Report, January-March 1980", Argonne National Laboratory, (1980), ANL-80-76.

The results of continuing studies on the encapsulation of radioactive waste in metal are reported. The possibility of interactions between

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simulated waste glass and lead in air and argon atmospheres was explored. A comparative economic analysis of four waste solidification processes subcontracted to TERA Corporation was completed. Ten drop-weight brittle fracture tests were conducted on Pyrex cylinders. The kinetics of fracture was studied by means of high-speed motion pictures. Developmental studies of the neutron activation method and the use of traces for determining very low leach rates were continued.

16.061

Stone, J.A., et al., "Preliminary Evaluation of Alternative Waste Forms for Immobilization of Savannah River Plant High-Level Waste", Savannah River Plant, DP-1545, (December 1979).

Eleven waste forms, representing all known proposed formulations with possible application to SRP waste, were evaluated in terms of process complexity and potential product performance. The results of the study are applicable only to SRP HLW. Detailed descriptions of borosilicate glass, high-silica glass, concrete (FUETAP process), hot-pressed concrete, supercalcine ceramic, SYNROC ceramic, cermet, glass marbles in a metal matrix, ceramic pellets in a metal matrix, coated ceramics, and coated ceramics via Sol-Gel are given in an appendix. Each waste form was evaluated based on its development status, waste loading, leachability, long-term stability, thermal conductivity, thermal stability, transportation safety, process complexity, state of development, quality assurance, yield and recycle, and process safety.

16.062

Sump, K.R., et al., "Nuclear Waste Storage Container with Metal Matrix", U.S. Patent 4,115,311, filed March 10, 1977, September 10, 1978, 6 p.

A metal matrix waste form is described which is formed by sintering a 0.05 to 2.0 mm diameter metal powder with 0.5 to 25 mm diameter HLW particles in a container. In the final waste form, the HLW particles occupy about 50 to 70 volume percent of the container with the metal matrix occupying about 35 to 65 volume percent of the remaining volume of the container.

16.063

Tempest, P.A., "Structure and Stability of Media Intended for the Immobilization of High-Level Radioactive Waste", Berkeley Nuclear Laboratories, CEGB-RD-B-N-4573, (May 1979), 28 p.

Alternate glassification technologies in the form of the French AVM continuous process and the United Kingdom HARVEST batch process are described and compared, and their likely effect on the structural properties of the final solid glass block are considered. The waste loading capacities of glasses and crystalline substrates such as SYNROC are compared.

16.064

Thompson, R.J., et al., "Waste Solidification Demonstration Program: Characterization of Nonradioactive Samples of Solidified High-Level Waste", Battelle Pacific Northwest Laboratories, BNWL-1393, (June 1970), 86 p.

The major activities associated with the waste solidification demonstration studies are the measurement of different physical and chemical properties of phosphate glass and spray calcined solid waste forms and an evaluation of the waste form properties with regards to safety and economic considerations. The product characteristics that have been measured include leachability, thermal diffusivity, heat capacity, density, drip point, thermal shock, dispersability, crystallinity, and linear coefficient of thermal expansion. These characteristics have been measured as a function of waste

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composition, melt-formation temperature, time at melt-formation temperature, rate-of-cooling to storage temperature, and storage temperature.

16.065

Thompson, R.J., et al., "Waste Solidification Demonstration Programs. Experiment Techniques for Characterization of High Level Radioactive Solidified Waste", Battelle Pacific Northwest Laboratories, BNWL-1425, (1970).

The overall objective of the Waste Solidification Demonstration Program is to provide the technological bases that will enable industry to convert the liquid HLW from the reprocessing of nuclear fuel to a safer solid form. Significant development efforts were devoted to establishing many of the measurement techniques that would allow the solidified waste and waste container to be evaluated and characterized. Techniques were developed for measuring leachability, thermal diffusivity, heat capacity, density, drip point, thermal shock, container wall thickness, dispersibility, crystallinity, linear coefficient of expansion, container dimensions, temperature distribution in large containers, leak checking, and container sealing by welding.

16.066

U.S. Department of Energy, "Management of Commercially Generated Radioactive Waste", DOE/EIS-0046-D, Vol. 1, (April 1979).

This document describes ten alternate methods for disposal of nuclear wastes, including the favored geologic disposal option. The first barrier is the waste form. Canisters provide the second barrier, then adsorptive over-pack such as zeolites and bentonites. Physical properties of the typical disposal media and conditions prevailing in the repository are discussed. Retrievable package design and packaging for long-term isolation are assessed. The performance of glass waste forms, calcine, glass ceramics, sintered

products, metal matrices, coated pellets, cermets, ion exchange media, and the "molecular stuffing" process to form a high-silica glass waste form are described. Canister materials and performance also are discussed. Canisters may be made more corrosion-resistant with the use of an overpack. An adequate data base for canister/package design has not yet been developed in the United States. Swedish HLW package design is given. Potential risks, safeguards, and possible breach of geologic repositories are discussed, as are waste management for spent fuel, reprocessed fuel, mixed-oxide fuel fabrication, once-through cycle fuel, and uranium and plutonium recycle fuel. Chemical resynthesis of liquid HLW with solidification and geologic disposal are discussed at length on a conceptual basis.

16.067

U.S. Department of Energy, "Management of Commercially Generated Radioactive Waste", DOE/EIS-0046-D, Vol. 2, (April 1979).

Several of the appendices in this document are of interest to waste packaging. Appendix I on groundwater migration discusses the solubility of the waste form and the containment package in repository conditions, rate of flow of the groundwater, flowpath to the biosphere, chemical make-up of the groundwater, and sorption of the geologic media and of the repository as factors that provide protection from and delay the release of nuclides to the biosphere. Appendix J discusses borosilicate glass characteristics. Appendix K discusses thermal load design values for HLW and spent fuel canisters and the geologic repository. Appendix Q contains information on engineered barriers including buffer materials which can absorb or slow down radioactive migration from the package and the repository.

16.068

U.S. Department of Energy, "Technology for Commercial Radioactive Waste Management", DOE/ET-0028, Vol. 2, (May 1979).

Volume 2 of this 5 volume set discusses waste treatment technology for high-, intermediate-, and low-level wastes. Included is a section on solidification of liquid HLW, using the waste calcination and vitrification processes.

16.069

U.S. Department of Energy, "Evaluation and Review of Alternative Waste Forms for Immobilization of High Level Radioactive Wastes", Savannah River Operations Office, DOE/TIC-10228, (August 1979).

The waste forms reviewed are pelletized calcine, stabilized calcine, cement, glass, stuffed glass, Synroc, supercalcine, titanates, cermet, and multibarrier waste forms. The discussion for each waste form includes a description of the process and its general characteristics. Each waste form is ranked according to its present scientific merits, its recommended research priority, and its present and potential engineering practicality.

16.070

U.S. Department of Energy, "The Evaluation and Review of Alternative Waste Forms for Immobilization of High Level Radioactive Wastes, Report Number 2", Alternative Waste Form Peer Review Panel, DOE/TIC-11219, (June 1980), 35 p.

Four generic categories of alternate nuclear waste forms are evaluated: glasses via calcination or liquid feed processes, ceramics via calcination or sol-gel processes, porous glass matrices, and multibarrier forms with glass or ceramics encapsulated in metal, glass, or ceramic matrices. It is indicated that there is an urgent need for standardized tests, standard reference materials, methods of data presentation, and comparative corrosion data. The performance and properties of alternate waste forms are briefly discussed. The review panel considered very attractive the multibarrier form

with which a zero release rate for a specified period of time may be achieved. Of these, glass marbles or ceramic pellets in lead or lead alloys, glass or ceramics with dense alumina or pyrolytic carbon coatings, an inert cladding of dense insoluble ceramics around the waste solid, and glass in a nonradioactive glass container are discussed separately.

16.071

Upson, U.L., and Weth, G.C., "Fixation of High-Level Radioactive Wastes in Phosphate Glass", Battelle Pacific Northwest Laboratories, Transactions of the American Nuclear Society, Vol. 8, (1965), pp 336-337, (also BNWL-SA-564 and CONF-660208).

The Brookhaven process for converting high-level aqueous fuel reprocessing waste directly to a phosphate glass, without an intermediate drying step, was demonstrated with actual Purex 1WW waste. Samples of glass were produced for observation and evaluation of stability and chemical durability.

16.072

Van Geel, J., et al., "Solidification of High-Level Liquid Wastes to Phosphates Glass-Metal Matrix Blocks", Eurochemic, Management of Radioactive Wastes from the Nuclear Fuel Cycle, Vol. 1, (CONF-760310), STI/PUB/433, International Atomic Energy Agency, Vienna, (1976), pp 341-358.

Solidification of HLW into phosphate glass-metal matrix blocks is described. The mechanical strength, dispersibility, leach resistance, and thermal conductivity of the final product are discussed.

16.073

Van Geel, J., et al., "Conditioning of Solid High-Level Waste Products by Dispersion into Metal Matrices", Eurochemic, Paper presented at 68th AICHE Meeting, Los Angeles, November 16-20, 1975.

Solidification processes for HLW solidification are described. The solidification processes are calcination to oxide granules, calcination to phosphate granules, and vitrification to phosphate glasses. The solid particles may be further conditioned by incorporating them into a metal matrix. Metal matrix encapsulation is being researched at Eurochemic. All of the composite products prepared show a homogeneous distribution of HLW particles within the metal matrix, no voids or indications of adverse reactions at the interphase, high mechanical strength, and a heat conductivity higher than that of the incorporated materials. The use of borosilicate glass is also described. Alloy compositions (lead, zinc, and aluminum), compatibility with the dispersed phase, leach tests, and corrosion resistance are discussed.

16.074

Weisenburger, S., et al., "Vitrification of High-Level Radioactive Waste in a Continuous Liquid-Fed Ceramic Melter", Nuclear Research Center, Karlsruhe, Ceramics in Nuclear Waste Management, Chikalla, T.D., and Mendel, J.E., (eds), Cincinnati, Ohio, April 30 - 2 May, 1979 CONF-790420, (May, 1979), pp 86-92.

The technology of vitrification of HLW in a continuous liquid-fed ceramic melter developed in Germany is described. The loss of some cesium and ruthenium is experienced. However, most of the lost material is removed in the dust removal wet scrubber and can be recycled with the scrub solutions.

SOLID

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see also

12.005	12.008	12.037
12.090	12.093	12.105
12.107	12.130	12.138
13.003	13.004	13.006
13.011	13.015	13.017
14.012	14.031	15.016
18.006	18.016	21.018
23.007		

17. Codes/Standards17.001

Birmingham, A.J., "Recommendations for Codes and Standards To Be Used for Design and Fabrication of High-Level Waste Canister", Nuclear Services Corp., RHO-C-13, (January 1978), 77 p.

This study is a part of the High-Level Waste Canister Envelope Study Program whose prime objective is to develop design criteria for the canisters that will contain the solidified HLW from commercial reprocessing operations. The primary objective is to identify codes, standards, and federal regulations that are applicable to the design and construction of HLW canisters. This study also makes recommendations for development of additional appropriate standards and requirements to ensure safety and the integrity of these canisters for the necessary period of time.

17.002

Gyllander, C., et al., "Safety and Radiation Protection in the Field of Nuclear Power. Laws, Standards and Grounds for Evaluation", KBS Technical Reports 1-120, Report No. 41, (1977-1978).

Applicable conventions, laws, standards, and regulations governing safeguards, criticality safety, and the transport and physical protection of spent fuel at the back-end of the fuel cycle are reviewed. Recommended standards as well as the basic principles for the system of dose limitations are presented. United States codes and standards are briefly summarized.

17.003

Office of Nuclear Waste Isolation, "The Disposal of Spent Nuclear Fuel", ONWI-59, (1979).

Spent fuel from BWR's and PWR's is characterized in great detail. Characteristics discussed include their: composition, form, and geometry; actinide content; thermal power; fissile isotope content; radioactive gas production rate; leach and dissolution resistance; and its possible resource value. Current knowledge concerning spent fuel isolation is reviewed, including the logistics of spent fuel transfer, packaging, handling, emplacement of a container in the repository, and the economics of spent fuel disposal. The important milestones relating to spent fuel disposal are also discussed.

17.004

U.S. Department of Energy, "United States of America Nuclear Regulatory Commission in the Matter of Proposed Rulemaking on the Storage and Disposal of Nuclear Waste (Waste Confidence Rulemaking) PR-50,51 (44FR61372)", DOE/NE-0007, (April 1980).

This rather large volume contains much information relevant to the disposal of spent fuel and HLW. Discussion that is pertinent to waste packaging is including on the following topics: (1) general performance objectives for the environmentally safe disposal of HLW; (2) spent fuel characteristics including physical characteristics of fuel rods and assemblies and thermal and radiation characteristics of a fuel assembly; (3) waste package system component interactions; and (4) alternatives and preferred methods for storage.

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see also

12.064

12.166

14.018

14.024

15.013

16.003

18.011

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18. Handling

18.001

Bahorich, R.J., et al., "Spent LWR Fuel Encapsulation and Dry Storage Demonstration", Westinghouse Advanced Energy Systems Division, Transactions of the American Nuclear Society, Vol. 34, (1980), pp 834-835.

Handling capabilities of the EMAD facility are discussed. These capabilities include: leak testing of canisters containing spent fuel, lag storage of canisters, fuel assembly calorimeter to measure the thermal output of a spent fuel assembly, and a canister cutting tool to recover an encapsulated fuel assembly after testing.

18.002

Cohen, J.J. et al., "Determination of Performance Criteria for High-Level Solidified Nuclear Waste", Lawrence Livermore Laboratory, NUREG-0279, (July 1977).

The results of a detailed study on the handling, storage, transportation, and disposal of HLW are given. The report is subdivided into two parts: the first part describes the study approach, calculational models, factors considered in the evaluation, and some preliminary results. The second part, never issued, was intended to detail a technical basis for a set of recommended criteria.

18.003

Davis, R.B., et al., "Remote Characterization of Spent LWR Fuel for Geologic Disposal Demonstrations", Hanford Engineering Development Laboratory, Transactions of the American Nuclear Society, Vol. 34, (1980), pp 838-839.

HANDLING

III-178

Eight spent fuel assemblies and forty-five spent LWR fuel rods have been remotely inspected to determine rod bow, rod ovalities, rod void volume, fission gas release, total neutron flux, and peak gamma surface dose rate.

18.004

Duncan, J.E., et al., "Spent-Fuel Handling System for a Geologic Storage Test at the Nevada Test Site", Lawrence Livermore Laboratory, Transactions of the American Nuclear Society, Vol. 34, (1980), pp 833-834.

The paper includes a brief and general description of the spent fuel handling system at the Climax Site at the Nevada Test Site. The handling system has three basic subsystems: a surface transport vehicle, a lowering system, and an underground transfer vehicle. Some physical dimensions, weights, etc., of each subsystem are given. All three subsystems are operated from a central control room which also includes closed circuit television monitors, remote radiation level indication, and ventilation system controls.

18.005

English, T., et al., "An Analysis of the Technical Status of High Level Radioactive Waste and Spent Fuel Management Systems", Jet Propulsion Laboratory, N-78-18897, (December 1977), 289 p.

The purpose of this report is to examine the technical status of the United States program for management and disposal of solidified HLW and spent fuel. A large amount of information is presented in this report, including discussions on the origin and nature of HLW and spent fuel, their heat and radiation properties, and methods for storage, solidification, and packaging.

18.006

Ensminger, D.A., et al., "Release of Radionuclides from High Level Waste During Transportation Fires", Analytic Sciences Corp., Scientific Basis for Nuclear Waste Management, Vol. 1, McCarthy, G.J., (ed.), Plenum Press, New York, (1979), pp 541-548.

An analysis is performed to determine doses to humans that resulted from five different accident scenarios in rail and truck transportation of solidified HLW. Four waste forms are analyzed: spray calcine, fluidized-bed calcine, borosilicate glass, and a special 'multibarrier' waste form. It is possible to establish discrete containment failure temperatures for each waste form.

18.007

Ericsson, A.M., "Safety in the Handling, Storage, and Transportation of Spent Nuclear Fuel and Vitrified High-Level Waste", KBS Technical Reports 1-120, Report No. 42, (1977-1978).

The safety of handling and transportation of spent fuel and vitrified HLW has been studied. This report includes a description of precautionary measures to be used in handling and transportation, analyses of radioactive releases from normal and accidental discharges, and the predicted dose to the public from such releases.

18.008

Fagerstrom, H., and Lundahl, B., "Handling of Buffer Material of Bentonite and Quartz", KBS Technical Reports 1-120, Report No. 37, (1977-1978).

Preferred methods for handling the bentonite/quartz buffer material when placing it around canisters and in tunnels are presented. Different bentonite/quartz ratios are desirable for different applications. The degree of compaction and layer thickness may also vary somewhat. Specifications for these variables for different applications are presented in the report.

18.009

Farwick, D.G., and Moen, R.A., "Properties of Light Water Reactor Spent Fuel Cladding", Hanford Engineering Development Laboratory, HEDL-TME-78-70, (1979).

The Commercial Waste and Spent Fuel Packaging program is developing containment packages for the safe storage or disposal of spent fuel. Maintaining containment of radionuclides during transportation, handling, processing, and storage is essential, so the best understanding of the properties of the material to be stored is necessary. This report provides data collection, assessment, and recommendations for spent fuel cladding materials properties. Major emphasis is placed on the mechanical properties of the Zircalloys and austenitic stainless steels. Limited information on elastic constants, physical properties, and anticipated corrosion behavior is also provided.

18.010

Henry, K.H., and Turner, D.A., "Storage of Spent-Unreprocessed Fuel (SURF)", Rockwell Hanford Operations, Waste Management and Fuel Cycles '78, Post, R.G., and Wacks, M.E., (eds), University of Arizona, Tucson, (1978), pp 356-370.

The three principal concepts for surface or near-surface storage of spent-unreprocessed fuel (SURF) are described. In the sealed-storage-cask concept, spent fuel canisters are enclosed in second canisters (or overpacks) and stored above ground in cylindrical concrete shields. Variations for single canister or up to 6 canisters in circular form, ventilated or

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nonventilated, are given. In the dry-well concept, spent fuel canisters are stored in steel-lined below-ground vertical shafts with a nonshrinkable grout at the bottom and concrete plug at top. Variations include supplemental cooling to permit storage of multiple canisters. The air-cooled-vault concept uses natural-draft air flow through a concrete vault. Fuel assemblies are contained in canisters with overpacks vertically arranged within the below-grade vault. Multiple packages (several hundred) of spent fuel may be so stored. Cladding failure, fuel-rod cladding temperatures, thermal analysis, criticality safety analyses and radiation shielding analyses work are discussed briefly.

18.011

Kaernbraenslesaeckerhet, "Handling of Spent Nuclear Fuel and Final Storage of Vitrified High Level Reprocessing Waste", INIS-MF-4460, (1978), 115 p.

This report summarizes the Swedish approach to the handling, storage, and transportation of spent fuel and vitrified HLW, and the long-term safety of the final storage of vitrified HLW in rock. The report discusses various potential dispersion processes including canister damage, leaching, and nuclide migration.

18.012

Kibbe, R.K., and Boch, A.L., "Contribution to Draft Generic Environmental Impact Statement on Commercial Waste Management: Radioactive Waste Isolation in Geologic Formations", Science Applications, Inc., Y/OWI/TM-44, (1978).

This extensive volume contains abundant data on the management of radioactive wastes. Wastes resulting from several different fuel cycles are described as are possible canisters, overpacks, and waste drums. Waste package thermal conditions and exterior contamination conditions are discussed. Surface facilities for waste handling are described. Backfilling operations

in repositories in numerous rock types are discussed. The thermal characteristics and acceptable thermal loads of waste packages are evaluated.

18.013

Likar, V.F., and Johnson, J.E., "Waste Isolation Pilot Plant (WIPP) Remote Handling Operations", Westinghouse Advanced Energy Systems Division, Transactions of the American Nuclear Society, Vol. 34, (1980), pp 846-847.

A step-by-step description of the handling procedures for disposal and retrieval of transuranic waste at WIPP are discussed. Contingency plans are also described if the waste canister becomes damaged or contaminated.

Conformance to the ONWI Criteria. Surface dose rate criteria for remotely handled waste are prescribed in this paper to be between 200 mRem/hr and 100 Rem/hr. This is not in conformance to the ONWI packaging criteria which do not apply to the WIPP.

18.014

Mecham, W.J., et al., "The Effect of Impact on Solid-Waste Composites of Brittle and Ductile Materials", Argonne National Laboratory, Ceramics in Nuclear Waste Management, Chikalla, T.D., and Mendel, J.E., (eds), Cincinnati, Ohio, April 30 - May 2, 1979, CONF-790420, (1979), pp 327-332.

Methods of analysis of impacts that could occur prior to final geologic isolation is outlined for two generic solid waste forms: (1) a glass monolith and (2) a composite of ceramic pellets embedded in a continuous metal phase. The resistance to dispersion from mechanical impacts during in-plant storage handling, and transportation and resistance to leaching by aqueous media are among the properties studied in this report.

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18.015

Perona, J.J., et al., "Design and Safety Considerations of Shipping High-Level Radioactive Wastes", Oak Ridge National Laboratory, ORNL-TM-2971, (December 1970), 28 p.

The HLW from processing a ton of spent fuel that has been irradiated to about 30,000 MWD/ton may contain from 20 to 30 g of curium-244. The neutron flux from spontaneous fission of this isotope is such as to require that a hydrogenous material be included in the shielding of casks to be used for shipment of the solidified wastes. This report presents design and materials considerations for neutron shields.

18.016

U.S. Department of Energy, "Environmental Aspects of Commercial Radioactive Waste Management", Office of Nuclear Waste Management, DOE/ET-0029, Vol. 2, (April 1979).

Chapter 9.3 on postulated accidents involving geologic isolation of LWR wastes (from DOE/ET-0028, Section 7.5) identifies maximum dose of radioactivity to individuals due to dropping a canister containing glass or calcined HLW down a mine shaft. For a glass waste form of 175kg/MTHM, about 13 kg of particles less than 10 microns in diameter will be released to mine filters, for a calcine waste form of 52.5 kg/MTHM, about 31 kg of particles will be released to mine filters. This type of accident has an expected frequency of occurrence of 7×10^{-7} /yr. Doses to the maximum individual are also given for breaching of the geologic repository by a meteorite strike, and by drilling accidents. Fault fracture and flooding is assumed and leach rates, radionuclides released and maximum body dose are given. Factors affecting the leach rate of the waste are the waste form and the temperature of the solid-liquid interaction zone. Leach rates for waste glasses, borosilicate glass and nepheline selenite glass are considered. In the

HANDLING

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discussion of solution mining for salt accidents, it is assumed the waste canister is laid bare and water contacts the waste form directly.

18.017

Westinghouse Advanced Energy Systems Division, "Spent Fuel Handling and Packaging Program, Hot Demonstration at the E-MAD Facility, Jackass Flats, Nevada", (September 1978).

Pictures and labeled diagrams describe the spent fuel handling and packaging facility, the drywall storage facility, the sealed storage cask and liners, the canister arrangement showing the canister lid closure and concrete shield plug, and the soil and fuel temperature tests.

see also

11.014	12.064	12.092
12.102	14.007	14.011
16.021	16.045	17.004
19.003	28.002	

19. Retrieval19.001

Bhaskaran, G., and McCleery, J.E., "Retrieval System for Emplaced Spent Unreprocessed Fuel (SURF) in Salt Bed Depository: Accident Event Analysis and Mechanical Failure Probabilities", Lawrence Livermore Laboratory, UCRL-15111, (October 1979), 250 p.

This report provides support in developing an accident prediction event tree diagram, with an analysis of the baseline design concept for the retrieval of emplaced spent unreprocessed fuel contained in a degraded canister. It contains an evaluation check list, accident diagrams, accident event tables, fault trees/event trees and discussions of failure probabilities for the following subsystems as potential contributors to a failure: canister extraction, including the "core and cram" units; canister transfer at the hoist area; and canister hoisting.

19.002

Boeing Engineering and Construction, "High-Level Waste Canister Emplacement and Retrieval Concepts Study", Y/OWI/SUB-76/16509, (September 1976).

This is a conceptual design report for emplacement and retrieval of HLW canisters in salt repositories. Four vertical, two horizontal and a 45 degree emplacement and retrieval concept were generated under this study. System flow charts are developed and presented which describe the required canister handling operations.

19.003

Dennis, A.W., and Stinebaugh, R.E. "Retrieval Methods and Equipment for WIPP High-Level Waste Experiment Area", Sandia Laboratories, Transaction of the American Nuclear Society, Vol. 33, (1980), pp 944-946.

The necessary equipment and procedures for in-situ testing of HLW at WIPP are discussed. The remotely controlled equipment under development includes: retrieval cask with adaptor for retrieval drilling equipment, manifold for attachment of the retrieval cask and salt cutting collectors, contaminated salt cutting collector, overcoring drill, the retrieval package, transporter for the retrieval cask, and an underground area transfer cell. Also discussed is a step-by-step procedure on retrieval of a waste canister, especially when contamination of the surrounding geological media is indicated.

19.004

Hudson, E.E., and McCleery, J.E., "Retrieval System for Emplaced Spent Unreprocessed Fuel (SURF) in Salt Bed Depository. Baseline Concept Criteria Specifications and Mechanical Failure Probabilities", Lawrence Livermore Laboratory, UCRL-15025, (May 1979), 189 p.

One of the integral elements of the Nuclear Waste Management Program is the material handling task of retrieving canisters containing spent unreprocessed fuel from their emplacement in a deep geologic salt bed repository. Alternative concepts for the tasks of retrieval are illustrated and critiqued, a baseline concept in scenario form is derived and basic retrieval subsystem specifications are presented with cyclic failure probabilities predicted. The report is based on these assumptions: (1) during retrieval a temporary radiation seal is placed over each canister emplacement; (2) a sleeve surrounding the canister was initially installed during the original emplacement; and (3) the emplacement room's physical and environmental conditions established in

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the report are maintained while the task is performed. The baseline design of spent fuel retrieval systems will be based on retrieving the spent fuel canisters from intact or degraded sleeves for a period of up to 60 yrs.

19.005

Irby, H.H., and Segura, M. "Retrievability of Waste at WIPP," Westinghouse Advanced Energy Systems Division, Transactions of the American Nuclear Society, Vol. 34, (1980), pp 845-846.

The general time schedule for the WIPP retrieval system development is shown. The system will primarily handle TRU waste, but will also be designed to handle HLW.

19.006

Mecham, W.J., et al., "An Analysis of Factors Influencing the Reliability of Retrievable Storage Canisters for Containment of Solid High-Level Radioactive Waste", Argonne National Laboratory, ANL-76-82, (1976).

The reliability of stainless steel type 304L canisters for the containment of HLW in glass and calcine forms was studied. The properties of stainless steel Type 304L, fission product oxides, calcine, and glass were reviewed, and mechanisms of corrosion were identified and studied. The modes of corrosion important for reliability were stress-corrosion cracking, internal pressurization of the canister by residual impurities present, intergranular attack at the waste-canister interface, and potential local effects due to migration of fission products. The key role of temperature control throughout canister lifetime is considered, together with interactive effects. Methods of ameliorating adverse effects and ensuring high reliability are identified and described.

RETRIEVAL

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19.007

Stinebaugh, R.E., "Retrieval of Canistered Experimental Waste at the Waste Isolation Pilot Plant", Sandia Laboratories, SAND-79-1239, (August 1979), 37 p.

This document describes the equipment and operational procedure to be used in the recovery of waste canisters at the WIPP. While methods proposed are specific to the WIPP experimental area and to canistered experimental waste, the application of the outlined procedures to perform retrieval on other waste types at other sites may be feasible.

see also

11.014
15.004
20.002

14.019
18.004
22.002

14.035
18.012

20. Dimensions/Weight20.001

Allard, B. "The Swedish Nuclear Fuel Safety Project for Long-Term Disposal of HLW", Oak Ridge National Laboratory, Transactions of the American Nuclear Society, Vol. 32, (1979), pp 162-164.

This paper briefly summarizes the Swedish program for long-term disposal of HLW. Backfill materials discussed include bentonite/quartz mixture and pressurized bentonite. Various canister materials and their properties are tabulated. The canister materials considered are titanium-lead, stainless steel (for vitrified reprocessed nuclear fuel), copper-lead (for unprocessed spent fuel), and alumina. Estimates on the lifetime of several canister designs due to corrosion are discussed. Tabulated are values for the distribution coefficients and values retention factors in granitic rock as a function of various elements.

20.002

Bergstrom, J., et al., "Fabrication of Copper Canisters for Final Storage of Spent Nuclear Fuel", KBS Technical Reports 1-120, Report No. 81, (1977-1978).

This report describes the manufacturing method for a copper canister, which is proposed by the Swedish KBS program for the disposal of spent fuel. Alternative casting and forging methods are discussed and compared. The design and dimensioning of the canister with respect to stress is considered.

20.003

Heimerl, W., "Techniques for High-Level Waste Solidification in Europe", DWK, Scientific Basis for Nuclear Waste Management, Vol. 1, McCarthy, G.J., (ed), Plenum Press, New York (1979), pp 21-29.

A review of progress toward the development and demonstration of a solidification process for HLW solutions on an industrial scale in four European countries (France, United Kingdom, Germany, and Belgium) is given. Three of the processes that are of major importance are described (AVM-France, HARVEST-United Kingdom, PAMELA-Germany), and data on the resulting solid waste forms are presented.

20.004

Swedish Institute for Metals Research, "Fabrication of Lead-Titanium Canister", KBS Technical Reports 1-120, Report No. 34, (1977-1978).

A proposed design for a titanium-lead canister, its manufacturing process, and cost are discussed.

20.005

Williams, L.D., et al., "Anticipated Developments in Packaging and Transport Systems for Radioactive Materials", Battelle Pacific Northwest Laboratories, Management of Wastes from the LWR Fuel Cycle, Denver, Colorado, July 11-16, 1976, CONF-760701, (July 1976), pp 475-485.

Conceptual package designs for solidified HLW, cladding hulls, plutonium from plutonium-recycle fuel, and radioactive gases are reviewed and described. Economic and safeguards considerations are discussed.

20.006

Wurm, K.J., "Conceptual Design of Packaging For Disposal of Spent Pressurized Water Reactor Fuel Assemblies in a Salt Formation Repository", Ohio State University, Department of Nuclear Engineering, MS Thesis, (1978).

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This thesis was an early attempt to design a container for disposal of an unprocessed PWR spent fuel assembly in salt. Handling and accident stresses, and corrosion were considered in determining the required wall thickness. Various canister geometric options and from one to five fuel assemblies per canister were analyzed. Criticality for normal emplacement and handling accidents was determined. A metallic coating on the liner was utilized to reduce the corrosion rate. The container was designed for 25-yr retrievability.

see also

11.025	12.033	12.090
12.095	12.101	12.115
13.003	13.010	13.017
14.007	14.013	14.016
14.024	14.045	15.008
16.033	16.037	16.047
16.065	18.003	18.004
18.012	19.003	22.005
22.008		

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21. Radiation Levels and Effects21.001

Allard, B., et al., "Radiolysis of Filler Material", KBS Technical Reports 1-120, Report No. 56, (1977-1978).

The effects of gamma irradiation on the properties of the potential buffer material 10 percent bentonite/90 percent quartz are discussed. Also discussed are changes in ion exchange properties and lattice structure of the clay mixture, production of hydrogen as a function of the radiation dose, and the corrosivity of titanium in the presence of radiation.

21.002

Antonini, M., et al., "Atomic Displacements and Radiation Damage in Glasses Incorporating HLW", Joint Research Centre, Italy, Scientific Basis for Nuclear Waste Management, Vol. 2., Northrup, C.J.M., Jr., (ed), Plenum Press, New York, (1980), pp 127-135.

Recent data of radiation effects in borosilicate glasses and amorphous SiO_2 are presented and discussed in terms of atomic displacements produced by direct nuclear collisions and ionization damage produced by interactions of the bombarding particles with the bonding electrons.

21.003

Antonini, M., et al., "Simulations of Radiation Damage in Glasses", Joint Research Center, Italy, Ceramics in Nuclear Waste Management, Chikalla, T.D., and Mendel, J.E., (eds), Cincinnati, Ohio, April 30 - May 2, 1979, CONF-790420, (1979), pp 289-293.

This report discusses the simulation of the damage produced in glasses by alpha emitters using fission fragments and accelerated nickel ions. The experiments were conducted with a type of borosilicate glass commonly used in Europe. Into the glass matrix has been incorporated 20 percent of simulated waste oxide. The amount of stored energy in a test sample is dependent on temperature of irradiation, and on the effect of radiation annealing caused by large amounts of ionization energy dissipated by fission fragments. Dose rate does not seem to have much influence on stored energy.

21.004

Barney, G.S., and Wood, B.J., "Identification of Key Radionuclides in a Nuclear Waste Repository in Basalt", Rockwell Hanford Operations, RHO-BWI-ST-9, (May 1980), 20 p.

Radionuclides are identified which appear to pose the greatest potential hazard to man during long-term storage of nuclear waste in a repository mined in the Columbia plateau basalt formation. Criteria used to select these are: quantity of stored waste, biological toxicity, leach rate of various wastes into groundwater, and transport rate via groundwater flow. Waste forms are assumed to be either unprocessed spent fuel or borosilicate glass containing reprocessed HLW. Radionuclides are ranked according to quantity, toxicity, and release rate from the repository. Hazardous nuclides of importance are technetium-99, iodine-129, neptunium-237, radium-226, lead-207, thorium-230, lead-210, tin-126, selenium-79, and plutonium-242.

21.005

Barton, G.B., "Solidification of High Level Wastes, Pt. IV. Phosphate Melts for Fixation of Radioactive Residues from Purex Type Wastes-Three to Fifty Per Cent Fission Product Oxides", Battelle Pacific Northwest Laboratories, BNWL-80, (July 1965), 39 p.

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The possible composition range for the HLW generated from various power reactor fuels processed through a Purex-type process is very broad. A partial survey of this wide range was conducted in an effort to establish the composition limits that bound the operable range of the phosphate melt system. Two screening criteria were used: (a) drip temperature, an empirical measure of flow temperature at which a container could be filled, and (b) solubility. Measurements indicated the occurrence of two (or possibly more) optimum phosphate additions near M_2O/P_2O_5 ratios of 2.1 and 1.0. It was found that fission products equivalent to at least 50 percent of the weight of oxides in the waste can be successfully incorporated in a phosphate melt that can be transferred as a fluid into containers at less than 900°C.

21.006

Bryan, G.H., et al., "High-Level Waste Vitrification Product Characterization", Battelle Pacific Northwest Laboratories, Scientific Basis for Nuclear Waste Management, Vol. 2, Northrup, C.J.M., Jr., (ed), Plenum Press, New York, (1980), pp 147-154.

Non-destructive testing was conducted on two 20.3 cm diameter by 2.44 m long canisters. A schematic view of the waste canister is provided. Measurements of the weight and height of fill, external canister dimensions and wall thickness, gross gamma measurement, axial gamma scan, canister internal thermocouple readings, external temperature scan by infrared camera, and calorimetry are provided. Further tests involving mechanical conditions and leaching are called for.

21.007

Christensen, H., "Evaluation of Radiolysis in Groundwater", KBS Technical Reports 1-120, Report No. 78, (1977-1978).

Computer calculations of the radiolysis of groundwater either inside or outside the canister are reported. The purpose was to determine whether hydrogen could be formed in concentrations exceeding the solubility limit and also to calculate the extent of corrosion of a copper canister.

21.008

Clinard, F.W., Jr., and Hurley, G.F., "Effects of Irradiation on Structural Properties of Crystalline Ceramics", Los Alamos Scientific Laboratory, Ceramics in Nuclear Waste Management, Chikalla, T.D., and Mendel, J.E., (eds), Cincinnati, Ohio, April 30 - May 2, 1979, CONF-790420, (1979), pp 300-304.

Stability of crystalline ceramic nuclear waste may be degraded by self-irradiation. Changes in the parameters of strength, density, thermal conductivity, and lattice structure are of concern. The structural damage of ceramics under various radiation conditions is discussed with regard to the possible effects on the waste form. Characteristics are listed which enhance irradiation stability of the waste ceramic.

21.009

Dran, J.C., et al., "A Microscopic Approach for the Simulation of Radioactive Waste Storage in Glass", Laboratoire Rene Bernas, Scientific Basis for Nuclear Waste Management, Vol. 2, Northrup, C.J.M., Jr., (ed), Plenum Press, New York, (1980), pp 135-140.

This paper discusses the previously overlooked effects of alpha-recoils on prospective waste glasses. The examination of the glass samples on a microscopic level appears to be a very powerful tool in the study of these alpha-recoil effects. Three types of glasses were studied, differing mostly with respect to the content of SiO_2 . This new method of investigation

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allows for a more rapid evaluation of a great number of possible combinations of composition and disposal conditions.

21.010

Ewing, R.C., and Haaker, R.F., "The Metamict State: Implications for Radiation Damage in Crystalline Waste Forms", University of New Mexico, Nuclear and Chemical Waste Management, Vol. 1, No. 1, (1980), pp 51-58.

The pertinent literature on metamictization is summarized. This includes; the properties of the metamict state, the methods of analysis of metamict minerals, a summary of observed metamict phases, and the rate of metamictization. Although structural stability and total alpha particle dosage are important in determining the rate of metamictization, absorbed water, unusual composition or alpha particle dose rate might result in anomalous radiation damage effects. It is recommended that (1) radiation damage effects in natural materials must be evaluated, (2) factors affecting the rate of metamictization must be determined, and (3) a careful analyses of the literature and additional experimental work on zircons must be completed.

21.011

Guber, W., et al., "Lab-Scale and Pilot-Plant Experiments on the Solidification of High-Level Wastes at the Karlsruhe Nuclear Research Centre", Nuclear Research Center, Karlsruhe, Management of Radioactive Wastes from the Nuclear Fuel Cycle, Vol. 1, (CONF-760310), STI/PUB/433, International Atomic Energy Agency, Vienna, (1976), pp 271-286.

This report primarily covers solidification processes, and presents the long-term effects of alpha-radiation on leaching and radiolysis/gas generation of the product.

21.012

Haaker, R.F., and Ewing, R.C., "The Metamict State Radiation Damage in Crystalline Materials", University of New Mexico, Ceramics in Nuclear Waste Management, Chikalla, T.D., and Mendell, J.E., (eds), Cincinnati, Ohio, April 30 - May 2, 1979, CONF-790420, (1979), pp 305-309.

Metamict minerals provide an excellent basis for the evaluation of long-term radiation damage effects, particularly such changes in physical and chemical properties as microfracturing, hydrothermal alteration, and solubility. This paper presents a summary of pertinent literature on metamictization and proposes experiments that are critical to the evaluation of structural controls on radiation damage in crystalline phases.

21.013

Jenks, G.H., and Bopp, C.D., "Energy Storage in High Level Radioactive Waste and Simulation and Measurement of Stored Energy with Synthetic Wastes", Oak Ridge National Laboratory, ORNL-TM-3781, (January 1973), 50 p.

The oxides which contain the HLW in a repository will be exposed to a variety of nuclear radiations including those formed in the alpha-disintegration of transuranic elements. Radiation damage in oxides and silicates will result mainly from displacements or spikes caused by elastic collisions of the heavy recoiling nuclei. Temperatures greater than 1000°C may be needed to release stored energy with some waste types. Radiation damage in wastes can be simulated.

21.014

Lam, D.J., et al., "Bonding of Metal Oxides in Sodium Silicate Glass", Argonne National Laboratory, Scientific Basis for Nuclear Waste Management, Vol. 1, G.L. McCarthy (ed), Plenum Press, New York, (1979), pp 97-107.

The silicate glasses have remarkably high solubility for numerous metal oxides from all regions of the periodic table. This universality is particularly important for waste disposal since many active nuclear species with great variation in atomic number must be stored. Because of the long-term stability requirements of the disposed glasses, fundamental information relating to the bonding properties of metal oxides in the glass are of importance. This report presents such data for iron and uranium oxides in silicate glasses.

21.015

Lundgren, K., "Radiation Level and Radiant Energy Imparted to Water Outside of Canisters in the Final Repository", KBS Technical Reports 1-120, Report No. 106, (1977-1978).

The radiation field outside of a capsule of copper containing approximately 500 spent BWR fuel rods has been calculated. An average burnup of 30,000 MWD/MTU and a cooling time of 40 years have been assumed. Interacting with water, ionizing radiation produces free radicals and oxygen. The extent of copper corrosion resulting from the produced radicals and oxygen is also discussed.

21.016

Malow, G., and Andresen, H., "Helium Formation from Alpha-Decay and Its Significance for Radioactive Waste Glasses", Hahn-Meitner-Institut, Scientific Basis for Nuclear Waste Management, Vol. 1, McCarthy, G.J., (ed), Plenum Press, New York, (1979), pp 109-115.

To produce high radiation damage and excessive helium concentrations, the B-10 (n, α) Li-7 nuclear reaction was applied to a zinc-borosilicate glass. The number of displaced atoms was estimated by using a collision

model. The fraction of displaced atoms in glasses containing actinide elements was calculated for α , β , and γ decay, and spontaneous fission. Helium diffusion was measured between 120 and 650°C and was found to be dependent upon the α dose up to 500°C. With respect to the long-term situation, helium release rates and saturation radiation damage were estimated.

21.017

McDonell, W.R., and Dillich, S., "Effects of Radiations from Transuranium Nuclides on Container Surfaces", Savannah River Laboratory, DP-MS-78-9, (1978), 60 p.

Container materials for short-lived transuranium nuclides such as curium-242 and californium-252 may be implanted with sufficiently high concentrations of helium to cause surface blistering and other deleterious effects in some environments. Conditions leading to the formation of such blisters were studied and are presented in this report.

21.018

McElroy, J.L., and Mendel, J.E., "Research and Development Activities: High-Level Waste Immobilization Program. Quarterly Progress Report", Battelle Pacific Northwest Laboratories, PNL-3050-1, (November 1979).

Accelerated radiation damage effects on vitreous and two partially devitrified forms of curium-244-doped 77-260 simulated waste glass are reported. Data on the effect of radiation on the leach rate of the glass is presented for the Soxhlet, pH-4, and pH-9 leach tests. Tests have been started on the corrosion of 304L stainless steel in varying water temperature and chemistry conditions. Six samples each of 76-68 simulated waste glass, PW-9 calcine sintered ceramic, PW-9 first sintered ceramic, and PW-9 cement were impact tested with an impact energy of 160 ft-lb. The weight percent of fines less than 38 microns is reported for each of the four waste forms.

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21.019

Platt, A.M., "Quarterly Progress Report - Research and Development Activities - Waste Fixation Programs", Battelle Pacific Northwest Laboratories, BNWL-1741, (April 1973), 38 p.

This is the second report of a series; the first was BNWL-1699. This report describes a shift from the accelerated leach test to the Soxhlet leach test. Radiation damage to borosilicate glass is discussed where the alpha particle emission rates were calculated from actinide concentrations in waste. During the first 100 years after reprocessing, approximately 1.2×10^{23} and 3.0×10^{24} alphas/MTHM would be emitted from the natural decay of actinides in the waste from LWR UO₂ fuel and LWR plutonium recycle fuel. Tabulated are data for total neutron emission from waste in borosilicate glass during the first 100 years after reprocessing.

21.020

Ringwood, A.E., et al., "Effect of Radiation Damage on SYNROC", Australian National University, Scientific Basis for Nuclear Waste Management, Vol 2., Northrup, C.J.M., Jr., (ed), Plenum Press, New York, (1980), pp 273-280.

The effects of alpha radiation on SYNROC are discussed. The majority of data comes from the study of natural zirconolites and perovskites. These studies show that both compositions appear to provide adequate protection to radiation damage.

21.021

Turcotte, R.P., "Radiation Effects in Solidified High Level Wastes. Part 2. Helium Behavior", Battelle Pacific Northwest Laboratories, BNWL-2051, (1976), 47 p.

An experimental investigation of helium diffusion in a simulated zinc borosilicate nuclear waste glass is described. The helium inventory was generated by curium-244 doping, and the influence of damage from alpha radiation was demonstrated. Canister pressurization as a result of helium generation as a function of storage time, temperature, and glass geometry is discussed.

21.022

Walker, C.T., and Riege, U., "Compatibility of Actinides with HLW Borosilicate Glass: Solubility and Phase Formation", Commission for European Communities, Ceramics In Nuclear Waste Management, Chikalla, T.D., and Mendel, J.E. (eds), Cincinnati, Ohio, April 30 - May 2, 1979, CONF-790420, (1979), pp 198-202.

For a stable HLW glass product, the actinides in HLW should be homogeneously dissolved in the glass. The formation of actinide-containing phases is undesirable because they may adversely affect the properties of the glass, e.g., increasing leach rates, promoting devitrification, or as a result of helium production, decreasing the mechanical strength of the glass. This report presents some data on PuO_2 solubility in borosilicate glass.

21.023

Weber, W.J., et al., "Radiation Effects in Vitreous and Devitrified Simulated Waste Glass", Battelle Pacific Northwest Laboratories, Ceramics in Nuclear Waste Management, Chikalla, T.D., and Mendel, J.E., (eds), Cincinnati, Ohio, April 30 - May 2, 1979, CONF-790420, (May 1979), pp 294-299.

The long-term radiation stability of vitreous and partially devitrified forms of HLW glass was investigated in accelerated experiments by curium-244 doping. The effects of radiation on microstructure, phase behavior, density, impact strength, stored energy, and leachability are

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reported to a cumulative radiation dose of 5×10^{18} alpha decays/cm³.
This dose produces saturation of radiation effects in most properties.

see also

11.009	12.003	12.015
12.017	12.018	12.031
12.040	12.055	12.063
12.064	12.065	12.066
12.073	12.090	12.094
12.098	12.099	12.101
12.104	12.111	12.114
12.115	12.121	12.122
12.126	12.129	12.138
12.143	12.165	13.001
13.004	13.008	13.014
14.007	14.011	14.012
14.017	14.031	14.033
14.037	16.020	16.021
16.022	16.023	16.025
16.027	16.031	16.032
16.033	16.034	16.037
16.043	16.044	16.046
16.050	16.053	16.061
17.003	18.003	18.005
18.010	20.006	21.011
21.016	21.017	22.005
22.011		

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22. Nuclide Retardation

22.001

Ahlstrom, P.E., "Ceramic and Pure-Metal Canisters", Kaernbraeuslesakerhet, Nuclear and Chemical Waste Management, Vol. 1, No. 1, (1980), pp 77-88.

The system proposed by KBS Swedish project for the isolation of nuclear waste is described. Special attention is given to the buffer material with desired characteristics described in full. The KBS has chosen a Volclay MX-80 bentonite as a reference material. The MX-80 is a sodium bentonite containing 90 percent montmorillonite. Physico-chemical properties of the MX-80 such as density, permeability coefficient, and thermal conductivity are reported. The proposed lead-titanium canister for vitrified waste is also described.

22.002

Coons, W.E., et al., "The Functions of an Engineered Barrier System for a Nuclear Waste Repository in Basalt", Rockwell Hanford Operations, RHO-BWI-LD-23, (January 1980).

Defined in this document are the functions of components selected for an engineered barrier system for a nuclear waste repository in basalt. A five-component barrier system, comprised of waste form, canister, buffer, overpack, and tailored backfill, is discussed in terms of effective operation throughout the course of repository history, recognizing that the emplacement environment changes with time. The operating philosophy of the conceptual engineered barrier system is clarified by examples pertinent to storage in basalt, and a technical approach to barrier design and material selections is proposed. A method for system validation and qualification is also included which considers performance criteria proposed by external agencies in

conjunction with site-specific models and risk assessment to define acceptable levels of system performance.

22.003

Engelbrektsson, A., et al., "Fabrication and Handling of Bentonite Blocks", KBS Technical Reports 1-120, Report No. 84, (1977-1978).

The rationale for employing 10 percent bentonite/90 percent quartz as a backfill material is provided. Fabrication techniques and methods for emplacement of the backfill in a repository are explained.

22.004

Jacobsson, A., and Pusch, R., "Deposition of High-Level Waste in Boreholes Containing Buffer Material", KBS Technical Reports 1-120, Report No. 3, (1977-1978).

Much of the available literature on the physicochemical stability of the 10 percent bentonite/90 percent quartz backfill material proposed by the Swedes is compiled. Specifically addressed is the: ability of the backfill to mechanically support heavy containers; homogeneity of the backfill material; permeability and ion-adsorption ability of the 10 percent bentonite/90 percent quartz mixture; and heat conduction properties of the backfill.

22.005

Jacobsson, A., and Pusch, R., "Deposition of High-Level Waste in Tunnels Containing Buffer Material", KBS Technical Reports 1-120, Report No. 4 (1977-1978).

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The desirable properties of backfill material and canisters for HLW are discussed. Relative advantages and disadvantages of a concrete container and a 10 percent bentonite/90 percent quartz backfill material are also reported.

22.006

Komarneni, S., and Roy, R., "SUPEROVERPACK: Tailor-Made Mixtures of Zeolites and Clays", Pennsylvania State University, Scientific Basis for Nuclear Waste Management, Vol. 2, Northrup, C.J.M, Jr., (ed), Plenum Press, New York, (1980), pp 411-418.

There are three distinct roles for highly sorbing mineral assemblages capable of absorbing the typical mixture of ions in HLW. These roles are as the overpack around the container surrounding the solid waste form, as a primary sorber of all the ions in the waste, and as the principal active agent for absorbing radionuclides in cases of leaks or accidents either before or after emplacement in a repository. Distribution coefficients have been determined for cesium, rubidium, strontium, and uranium for the following mineral phases: conasauga shale, montmorillonite, vermiculite, gibbsite, clinoptilolite, mordenite, erionite, chabazite, phillipsite, and Linde 3A, 4A, and 5A. In addition, similar measurements have been made for some twenty binary mixtures, including several selected mixtures which are especially highly absorbing.

22.007

LeBell, J.C., "Colloid Chemical Aspects of the Confined Bentonite Concept", KBS Technical Reports 1-120, Report No. 97, (1977-1978).

The basic principles of coagulation and swelling by electrolytes and swelling pressure are discussed. A literature review on the colloidal

characteristics of bentonite, which is to be used as a backfill material, is also provided. The swelling of confined bentonite was studied in an artificial crack of 0.5 mm width.

22.008

Lester, D.H., et al., "Waste Package Analysis", Science Applications, Inc., National Waste Terminal Storage Program Information Meeting, Columbus, Ohio, December 9-11, 1980, ONWI-212, (December 1980), pp 328-334.

A waste package performance model is described. Barrier performance is determined in terms of time of initial release of radionuclides to the geosphere and duration of radionuclide release. Expected performance results of about 60 package designs are tabulated for the following parameters: media, stabilizer material, canister thickness, sleeve material, sleeve thickness, backfill material, backfill thickness, maximum waste temperature, leach begin time, release begin time for plutonium and release end time for plutonium. A flow chart for the waste package performance model is given. The results are intended for use in the NWTs program to guide program plans for waste package development.

22.009

Nowak, E.J., "The Backfill as an Engineered Barrier for Nuclear Waste Management", Sandia Laboratories, Scientific Basis for Nuclear Waste Management, Vol. 2, Northrup, C.J.M., Jr., (ed), Plenum Press, New York, (1980), pp 403-410.

The backfill surrounding radioactive wastes emplaced in a geologic repository can be an effective engineered barrier to retard the migration of radioactive waste species. The backfill can be designed to limit the rate of waste migration by reducing the flow of groundwater and by physically or

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chemically reacting with waste species. Measurements of the batch absorption properties of clays, soils, and a zeolite were conducted using europium-152 dissolved in nearly saturated brines to simulate transuranic contaminated groundwater.

22.010

Pusch, R., "Water Uptake in a Bentonite Buffer Mass", University of Lulea, KBS Technical Reports 1-120, Report No. 23, (1977-1978).

Water uptake characteristics of the 10 percent bentonite/90 percent quartz backfill material are reported. Water uptake is important because the thermal conductivity of the backfill material depends on its water content. The effect of near-field temperature on the water content of the backfill is also discussed.

22.011

Pusch, R., and Jacobsson, A., "Properties of Bentonite-Based Buffer Material", University of Lulea, KBS Technical Reports 1-120, Report No. 32, (1977-1978).

Some chemical and biological properties of the bentonite-based buffer material were investigated. The temperature and pressure stability of bentonite was reviewed from the geological literature. The ability of bentonite to support biological activity and the mobility and activity of bacteria within the bentonite/quartz buffer were also evaluated.

22.012

Pusch, R., "Highly Compacted Na Bentonite as Buffer Substance", University of Lulea, KBS Technical Reports 1-120, Report No. 74, (1977-1978).

This report describes the results of a number of investigations of mechanical and physical properties of 10 percent bentonite/90 percent quartz backfill material, compacted to a bulk density of 2.0-2.3 tons/m³.

22.013

Pusch, R. "Highly Compacted Sodium Bentonite for Isolating Rock-Deposited Radioactive Waste Products", University of Lulea, Nuclear Technology, Vol. 45, (September 1979), pp 153-157.

Sodium saturated bentonite clay compacted to a high density is very efficient barrier for preventing radioactive wastes from reaching the biosphere. The bentonite, emplaced in the form of blocks between the rock and the canisters in large boreholes, is used to provide a practically impervious barrier. The bentonite blocks take up water and swell, completely filling the space between the rock and the canister. The swelling potential, which is permanent, makes the bentonite self-sealing. This means rock joints which may be opened are sealed by extruding bentonite. The swelling and the permeability of the bentonite are discussed. The swelling pressure will be in the order of 10 MPa at a maximum probably dropping to 5 MPa. The permeability coefficient is on the order of 5×10^{-13} to 10^{-14} which makes it practically impervious.

22.014

Strickert, R., et al., "The Sorption of Technetium and Iodine Radioisotopes by Various Minerals", Argonne National Laboratory, Nuclear Technology, Vol. 49, (July 1980), pp 253-266.

Radioisotopes of technetium and iodine were found to be strongly sorbed from aqueous solutions by the naturally occurring minerals bournonite, chalcopyrite, pyrite, tennantite, and tetrahedrite. Tables comparing the K_D

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values for the above minerals for TcO_4 , I, and IO_3 against the K_d values for basalts, shales, granites, and tuffs, and limestone, siltstone, dolomite, chalk, witherite, anhydrite, and apatite are included.

22.015

Wood, B.J., "Backfill Performance Requirements, Estimates from Transport Models", Rockwell Hanford Operations, RHO-BWI-SA-58, (November 1980).

Compressed clay as a backfill is considered a material for meeting the 1,000-yr waste package containment criterion established in 10CFR60. Water ingress during the early life of the repository should be very limited. Appropriate containment for such isotopes as iodine-129 will require the support of barriers in addition to the canister, overpack, and waste form. A conservative, one-dimensional transport model has been used to determine maximum permissible release rates from the waste package for each isotope after the 1,000 year period. From this, the most important nuclides within spent fuel located in a repository in basalt at Hanford have been determined to be iodine-129, selenium-79, technetium-99, lead-107, neptunium-237, plutonium-239, -240, -242, americium-241, -243, and members of the uranium-238 decay chain. Backfill can operate as a diffusion-controlled retardation barrier for most of the key radionuclides under conditions of relatively rapid water flow. The maximum permissible release rates for the waste package have been translated into concentrations at the inner and outer surfaces of the backfill under low- and high-flow conditions.

see also

12.135	13.003	13.015
14.031	16.066	16.067
18.011	20.001	21.001
20.004	23.003	23.008

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23. Criticality23.001

Allen, E.J., "Criticality Analysis of Aggregations of Actinides from Commercial Nuclear Waste in Geological Storage", Oak Ridge National Laboratory, ORNL-TM-6458, (August 1978).

An underground nuclear-waste disposal facility must be designed to avoid a nuclear criticality event. A computational model that performs criticality calculations rapidly and efficiently is presented. Critical mass calculations are made for five waste types, five waste ages, five actinide elements, and four geologic compositions. The possibilities for criticality in undisturbed waste packages and later after packages have been breached and radionuclides have begun to migrate are considered.

23.002

Behrenz, P., and Hannerz, K., "Criticality in a Spent Fuel Repository in Wet Crystalline Rock", KBS Technical Reports 1-120, Report No. 108, (1977-1978).

This report considers the risks associated with the assembly of fissile radionuclides into critical configurations in a repository. Scenarios leading to the formation of critical configurations are discussed. The environmental impact of a criticality reaction in the deep repository is also assessed.

23.003

Brookins, D.G., "Geochemical Constraints on Accumulation of Actinide Critical Masses from Stored Nuclear Waste in Natural Rock Repositories", University of New Mexico, ONWI-17, (December 1978).

This study addresses the possibility that critical masses of actinides may form after radioactive waste has been buried in a geologic repository. The relative mobilities of lanthanides and actinides in groundwater are discussed in relation to the possibility that fissile actinides might segregate from the lanthanides and form a critical mass in the geologic bedrock. The conditions that must prevail in order for critical masses to congregate and the likelihood of such occurrences are discussed.

23.004

Gore, B.F., et al., "Criticality Safety Considerations in the Geologic Disposal of Spent Nuclear Fuel Assemblies", Battelle Pacific Northwest Laboratories, PNL-3268, (May 1980), 46 p.

This study identifies and addresses problems encountered in assessing the criticality potential of spent fuel disposal in geologic repositories. The effect of changes in the shape and form of the spent fuel, the effect of water intrusion into the container, and the effect of selective leaching of spent fuel constituents on the potential for criticality in the geologic repository are discussed. Also discussed is whether criticality calculations should be based on fresh fuel or fuel with some assumed burnup.

23.005

Heaberlin, S.W., and Selby, G.P., "Effect of Fissile Isotope Burnup on Criticality Safety for Stored Disintegrated Fuel Rods", Battelle Pacific Northwest Laboratories, PNL-2589, (September 1978), 18 p.

Fuel exposures from 0 to 20,000 MWD/MTU were included. Isotopic concentrations were generated with the ORIGEN computer code. The fuel form for the disintegrated rods was assumed to be the fissionable isotopes as oxides mixed with water. For each burnup level the optimum degree of water

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moderation was determined with the EGGNIT-II code using ENDF/B cross section data. In use, the spent fuel canister will be surrounded by some shielding material. Four reflectors were considered: water, concrete, and Hanford basalt, with either 30 weight percent or 40 weight percent water content.

23.007

McNair, G.W., and Gore, B.F., "Feasibility of Close-Packing Fuel Rods for Geologic Disposal of Spent Fuel," Battelle Pacific Northwest Laboratories, Transactions of the American Nuclear Society, Vol. 34, (1980), pp 406-407.

This paper addresses criticality safety of spent fuel assemblies in geological media by close packing the fuel rods within a canister, thus limiting the water/fuel volume. Calculations were performed using a one-dimensional HFN diffusion theory code to predict the critical radius of the fuel pin region. Calculations were also performed to assess the effects of fuel pellet disintegration on criticality. All of these results are presented in a graph of critical cylinder radius vs. water/fuel volume ratio.

23.008

Neretnieks, I., "Some Notes in Connection with the KBS Studies of Final Disposal of Spent Fuel", Kungliga Tekniska Hogskolan, KBS-TR-120, (September 1978).

This report is a compilation of four short notes on problems studied in connection with the KBS investigation on final disposal of spent unprocessed fuel. The first and the third notes are of particular interest to a waste package designer. The first note discusses the possibility for convection flow of water in the buffer material; the third note discusses the time required to accumulate a critical mass of uranium in the buffer material.

23.009

Vreeland, J.A., and Jain, R., "Criticality Considerations for Geological Disposal of Spent Fuel", California State University, Transactions of the American Nuclear Society, Vol. 33, (1980), pp 435-436.

This brief paper discusses the development of an analytical model to predict criticality resulting from leaching of uranium or plutonium from a spent fuel assembly and the difference in mobility of the two elements. An example of such an event occurring is presented. The event is initiated by water flow through a simulated buried spent fuel assembly producing an increase in the fissile enrichment through the leaching/mobility process resulting in the formation of a critical mass.

see also

13.017

14.007

16.006

17.002

17.003

18.010

20.006

<u>Report Number</u>	<u>Abstract Number</u>
AAEC/E-364	No abstract
AECL-5249	No abstract
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AERE-R-4685	14.002
AERE-R-9188	12.017
AERE-R-9343	14.010
AERE-R-9417	16.021
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ANL-76-82	19.006
ANL-76-110	No abstract
ANL-77-36	14.042
ANL-78-11	12.139
ANL-78-19	14.017
ANL-78-34	16.059
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ORNL/SUB-3706/5	15.006
ORNL/SUB-3706/6	14.034
ORNL/SUB-4269/1	15.009
ORNL/SUB-4269/2	15.010
ORNL/SUB-4269/3	14.035
ORNL/SUB-4269/4	15.004
ORNL/SUB-4269/5	No abstract
ORNL/SUB-4269/7	14.008
ORNL-TM-882	No abstract
ORNL-TM-1081	No abstract
ORNL-TM-2971	18.015
ORNL/TM-3688	No abstract
ORNL-TM-3717	No abstract
ORNL-TM-3781	21.013
ORNL-TM-4481	No abstract
ORNL-TM-4639	12.013
ORNL/TM-5462	No abstract
ORNL/TM-5863	No abstract
ORNL/TM-6159	No abstract
ORNL/TM-6308	11.020
ORNL/TM-6404	12.001
ORNL/TM-6458	23.001
ORNL/TM-6573	No abstract
ORNL/TM-6863	No abstract
ORNL-TM-6866	No abstract
ORNL-TM-7013	12.009
ORNL-TM-7253	No abstract
ORNL-TR-4302	No abstract
ORNL-TR-4626	12.147
ORNL-TR-4629	12.148
ORNL-TR-4632	12.069
ORP/TAD-78-2	16.019
PB-265-197	No abstract
PNL-2247	14.046
PNL-2264	15.007
PNL-2265-1	12.094
PNL-2265-2	12.095
PNL-2278	12.021
PNL-2377-3	No abstract
PNL-2378-3	No abstract
PNL-2378-4	No abstract
PNL-2379	No abstract
PNL-2451	No abstract
PNL-2456	11.022

<u>Report Number</u>	<u>Abstract Number</u>
PNL-2481	12.124
PNL-2589	23.005
PNL-2590	No abstract
PNL-2598	No abstract
PNL-2625	12.125
PNL-2642	12.030
PNL-2664	12.022
PNL-2668-1	16.049
PNL-2668-2	15.012
PNL-2751	No abstract
PNL-2759	12.160
PNL-2765-4	12.096
PNL-2776	No abstract
PNL-2904	16.008
PNL-2954	15.003
PNL-2982	12.066
PNL-2990	13.016
PNL-2999-1,2,3,4	No abstract
PNL-3000-1	No abstract
PNL-3000-2	No abstract
PNL-3000-3	No abstract
PNL-3000-4	12.114
PNL-3000-7	No abstract
PNL-3035	16.050
PNL-3036	15.013
PNL-3038	No abstract
PNL-3050-1	21.018
PNL-3050-2	12.097
PNL-3060	16.046
PNL-3078	No abstract
PNL-3094	12.014
PNL-3152	12.023
PNL-3181	No abstract
PNL-3268	23.004
PNL-3318	12.103
PNL-3367	14.025
PNL-SA-6416	12.067
PNL-SA-7072	12.115
PNL-SA-7292	13.013
PNL-SA-7599	12.129
PNL-TR-377	No abstract
RFP-2749	No abstract
RHO-BWI-80-100-19	No abstract
RHO-BWI-C-2	13.010
RHO-BWI-C-12	14.028
RHO-BWI-C-16	No abstract
RHO-BWI-C-33	14.039
RHO-BWI-LD-11	No abstract
RHO-BWI-LD-23	22.002
RHO-BWI-SA-1	No abstract

<u>Report Number</u>	<u>Abstract Number</u>
RHO-BWI-SA-7A	No abstract
RHO-BWI-SA-8A	No abstract
RHO-BWI-SA-12A	No abstract
RHO-BWI-SA-15	14.029
RHO-BWI-SA-19	No abstract
RHO-BWI-SA-58	22.015
RHO-BWI-ST-7	13.015
RHO-BWI-ST-9	21.004
RHO-C-12	12.087
RHO-C-13	17.001
RHO-CD-568	10.008
RHO-CD-711	10.013
RHO-CD-772	10.014
RHO-LD-77-3D	No abstract
RHO-LD-79-83-1Q	No abstract
RHO-LD-79-101-D	No abstract
RHO-LD-83-2Q	No abstract
RHO-LD-95	15.015
RHO-SA-24A	No abstract
RHO-SA-102	No abstract
RHO-SA-117	12.059
RHO-SA-141	10.002
RHO-ST-10	10.009
RHO-ST-33	No abstract
SAI-OR-629-1	10.015
SAND-75-5907	No abstract
SAND-77-0369	No abstract
SAND-77-1237C	No abstract
SAND-77-1257	No abstract
SAND-78-0029	No abstract
SAND-78-0297	No abstract
SAND-78-0566	No abstract
SAND-78-0893	No abstract
SAND-78-0924	No abstract
SAND-78-1289C	14.043
SAND-78-2111	11.004
SAND-79-0172	No abstract
SAND-79-0531	No abstract
SAND-79-1239	19.007
SFPD-T1-R-79-1	10.006
SKBF-KBS-TR-79-01	No abstract
SRO-TWM-77-1	No abstract
TC-1076	No abstract
TID-7699	No abstract
TID-29442	No abstract
UCID-18121	10.003
UCID-18133 (Rev. 1)	No abstract
UCRL-13766	No abstract
UCRL-13804	No abstract
UCRL-15025	19.004

<u>Report Number</u>	<u>Abstract Number</u>
UCRL-15048	No abstract
UCRL-15111	19.001
UCRL-15167	No abstract
UCRL-15168	14.011
UCRL-50050-77	No abstract
UCRL-50050-78	No abstract
UCRL-80850	No abstract
UCRL-81147	12.157
68-WA/NE-13	No abstract
WASH-1339	No abstract
WNRE-466	12.143
Y/OWI/SUB-76/04220	No abstract
Y/OWI/SUB-76/16502	14.047
Y/OWI/SUB-76/16509	19.002
Y/OWI/SUB-78/16549/1	No abstract
Y/OWI/TM-9	10.017
Y/OWI/TM-36/19	No abstract
Y/OWI/TM-36-20	No abstract
Y/OWI/TM-36/23	No abstract
Y/OWI/TM-42	14.019
Y/OWI/TM-43/1	No abstract
Y/OWI/TM-43/2	No abstract
Y/OWI/TM-43/3	No abstract
Y/OWI/TM-43/4	No abstract
Y/OWI/TM-44	18.012
Y/OWI/TM-56	10.010

V. LISTING OF REPORTS/CONFERENCES WITH MULTIPLE PAPERS

<u>Report Number</u>	<u>Report/Conference Title</u>
CONF-660208	Solidification and Long-Term Storage of Highly Radioactive Wastes, Richland, Washington
CONF-731101	American Nuclear Society Winter Meeting, San Francisco, California
CONF-740901	Packaging and Transportation of Radioactive Materials, Miami Beach, Florida
CONF-750542	American Ceramic Society, Washington, DC
CONF-750699	15th Annual International Conference, Ottawa, Canada
CONF-760310	Management of Radioactive Wastes from the Nuclear Fuel Cycle, Vienna
CONF-760701	Management of Wastes from the LWR Fuel Cycle, Denver, Colorado
CONF-770102	Ceramic and Glass Radioactive Waste Forms, Germantown, Maryland
CONF-770159	Properties and Characteristics of High-Level Waste Glass, Washington, DC
CONF-770303	Corrosion/77 NACE Meeting, San Francisco, California
CONF-771021	Energy Sources and Development, Barcelona, Spain
CONF-771208	Treatment, Conditioning, and Storage of Solid Alpha-Bearing Waste and Cladding Hulls, Paris
CONF-780243	Waste Management and Fuel Cycles, Tucson, Arizona
CONF-780316	Waste Management, Tucson, Arizona
CONF-780819	Nuclear Air Cleaning, Boston, Massachusetts
CONF-781241	High-Level Radioactive Solid Waste Forms, Denver, Colorado
CONF-790420	Ceramics in Nuclear Waste Management, Cincinnati, Ohio
CONF-800305	Corrosion/80 NACE Meeting
NUREG/CP-0005	High-Level Radioactive Solid Waste Forms, Denver, Colorado
ONWI-62	National Waste Terminal Storage Program Information Meeting, Columbus, Ohio
ONWI-88	U.S./FRG Bilateral Workshop, Waste Isolation Performance Assessment and In Situ Testing, Berlin
ONWI-212	National Waste Terminal Storage Program Information Meeting, Columbus, Ohio
PNL-SA-7992	Waste Isolation Safety Assessment Program, Task 2 Workshop, Richland, Washington
STI/PUB/156	Disposal of Radioactive Wastes into the Ground, Vienna
STI/PUB/433	Management of Radioactive Wastes from the Nuclear Fuel Cycle, Vienna
STI/PUB/465	Nuclear Power and Its Fuel Cycle, Salzburg
Y/OWI/SUB-77/14268	National Waste Terminal Storage Program Conference on Waste-Rock Interactions, University Park, Pennsylvania

Book/Conference Title

Geologic Disposal of Nuclear Waste, 19th Annual Symposium, University of New Mexico, Albuquerque, New Mexico

High-Level Radioactive Waste Management, Los Angeles, California, Campbell, M.H., (ed). Advances in Chemistry Series 153, American Chemical Society

Management of Wastes from Nuclear Fuel Reprocessing for the Protection of Man and His Environment, 9th World Energy Conference

Radioactive Waste in Geologic Storage, Fried, S., (ed). American Chemical Society Symposium Series 100.

Scientific Basis for Nuclear Waste Management, Boston, Massachusetts, 1978.

Scientific Basis for Nuclear Waste Management, Boston, Massachusetts, 1979.

Waste Management, Post, R.G., (ed), Tucson, Arizona, 1975.

Waste Management, Post, R.G., and Wacks, M.E., (eds), 1978

Waste Management and Fuel Cycles '79, Post, R.G., and Wacks, M.E., (eds), 1979

Waste Management '80, Post, R.G., (ed), 1980