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Annotated Bibliography of Radioactive Waste Management Publications at Pacific Northwest Laboratory January 1978 Through July 1982

Issued by the
Nuclear Waste Technology Program Office

September 1982

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

Pacific Northwest Laboratory
Operated for the U.S. Department of Energy
by Battelle Memorial Institute

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ANNOTATED BIBLIOGRAPHY OF RADIOACTIVE WASTE MANAGEMENT PUBLICATIONS AT PACIFIC NORTHWEST LABORATORY JANUARY 1978 THROUGH JULY 1982

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Pacific Northwest Laboratory
Richland, Washington 99352

PREFACE

This bibliography lists publications from the Pacific Northwest Laboratory's Department of Energy sponsored research and development programs from January 1978 through July of 1982. The abstracts are grouped in subject categories, as shown in the table of contents. Entries in the subject index also facilitate access by subject, e.g., High-Level Radioactive Wastes. Three indexes, each preceded by a brief description, are provided: Personal Author, Subject, and Report Number.

Cited are research reports, journal articles, books, patents, theses, and conference papers. Excluded are technical progress reports. Since 1978 the Nuclear Waste Management Quarterly Progress Report has been published under the series number PNL-3000. Beginning in 1982, this publication has been issued semiannually, under the series number PNL-4250.

This bibliography is the successor to two others, BNWL-2201 (covering the years 1965-1976) and PNL-4050 (1975-1978). It is intended to provide a useful reference to literature in waste management written or compiled by PNL staff.

The efforts of Catherine Grissom at the Technical Information Center and her staff in compiling this document are gratefully acknowledged. This organization publishes a semimonthly abstract bulletin, Radioactive Waste Management, available on subscription as PB-902900.

J. A. Powell

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ABSTRACTS

NUCLEAR FUELS

Exploration

1 (BNWL--2100(Pt.4), pp 55-61) RADIATION INSTRUMENTATION: RADIOLOGICAL CHEMISTRY. Jun 1977.

Pacific Northwest Laboratory annual report for 1976 to the ERDA Assistant Administrator for Environment and Safety. Part IV. Physical and technological programs.

Program efforts were concentrated in the following areas: development of low-level radiochemical laboratory techniques, in situ monitoring techniques, and activation analysis technology. Four different techniques were evaluated for borehole analysis of uranium and thorium ores. These involved the detection of fission product photons after ^{252}Cf activation, the detection of low-energy gamma rays, the direct measurement of the 1001 keV photon from / sup 242m/Pa, a ^{238}U daughter, and isotopic excitation x-ray fluorescence spectroscopy. X-ray fluorescence spectroscopy and detection of uranium daughter photons allowed 0.01% uranium to be detected. X-ray spectrometry of rare earth elements following activation analysis also allowed better elemental detection sensitivities to be obtained than did the detection of high-energy photons identified with a Ge(Li) detector in typical instrumental neutron activation analysis. An activation analysis facility is under development utilizing ^{252}Cf in a ^{235}U-fueled subcritical assembly. This assembly will be used to develop cyclic activation analysis techniques for some 65 elements in environmental matrices with sensitivities varying from parts per million to parts per billion. A study was initiated to identify instrumentation required for the measurement of transuranium elements associated with power reactor fuels. Transuranium isotopes, ^{237}Pu, ^{238}Pu, ^{239}Pu, ^{241}Am, ^{243}Am, and ^{244}Cm can be identified through low energy photons or x-rays emitted following alpha emission with planar intrinsic Ge detectors.

Feed Processing

2 (PNL--2593) ASSESSMENT OF U.S. DOMESTIC CAPACITY FOR PRODUCING REACTOR-GRADE THORIUM DIOXIDE AND CONTROLLING ASSOCIATED WASTES AND EFFLUENTS. Enderlin, W.I. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Feb 1978. Contract EY-76-C-06-1830. 58p. Dep. NTIS, PC A04/MF A01.

Demand for reactor-grade ThO_2 is likely to increase as a result of the growing interest in the application of the thorium-uranium fuel cycle to nuclear reactors. The wastes and effluents identified with the production of ThO_2 from monazite sand are waste water, tailings, dust, smoke and gas, and radionuclides (primarily, ^{232}Th and ^{226}Ra). There are currently an estimated 1,500 short tons of crude thorium hydroxide byproduct that can be readily converted to reactor-grade ThO_2. The present maximum domestic capacity for producing reactor-grade ThO_2 is about 65 to 100 ton/year. The current domestic capacity for producing reactor-grade ThO_2 is sufficient to sustain a thorium-uranium fuel cycle of up to 11,000 MW(e) without recycling thorium, depending on the mix of reactor types selected. This range can be increased to 28,000 MW(e) by expanding ThO_2 purification capacity to match the current production rate of crude thorium byproduct. Potential constraints identified which may impact the expansion of domestic ThO_2 production are (1) uncertainty in the marketplace, (2) limited available thorium for production of reactor-grade ThO_2, (3) limited production capacity, and (4) mounting public concern over current levels of radioactivity detected at various points in the production process of thorium and uranium products.

Spent Fuels Reprocessing

3 (NUREG/CR--0131) DECOMMISSIONING OF NUCLEAR FACILITIES: AN ANNOTATED BIBLIOGRAPHY. Konzek, G.J.; Sample, C.R. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Sep 1978. Contract EY-76-C-06-1830. 532p. Dep. NTIS, PC A23/MF A01.

An information center containing literature relating to the decontamination, decommissioning, and disposal of nuclear equipment and facilities has been established at Battelle's Pacific Northwest Laboratory. This literature review has been compiled and abstracted to serve scientists, engineers, utility planners, and private citizens by directing them to sources of information on various aspects of decommissioning of nuclear fuel cycle facilities. The unclassified references cover the period from 1944 through mid-1978. Alphabetized author, subject, and report number indexes are provided and are cross-referenced to bibliography numbers 1 through 726.

4 (PNL--2985) CLADDING HULL DECONTAMINATION PROCESS: PRELIMINARY DEVELOPMENT STUDIES. Griggs, B.; Bryan, G.H. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Dec 1979. Contract EY-76-C-06-1830. 63p. Dep. NTIS, PC A04/MF A01.

An investigation of the chemical and radioactive properties of fuel hulls was conducted to assist in a decontamination process development effort. The removal of zirconium oxide layers from zirconium was accomplished by a treatment in 600°C HF followed by a dilute aqueous reagent. Similar treatment in fused alkali-zirconium fluoride salt baths was examined. A remotely operated small batch facility was developed and process parameters determined. 16 figures, 9 tables.

5 (PNL--3153) COMPARATIVE TECHNIQUES FOR NUCLEAR FUEL CYCLE WASTE MANAGEMENT SYSTEMS. Pelto, P.J.; Voss, J.W. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Sep 1979. Contract EY-76-C-06-1830. 177p. Dep. NTIS, PC A09/MF A01.

A safety assessment approach for the evaluation of predisposal waste management systems is described and applied to selected facilities in the light water reactor (LWR) once-through fuel cycle and a potential coprocessed UO_2-PuO_2 fuel cycle. This approach includes a scoping analysis on pretreatment waste streams and a more detailed analysis on proposed waste management processes. The primary evaluation parameters used in this study include radiation exposures to the public from radionuclide releases from normal operations and potential accidents, occupational radiation exposure from normal operations, and capital and operating costs. On an overall basis, the waste management aspects of the two fuel cycles examined are quite similar. On an individual facility basis, the fuel coprocessing plant has the largest waste management impact.

6 (PNL--3166(Pt.1)) CLADDING HULL DECONTAMINATION AND DENSIFICATION PROCESS. PART 1. THE PROTOTYPE CLADDING HULL DECONTAMINATION SYSTEM. Lembright, T.M.; Montgomery, D.R. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Apr 1980. Contract EY-76-C-06-1830. 56p. NTIS, PC A04/MF A01.

A prototype system for decontaminating Zircaloy-4 cladding hulls has been assembled and tested at Pacific Northwest Laboratory. The decontamination process consists of treatment with a gaseous mixture of hydrogen fluoride (HF) and argon (Ar) followed by a dilute aqueous etch of ammonium oxalate, ammonium citrate, ammonium fluoride, and hydrogen peroxide. The continuous cleaning process described in this report successfully descaled small portions of most charges, but was unable to handle the original design capacity of 4 kg/hr because of problems in the following areas: control of HF reactor temperatures, regulation of HF and argon mixtures and flows, isolation of the HF reactor atmosphere from the aqueous washer/rinser atmosphere, regulation of undesirable side reactions, and control over hull transport through the system. Due to the limited time available to solve these problems, the system did not attain fully operational status. The work was performed with unirradiated hulls that simulated irradiated hulls. The system was not built to be remotely operable. The process chemistry and system equipment are described in this report with particular emphasis on critical operating areas. Recommendations for improved system operation are included.

7 (PNL-SA--7752) KRYPTON-85 STORAGE IN SOLID MATRICES. Tingey, G.L.; McClanahan, E.D.; Bayne, M.A.; Gray, W.J.; Hinman, C.A. (Battelle Pacific Northwest Labs., Richland, WA (USA); Hanford Engineering Development Lab., Richland, WA (USA)). 1979. Contract AC06-76RL01830. 9p. (CONF-791112--67). NTIS, PC A02/MF A01.

From Materials Research Society annual meeting; Boston, MA, USA (26 Nov 1979).

Storage of Kr-85 will be required in support of nuclear power reactors beginning in 1983. Both approaches described here appear to meet the requirements for such a storage medium. Entrapment of the Kr during sputtering has several rather obvious safety advantages. The operation of the process at low rho and at or below room temperature should reduce markedly the potential for significant Kr-85 release to the environment during processing of the waste stream. It also appears that adaptation of this process for handling radioactive materials would also be simpler than the large high pressure, high temperature apparatus required for loading the glass sample. Furthermore, a significantly higher Kr loading is possible in the sputtered metals thus reducing the volume required for storage by as much as a factor of 50 to 100. On the other hand, the low density loaded glass process takes advantage of a very inexpensive starting material and existing commercial technology for high temperature, high pressure processes. The volume of the Kr-loaded glass matrix could be reduced by going to still higher pressures.

8 (PNL-SA--7844) KILOGRAM-SCALE PURIFICATION OF AMERICIUM BY ION EXCHANGE. Wheelwright, E.J. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 1979. Contract EY-76-C-06-1830. 20p. (CONF-791008--4). NTIS, PC A02/MF A01.

From Symposium on separation science and technology for energy applications; Gatlinburg, TN, USA (30 Oct 1979).

Sequential anion and cation exchange processes have been used for the final purification of ^{241}Am recovered during the reprocessing of aged plutonium metallurgical scrap. Plutonium was removed by absorption of Dowex 1, X-3.5 (30 to 50 mesh) anion exchange resin from 6.5 to 7.5 M HNO_3 feed solution. Following a water dilution to 0.75 to 1.0 M HNO_3, americium was absorbed on Dowex 50W, X-8 (50 to 100 mesh) cation exchange resion. Final purification was accomplished by elution of the absorbed band down 3 to 4 successive beds of the same resin, preloaded with Zn^{2+}, with an NH_4OH buffered chelating agent. The recovery of mixed ^{241}Am-^{243}Am from power reactor reprocessing waste has been demonstrated. Solvent extraction was used to recover a HNO_3 solution of mixed lanthanides and actinides from waste generated by the reprocessng of 13.5 tons of Shippingport Power Reactor blanket fuel. Sequential cation exchange band-displacement processes were then used to separate americium and curium from the lanthanides and then to separate approx. 60 g of ^{244}Cm from 1000 g of mixed ^{241}Am-^{243}Am.

9 (PNL-TR--409) INVESTIGATIONS OF THE SOLIDIFICATION OF FEED CLARIFICATION SLUDGE IN AN INORGANIC MATRIX RESISTANT TO RADIATION AND HEAT (VITRIFICATION), SUBPROJECT B. Stammler, M.; Eggersdorfer, R.; Gaar, H.; Harr, K.; Schwaemmlein, W.; Sherriff, R. (Battelle-Institut e.V., Frankfurt am Main (Germany, F.R.)). 13 Oct 1980. Contract AC06-76RL01830. Translation of BMFT-FB (KWA 1688 A), April 1980 105p. NTIS, PC A06/MF A01.

After concentration, dissolver sludge can be embedded into a glass matrix. The resulting product shows high radiation and recrystallization stability. The process to make the product is simple and causes practically no secondary waste. Mixtures of simulated FKS and frit can be processed into spherical pellets on a pelletizing disc. Another possibility is to pour the mixture into hemispherical cavities in a suitable substrate material. Sintering at about 700°C results in pellets with a vitrified surface. Some of the equipment necessary for the process has been tested.

10 (PNL-TR--352) AVAILABILITY OF PROCESSING SYSTEMS WITH PARALLEL AND SERIES STRUCTURE AND WITH INTERMEDIATE STORAGES. Schumacher, F. (Gesellschaft fuer Kernforschung m.b.H., Karlsruhe (Germany, F.R.)). Dec 1978. Contract EY-76-C-06-1830. Translation of KFK-2469 66p. Dep. NTIS, PC A04/MF A01.
The operational reliability of parallel-series connected procesing units and of processing units connected by intermediate storages is analyzed with analytical models and with simulation models. The influence of individual input variables and model assumptions on the system availability is examined with a sensitivity analysis. The results of the studies are presented in the form of graphs.

11 (BNWL-tr--321) STATUS OF FRENCH RESEARCH AND DEVELOPMENT IN THE AREA OF TREATMENT OF GASES PRODUCED BY REPROCESSING PLANTS. Chesne, A.; Miquel, P.; Goumondy, J.P.; Leseur, A. (CEA Centre d'Etudes Nucleaires de Fontenay-aux-Roses, 92 (France). Div. de Chimie). Jun 1978. Contract EY-76-C-06-1830. Translation of French report 18p. Dep. NTIS, PC A02/MF A01.
This report reviews current and planned future research of CEA on treatment of gaseous effluents of fuel reprocessing plants for both LWRs and fast reactors. Trapping of iodine (by precipitation of Cu_2I_2 and PbI_2) and of rare gases is covered. 5 figures. (DLC)

12 (BNWL-tr--322) TRITIUM CONTROL IN REPROCESSING PLANTS. Goumondy, J.P.; Miquel, P. (CEA Centre d'Etudes Nucleaires de Fontenay-aux-Roses, 92 (France). Div. de Chimie). Jun 1978. Contract EY-76-C-06-1830. Translation of French report 32p. Dep. NTIS, MF A01.
Portions of document are illegible.
The tritium balance at the reprocessing stage of the light-water fuel cycle should be considered from the two aspects of operational exploitation and environmental pollution. The Trilex procedure may well prove to be a viable means of controlling the element and preventing its dissemination in the water from future reprocessing plants. It consists of the following stages: (1) establishing a tritium barrier by washing the solvent charged with the first uranium and plutonium extract with natural water and (2) managing the recycling steps and the addition of water to the reprocessing operation so as to regulate the dilution of the tritiated water depending on whether the tritiated water is to be stored or discharged. Thus, the tritiated water can be confined to a volume of from 500 to several thousand liters per ton of treated fuel. Laboratory tests and pilot studies have shown that the tritium barrier is effective and that the Trilex procedure can be applied in conventional reprocessing plants. 7 figures.

13 (BNWL-tr--323) MINIMIZATION OF THE PRODUCTION OF ALPHA-BEARING WASTES IN FUEL REPROCESSING PLANTS. Bethellier, A. (CEA Centre d'Etudes Nucleaires de Fontenay-aux-Roses, 92 (France)). Jun 1978. Contract EY-76-C-06-1830. Translation of French report 19p. Dep. NTIS, PC A02/MF A01.
Various methods for concentrating and separating alpha contaminants are reviewed: evaporation, combustion, washing and lixiviation. Their effects on the operation of a fuel reprocessing plant are considered. Guidelines for design of the plant are given. Finally, economic aspects are considered briefly. (DLC)

14 (BNWL-tr--312) IODINE SEPARATION IN REPROCESSING PLANTS. Wilhelm, J.G.; Furrer, J. 24 Apr 1978. Contract EY-76-C-06-1830. Translation: source information not available 25p. Dep. NTIS, PC A02/MF A01.
This review deals briefly with the quantities of radioiodine occurring in reprocessing plants and the requirements on separation of iodine from the offgas. The various processes for separation of radioiodine are described, together with their technical development level, and their advantages and disadvantages are discussed. Details are given on separation of iodine from the gas phase by wet scrubbing and collection on solid sorbents. A method for precipitating iodine from the fual solution is also mentioned. Problems yet to be solved by research and development are discussed. Those include separation of iodine from the vessel offgas, and the stability of the separated products in regard to their storage.

Transport and Storage

15 (BNWL-SA--6378) CURRENT PERCEPTIONS OF SPENT NUCLEAR FUEL BEHAVIOR IN WATER POOL STORAGE. Johnson, A.B. Jr. (Lawrence Livermore National Lab., CA (USA)). Jun 1977. Contract EY-76-C-06-1830. 5p. (CONF-771109--108). Dep. NTIS, PC A02/MF A01.
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
A survey was conducted of a cross section of U.S. and Canadian fuel storage pool operators to define the spent fuel behavior and to establish the range of pool storage environments. There is no evidence for significant corrosion degradation. Fuel handling causes only minimal damage. Most fuel bundles with defects generally are stored without special procedures. Successful fuel storage up to 18 years with benign water chemistry has been demonstrated. 2 tables. (DLC)

16 (CONF-780506--(Vol.1), pp 193-198) PLACING THE SPECIAL TRAINS ISSUE IN PERSPECTIVE. Rhoads, R.E. (Battelle Pacific Northwest Labs., Richland, WA); DeSteese, J.G.; Loscutoff, W.V.; Chais, M. 1978. NTIS, PC A23/MF A01.
From 5. symposium on packaging and transportation of radioactive materials; Las Vegas, NV, USA (7 May 1978).
The Association of American Railroads has proposed changes in the way railroads handle shipments of radioactive materials. These changes are embodied in a set of recommended operating practices that would require shipments of spent fuel and radioactive waste to be moved only in special train service. The proposed operating practices include a 35 mph maximum speed restriction, a passing restriction and a no-other freight restriction. Shippers of radioactive materials oppose the imposition of these operating practices. The special train issue is currently being argued in hearings before the Interstate Commerce Commission. The history and current status of

these hearings are reviewed. Pacific Northwest Laboratories has undertaken a study to provide perspective on the safety and economic factors related to the use of special trains for shipping spent fuel. The results of this study for the amount of spent fuel anticipated to be shipped in 1986 are reviewed. The safety analysis determines the frequencies and severities of accidents for conventional freight trains and compares these to extrapolated frequencies and severities of accidents for conventional freight trains subject to the special train operating restrictions. Results of the study show that the adoption of special trains and attendant operating restrictions has limited potential for improving safety during shipment. The economic analysis compares the cost of spent fuel shipments made by special train and by conventional freight train service.Results of the economics phase of the study show that the use of special trains will most likely increase the cost of shipments by about 50%, although under certain circumstances shipping costs for spent fuel by special trains may be up to 20% lower than by conventional train service. Possible methods to resolve the special train issue are explored.

17 (CONF-780506--(Vol.1), pp 52-58)
CONCEPTUAL DESIGN OF A SHIPPING CASK FOR RAIL TRANSPORT OF SOLIDIFIED HIGH LEVEL WASTE.
Rhoads, R.E.; Peterson, P.L. (Battelle Pacific Northwest Labs., Richland, WA). 1978. NTIS, PC A23/MF A01.

From 5. symposium on packaging and transportation of radioactive materials; Las Vegas, NV, USA (7 May 1978).

No commercial facilities to reprocess spent fuel from light water reactors are currently in operation in the US. Whether or not such facilities will be operated during the remainder of this century is currently a topic of national debate. If it is decided to proceed with the reprocessing option, much work remains to be done to develop commercial facilities to solidify the liquid high level waste generated from the process, transport it to a federal repository and isolate it from the environment. The Department of Energy is currently conducting a variety of research activities in these areas so that reprocessing facilities could be developed and operated in a timely fashion if the decision to use this technology is made. This paper reports the results of a study to develop a conceptual design for a shipping cask to transport solidified high level waste from a reprocessing plant to a federal repository by rail. Scale models of the cask design and accompanying graphics displays have also been developed. The design of a shipping container for radioactive materials is dependent on a variety of design constraints. The design constraints important to development of a cask for shipping solidified high level waste are reviewed. Since commercial facilities to solidify high level waste have not been operated, assumptions based on current technology were made concerning the properties of the solidified high level waste that would be shipped.The conceptual cask would transport nine canisters of waste. The canisters are assumed to be 0.3 m in diameter and 3 m long. Possible cask configurations for other canister sizes are described.

18 (CONF-780506--(Vol.2), pp 939-945)
TRANSPORTATION OF POSTFISSION RADIOACTIVE WASTES FROM THE COMMERCIAL LWR FUEL CYCLE.
Murphy, E.S.; Clark, L.L.; Napier, B.A. (Battelle Pacific Northwest Labs., Richland, WA). 1978. NTIS, PC A22/MF A01.

From 5. symposium on packaging and transportation of radioactive materials; Las Vegas, NV, USA (7 May 1978).

Requirements for transportation of postfission radioactive wastes from the commercial LWR fuel cycle were studied. Only radioactive wastes that would require transport to federal facilities for interim storage or final disposal were considered. These wastes included spent fuel, solidified high-level waste, fuel residues and non-high-level TRU waste. Transportation requirements were determined and transport facilities described; packaging and shipping costs were estimated; and the potential environmental impacts of transportation and of postulated accidents were evaluated. Projections are given for the number one cost of waste shipments for the once-through fuel cycle and for a uranium plus plutonium recycle option. The projections are based on an assumed nuclear generating capacity of 400 GWe in the year 2000. On a cost per kg of heavy metal basis, truck cask shipments appear to cost about the same as rail cask shipments. However, shipping cost estimates do not take into account the large increase in cask handling capacity required if all shipments were by truck. It is therefore assumed that cask shipments will be by rail. Under normal transport conditions, no radioactive materials will be released from shipping containers to the atmosphere, ground or water. Direct radiation doses calculated for members of the general population from passing rail or truck shipments of radioactive wastes are insignificant when compared to doses from natural background radiation. In the absence of accident data involving material releases, a series of minor, moderate and severe accident scenarios for radioactive shipments were postulated. Expected frequencies for these accidents were calculated on the basis of accident statistics and assumed total shipment miles; material releases were postulated for each accident scenario; and radiation doses from these postulated releases were calculated.

19 (DOE/ET--0028(Vol.4)) TECHNOLOGY FOR COMMERCIAL RADIOACTIVE WASTE MANAGEMENT.
(Battelle Pacific Northwest Labs., Richland, WA (USA)). May 1979. Contract EY-76-C-06-1830. 547p. Dep. NTIS, PC A23/MF A01.

A general analysis of transportation requirements for postfission radioactive wastes that are produced from the commercial light water reactor (LWR) fuel cycle and that are assumed to require Federal custody for storage or disposal is given. Possible radioactive wastes for which transportation requirements are described include: spent fuel, solidified high-level waste, fuel residues (cladding wastes), plutonium, and non-high-level transuranic (TRU) wastes. Transportation is described for wastes generated in three fuel cycle options: once-through fuel cycle, uranium recycle only, and recycle of uranium and plutonium. The geologic considerations essential for repository selection, the nature of geologic formations that are potential repository media, the thermal criteria for waste placement in geologic repositories, and conceptual repositories in four different geologic media are described. The media are salt deposits, granite, shale, and basalt. Possible alternatives for managing retired facilities and procedures for decommissioning are reviewed. A qualitative comparison is made of wastes generated by the uranium fuel cycle and the thorium fuel cycle. This study presents data characterizing wastes from prebreeder light water breeder reactors using thorium and slightly enriched uranium-235. The prebreeder LWBRs are essentially LWRs using thorium. The operation of HTGR and LWBR cycles are conceptually designed, and wastes produced in these cycles are compared for potential differences.

20 (HEDL-SA--2484-FP) HIGH TEMPERATURE POST-IRRADIATION PERFORMANCE OF SPENT PRESSURIZED-WATER-REACTOR FUEL RODS UNDER DRY-STORAGE CONDITIONS. Einziger, R.E.; Atkin, S.D.; Stellrecht, D.E.; Pasupathi, V. (Hanford Engineering Development Lab., Richland, WA (USA); Battelle Columbus Labs., OH (USA)). Jun 1981. Contract AC06-76FF02170. 48p. NTIS, PC A03/MF A01. Order Number DE82003780.

Post-irradiation studies on failure mechanisms of well characterized PWR rods were conducted for up to a year at 482, 510 and 571°C in unlimited air and inert gas atmospheres. No cladding breaches occurred even though the tests operated many orders of magnitude longer in time than the lifetime predicted by Blackburn's analyses. The extended lifetime is due to significant creep strain of the Zircaloy cladding which decreases the internal rod pressures. The cladding creep also contributes to radial cracks, through the external oxide and internal FCCI layers, which propagated into and arrested in an oxygen stabilized α-Zircaloy layer. There were no signs of either additional cladding hydriding, stress-corrosion cracking or fuel pellet degradation. Using the Larson-Miller formulization, a conservative maximum storage temperature of 400°C is recommended to ensure a 1000-year cladding lifetime. This accounts for crack propagation and assumes annealing of the irradiation-hardened cladding.

21 (NUREG/CR--0956) COMMENTARY ON SPENT FUEL STORAGE AT MORRIS OPERATION. Eger, K.J.; Zima, G.E. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Oct 1979. Contract EY-76-C-06-1830. 122p. (PNL--3065). Dep. NTIS, PC A06/MF A01.

The General Electric Company is providing technical support to Battelle Pacific Northwest Laboratories in the analysis of the design, operation, and maintenance experience in the handling of nuclear fuel at the Independent Spent Fuel Storage Facility. The purpose of this report is to provide a description of spent fuel handling activities and systems, and an analysis of the storage performance as developed over the seven year operational history of the Morris Operation. Design considerations and performance are analyzed for both the basin and key supporting systems. The bases for this analysis are the provisions for containing radioactive by-product materials, for shielding from the radiation they emit, and for preventing the formation of a critical array. These provisions have been met effectively over the history of storage at Morris. The release of radioactive materials is minimized by the protection of the cladding integrity, the containment of the basin water, the removal of radioactive and other contaminants from the water, and by filtering and then dispersing the basin air. Four auxiliary systems are provided to accomplish this, the basin leak detection system, the filter, the coolers, and the building ventilation system. This successful history notwithstanding, action to reduce personnel exposure, to improve fuel handling reliability and to lessen the potential for accidents continues to be taken.

22 (PNL--2244) CONCEPTUAL DESIGN OF A SHIPPING CONTAINER FOR TRANSPORTING HIGH-LEVEL WASTE BY RAILROAD. Peterson, P.L.; Rhoads, R.E. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Dec 1978. Contract EY-76-C-06-1830. 70p. Dep. NTIS, PC A04/MF A01.

A shipping cask uniquely designed to transport solidified high-level wastes (SHLW) from a reprocessing plant to a federal repository has not yet been developed. The amount of material that would be transported and the anticipated characteristics of the SHLW suggest that rail casks will be favored for this transportation because of cost and logistic considerations. The document presents the results of a study to develop a conceptual design for a rail cask for transporting SHLW and to construct scale models of the conceptual cask with accompanying graphics for use at technical meetings and in public information displays. Two 1/10 scale models of the conceptual cask and two HO gauge (1/87 scale) models of the cask/railcar system have been constructed. A description of the models and accompanying graphics is presented in Appendix A.

23 (PNL--2457) LEGAL, INSTITUTIONAL, AND POLITICAL ISSUES IN TRANSPORTATION OF NUCLEAR MATERIALS AT THE BACK END OF THE LWR NUCLEAR FUEL CYCLE. Lippek, H.E.; Schuller, C.R. (Battelle Human Affairs Research Center, Seattle, WA (USA)). Mar 1979. Contract EY-76-C-06-1830. 188p. Dep. NTIS, PC A09/MF A01.

A study was conducted to identify major legal and institutional problems and issues in the transportation of spent fuel and associated processing wastes at the back end of the LWR nuclear fuel cycle. (Most of the discussion centers on the transportation of spent fuel, since this activity will involve virtually all of the legal and institutional problems likely to be encountered in moving waste materials, as well.) Actions or approaches that might be pursued to resolve the problems identified in the analysis are suggested. Two scenarios for the industrial-scale transportation of spent fuel and radioactive wastes, taken together, high-light most of the major problems and issues of a legal and institutional nature that are likely to arise: (1) utilizing the Allied General Nuclear Services (AGNS) facility at Barnwell, SC, as a temporary storage facility for spent fuel; and (2) utilizing AGNS for full-scale commercial reprocessing of spent LWR fuel.

24 (PNL--2500(Pt.5), pp 1.1-1.31) ENVIRONMENTAL CONTROL TECHNOLOGY. Feb 1978.

Pacific Northwest Laboratory annual report for 1977 to the DOE Assistant Secretary for Environment. Part 5. Control technology, overview, health, safety and policy analysis.

Status reports are given of the following projects: environmental control technology for shale oil wastewaters; assessment of environmental control technologies for commercial coal gasification systems; transportation safety studies; integrated LNG safety and control program; energy material transport, now through 2000; nuclear fuel cycle analysis; decommissioning of retired facilities at Hanford; characterization of 300 area burial ground; geothermal liquid waste disposal; management program plan for environmental concerns in compressed air energy storage; and assessment of energy-conserving industrial waste treatment technology. (LK)

25 (PNL--2637) SENSITIVITY OF THE FEDERAL FEE FOR MANAGING SPENT FUEL TO FINANCIAL AND LOGISTICAL VARIATIONS. White, M.K.; Lewallen,

M.A.; Merrill, E.T.; Fleischman, R.M. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jun 1978. Contract EY-76-C-06-1830. 49p. Dep. NTIS, PC A03/MF A01.

Three types of fees for federal spent fuel management service were calculated for a reference case and a number of variations. These fee types are a uniform fee applicable to all customers, a fee for disposal of spent fuel, and a fee for interim storage plus disposal of spent fuel. Results ranged from $124/kg to $256/kg for the uniform fee, $112/kg to $213/kg for the disposal fee, and $144/kg to $319/kg for the storage plus disposal fee. The reference case assumed that spent fuel would first be received by the government in 1983 at a 5,000 MT away-from-reactor (AFR) basin. The first repository (45,000 MT) was assumed ready for fuel in 1988, and the second (100,000 MT) in 1997. The reference case results in fees of $129/kg for the uniform fee, $117/kg for disposal, and $232/kg for storage plus disposal. The sensitivity cases were grouped in five general categories of variations from the reference case assumptions: demand for storage/disposal services, facility schedules and characteristics, methodology for calculating the fee, discount rate and AFR financing, and delays or failure of the first repository.

26 (PNL--3566) SPENT-FUEL SPECIAL-STUDIES PROGRESS REPORT: PROBABLE MECHANISMS FOR OXIDATION AND DISSOLUTION OF SINGLE-CRYSTAL UO_2 SURFACES. Wang, R. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Mar 1981. Contract AC06-76RL01830. 102p. NTIS, PC A06/MF A01.

Due to the complexity of the structural, microstructural and compositional characteristics of spent fuel, basic leaching and dissolution mechanisms were studied with UO_2 matrix material, specifically with single-crystal UO_2, to isolate individual contributory factors. The effects of oxidation and oxidation-dissolution were investigated in different oxidation conditions, such as in air, oxygenated solutions and deionized water containing H_2O_2. In addition, the effects of temperature on dissolution of UO_2 were studied in autoclaves at 75 and 150°C. Also, oxidation and dissolution measurements were investigated via electrochemical methods to determine if those techniques could be applied to the characterization of leaching and dissolution of spent fuel in a hot cell. Finally, the effects of radiation were explored since the radiolysis of water may create a localized oxidizing condition at or near the spent fuel-solution interface, even in neutral or reducing conditions as commonly found in deep geological environments. The oxidation and oxidation-dissolution mechanisms for UO_2 are proposed as follows: The UO_2 surface is first oxidized in solution to form a UO/sub 2+x/ surface layer several angstroms thick. This oxidized surface has a high dissolution rate since the UO/sub 2+x/ reacts with the dissolved O_2, or H_2O_2, to form uranyl complex ions in a U(VI) state. As the uranyl ions exceed the solubility limits in solution, they become hydrolyzed to form solid deposits and suspended particles of UO_3 hydrates. The thickness and porosity of the deposited UO_3 hydrate surface-film is dependent on temperature, pH and deposition time. A long-term dissolution rate is then determined by the nature of the surface film, such as porosity, solubility and mechanical properties.

27 (PNL--3841) TRANSPORTATION CONSIDERATIONS RELATED TO WASTE FORMS AND CANISTERS FOR DEFENSE TRU WASTES. Schneider, K.J.; Andrews, W.B.; Schreiber, A.M.; Rosenthal, L.J.; Odle, C.J. (Pacific Northwest Lab., Richland, WA (USA)). Sep 1981. Contract AC06-76RL01830. 122p. (TTC--0241). NTIS, PC A06/MF A01. Order Number DE82001159.

This report identifies and discusses the considerations imposed by transportation on waste forms and canisters for contact-handled, solid transuranic wastes from the US Department of Energy (DOE) activities. The report reviews (1) the existing raw waste forms and potential immobilized waste forms, (2) the existing and potential future DOE waste canisters and shipping containers, (3) regulations and regulatory trends for transporting commercial transuranic wastes on the ISA, (4) truck and rail carrier requirements and preferences for transporting the wastes, and (5) current and proposed Type B external packagings for transporting wastes.

28 (PNL-SA--6416) LEACHING OF IRRADIATED LWR FUEL PELLETS IN DEIONIZED WATER, SEA BRINE, AND TYPICAL GROUND WATER. Katayama, Y.B.; Mendel, J.E. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Dec 1977. Contract AC06-76RL01830. 17p. (CONF-771109--114). NTIS, PC A02/MF A01.

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

The goal of the fuel leaching program is to generate durability data for irradiated fuels and to compare this data with other waste forms. The two major experimental parameters that were considered in the LWR fuel leaching studies are as follows: LWR fuel leaching studies; worst case fuel; and ground water plus reference leachant. The first experimental parameter was determined by locating a source of high burnup LWR fuel at the Battelle-Columbus Hot Laboratories which was one year out-of-reactor. The second experimental parameter was established by selecting Hanford well water as the ground water and deionized water as the reference leachant for comparing leach rates with other waste forms. A third leachant, was a synthetic sea salt brine solution. The leachant sampling frequency was patterned like the standard IAEA leaching procedure. The radiochemical analysis performed on the leach solutions were as follows: Cs quantified by gamma spectroscopy; Sb, Eu, Ru, Ce by gamma spectroscopy after removal of Cs; Sr + Y by beta analysis; Am analysis done after separation from ^{238}Pu. Leach rates were determined by the concentration of specific radioisotopes in the leach solution and calculated by the equation, leach rate = (α/sub n//A_0)(W/S)(1/n) = grams of solid/cm^2-day. The following leach rates were observed following 616 days of leaching in deionized water: ^{137}Cs 8×10^{-8}; ^{239}Pu and ^{240}Pu 2×10^{-9}; ^{244}Cm 6×10^{-7}. The release mechanism for these radioisotopes are all different and there is no single mechanism to describe the release of radioisotopes after 616 days of leaching.

29 (PNL-SA--6526) CONCEPTUAL DESIGN OF A SHIPPING CASK FOR RAIL TRANSPORT OF SOLIDIFIED HIGH LEVEL WASTE. Rhoads, R.E.; Peterson, P.L. (Battelle Pacific Northwest Labs., Richland, WA (USA)). May 1978. Contract EY-76-C-06-1830. 7p. Dep. NTIS, PC A02/MF A01.

A shipping cask uniquely designed to transport solidified high level waste (SHLW) from reprocessing spent LWR fuel has not yet been developed. Battelle, Pacific Northwest Laboratory (PNL) has conducted a program for the Department of Energy to develop a conceptual design for a rail cask for transporting SHLW, and to produce models and graphical displays based on the conceptual design for use at technical meetings and in public information displays. The results of this work are summarized in this paper.

30 (PNL-SA--7344) NEAR SURFACE SPENT FUEL STORAGE: ENVIRONMENTAL ISSUES. Nelson, I.C.; Shipler, D.B.; McKee, R.W.; Glenn, R.D. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 1979. Contract EY-76-C-06-1830. 20p. (CONF-791004--2). Dep. NTIS, PC A02/MF A01.

From ASCE national meeting; Atlanta, GA, USA (Oct 1979).

Interim storage of spent fuel appears inevitable because of the lack of reprocessing plants and spent fuel repositories. This paper examines the environmental issues potentially associated with management of spent fuel before disposal or reprocessing in a reference scenario. The radiological impacts of spent fuel storage are limited to low-level releases of noble gases and iodine. Water needed for water basin storage of spent fuel and transportation accidents are considered; the need to minimize the distance travelled is pointed out. Resource commitments for construction of the storage facilities are analyzed.

31 (PNL-SA--7734) LONG-TERM LEACHING OF IRRADIATED SPENT FUEL. Katayama, Y.B.; Bradley, D.J. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 1979. Contract AC06-76RL01830. 29p. (CONF-791112--68). NTIS, PC A03/MF A01.

From Materials Research Society annual meeting; Boston, MA, USA (26 Nov 1979).

Spent Light Water Reactor (LWR) fuel with burnups of 9, 28 and 54 MWd/kg U were leach tested at 25°C in deionized water in a Paige apparatus. No discernible differences in leach rates were observed due to burnup. Additionally, the 28 MWd/kg U fuel was IAEA leach tested in five different leachants using the IAEA method. Deionized water gave the highest leach rates and a calcium chloride solution gave the lowest leach rates. An accelerated leaching period was observed during the Paige leach test of the 54 MWd/kg U spent fuel. Comparison between spent fuel and borosilicate waste glass leach rates was made. In sodium bicarbonate solution the leach rates are near equal and the glass becomes increasingly more durable with $CaCl_2$ solution, followed by sodium chloride solution, WIPP B brine and deionized water where the glass is two to three orders of magnitude more leach resistant then spent fuel. 16 figures.

32 (PNL-SA--9247) TWO FACTORS IMPORTANT TO THE CRITICALITY POTENTIAL OF SPENT FUEL IN GEOLOGIC REPOSITORIES. Gore, B.F.; Jenquin, U.P. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Feb 1981. Contract AC06-76RL01830. 29p. (CONF-810217--10). NTIS, PC A02/MF A01.

From ANS waste management conference; Tucson, AZ, USA (23 Feb 1981).

Two factors important to the criticality potential of spent fuel in geologic repositories are: the residual fissile content of the fuel, and the extent to which geochemical processes might somehow separate and accumulate plutonium from other spent fuel materials. This paper presents the results of two calculational surveys defining conditions required for criticality. In the first, homogeneous spherical mixtures of spent fuel actinide oxides and water with water reflection are analyzed. Graphs of minimum critical mass vs duration of in-reactor exposure are presented. Parametric variations from a base case are explored, including the effects of initial enrichment, post exposure radioactive decay and addition of rock materials to the mixture. In the second study, homogeneous spherical mixtures devoid of water, containing plutonium and a neutronically optimized rock material, with a thick rock neutron reflector are analyzed. Graphs of Pu critical mass are presented as a function of concentration over the range from 2 to 100 g Pu/l. Parametric variations from a base case are explored, including effects of rock composition, ^{240}Pu content and uranium contamination of the plutonium.

33 (PNL-SA--9826) REFERENCE ANALYSIS ON THE UTILITY OF ENGINEERED BARRIERS FOR GEOLOGIC DISPOSAL OF SPENT NUCLEAR FUEL: OVERVIEW. Cloninger, M.O. (Pacific Northwest Lab., Richland, WA (USA)). 1 Sep 1981. Contract AC06-76RL01830. 16p. (CONF-810998--1). NTIS, PC A02/MF A01. Order Number DE82003843.

From National Academy of Sciences - Waste Isolation Systems Panel review meeting; Washington, DC, USA (2 Sep 1981).

The development and characterization of waste forms, containers and other engineered barriers destined for use in the isolation of nuclear waste in deep geologic repositories has progressed to the point where there are several options for barrier systems that are available to help assure safe disposal of nuclear wastes. However, a rigorous basis has not yet developed to define whether various concepts or products are required or desirable, or how effective they should be for how long. This analysis is an attempt to contribute to that basis. Intent of the study is to determine what incentives exist for providing highly effective engineered barriers for the isolation of radioactive waste (spent fuel in this case) in a deep geologic repository. 6 figures.

34 (PNL-SA--10022) PACKAGE RECEIPT ASSAY SYSTEMS. Brodzinski, R.L. (Pacific Northwest Lab., Richland, WA (USA)). Nov 1981. Contract AC06-76RL01830. 7p. (CONF-811130--15). NTIS, PC A02/MF A01. Order Number DE82005781.

From 3. annual DOE participants information meeting on low-level waste management; New Orleans, LA, USA (4 Nov 1981).

Relevant waste package parameters are identified. Waste package measurement requirements and capabilities are evaluated and prioritized. A passive TRU package assayer is described.

35 FEASIBILITY OF CLOSE-PACKING FUEL RODS FOR GEOLOGIC DISPOSAL OF SPENT FUEL. McNair, G.W.; Gore, B.F. (Battelle Pacific Northwest Labs., Richland, WA). Transactions of the American Nuclear Society ; 34: 406-407(1980). (CONF-800607--).

From American Nuclear Society annual meeting; Las Vegas, NV, USA (8 Jun 1980).

SPENT FUEL STORAGE;UNDERGROUND STORAGE; GEOLOGIC DEPOSITS;RISK ASSESSMENT;FEASIBILITY STUDIES;SPENT FUEL ELEMENTS;ENVIRONMENTAL EFFECTS;CASKS;VOLUME;SIZE;CRITICALITY; CONFIGURATION;SPATIAL DEPENDENCE;RADIOACTIVE WASTE DISPOSAL

36 OPPORTUNITIES TO INCREASE THE PRODUCTIVITY OF SPENT-FUEL SHIPPING CASKS. Franklin, A.L.; DeSteese, J.G. . (Battelle Pacific Northwest Labs., Richland, WA). Transactions of the American Nuclear Society ; 34: 428-429(1980). (CONF-800607--).

From American Nuclear Society annual meeting; Las Vegas, NV, USA (8 Jun 1980).
SPENT FUEL CASKS;PRODUCTIVITY;RAIL TRANSPORT; RAILROAD CARS;TRUCKS;LAND TRANSPORT;DESIGN; MATERIALS HANDLING;RADIOACTIVE WASTE FACILITIES; SPENT FUELS;COST

37 EVALUATION OF ALTERNATIVE SPENT-FUEL MANAGEMENT FEES. White, M.K. (Battelle Northwest Labs., Richland, WA); Engel, R.L.; Fiore, J.J. Transactions of the American Nuclear Society ; 32: 378-379(Jun 1979). (CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).
SPENT FUELS;CHARGES;EVALUATION;GOVERNMENT POLICIES;RADIOACTIVE WASTE MANAGEMENT;SPENT FUEL STORAGE;RADIOACTIVE WASTE DISPOSAL

38 SPENT FUEL STORAGE EXPERIENCE. Johnson, A.B. Jr. (Battelle-Pacific Northwest Labs., Richland, WA). Nuclear Technology; 43: No. 2, 165-173(Apr 1979).
Irradiated nuclear fuel has been stored in water pools at essentially all nuclear reactors, beginning with the earliest plants in 1943. Fuel from water-cooled power reactors is clad either with Zircaloy or with stainless steel. Zircaloy-clad fuel has been stored US pools since 1959. Some experimental stainless-steel-clad fuel was stored for 12 yr in the US before reprocessing. Canadian Zircaloy-clad fuel has been stored since 1962. There has been no evidence that the fuel has degraded during pool storage, based principally on visual observations and radiation monitoring of pool air and water. However, several fuel rods have been subjected to metallographic examination after pool exposures up to 11 yr, also with no evidence that the fuel cladding has degraded in the pool. Canadian fuel stored up to 10 yr was returned to a reactor and performed well. Favorable storage experience also has been indicated for other countries with fuel residence times of 5 to 10 yr. The pool storage environment is high-purity water at 5.3 to 7.5 pH, except for pools for pressurized water reactors, which utilize boric acid pool chemistry at 4.5 to 6.0 pH. Pool water temperatures generally range between 20 and 50°C. The favorable storage experience, demonstrated technology, successful handling of fuel with reactor-induced defects, benign storage environments, and corrosion-resistant materials offer sufficient bases to proceed with expanded storage capacities and extended fuel storage until questions regarding fuel reprocessing and final storage of nuclear wastes have been resolved. Some surveillance is justified to detect degradation if it becomes significant. Surveillance programs are already under way in several countries. 1 figure, 6 tables.

39 (PNL-TR--399) PERMANENT STORAGE FACILITY FOR SPENT FUEL ASSEMBLIES. Klein, D.; Stuger, R. Translated from ATW, Atomwirtschaft, Atomtechnik ; 401-403(Sep 1978). Contract FY-76-C-06-1830. 11p. NTIS, PC A02/MF A01.
The capital costs for a permanent storage facility for spent LWR fuel assemblies with a storage capacity of 540 tons of uranium and with its own infrastructure lie on the order of ca. 260 DM/kg U. In the operating costs, a distinction must be made between costs in the charging stage and costs during the subsequent storage period. The specific operating costs amount to ca. 9 DM/kg U during the charging stage and decrease to 3 to 4 DM/kg U during exclusive storage operation. In conclusion, this prototype shows: fuel assemblies can be admitted to an air-cooled storage facility as early as a few years after their removal from the reactor, without being subjected to initial temperature stresses in excess of those during the preceding reactor operation. The system in which the fuel assemblies are cooled by natural convection of outside air offers inherent safety and low operating expenses. Radiological pollution of the environment can be reduced to extremely low levels by suitable precautions.

40 TRANSPORTATION LOGISTICS FOR SPENT-FUEL STORAGE AND DISPOSAL. Andrews, W.B.; Burnett, R.A.; Engel, R.L. (Battelle Northwest Labs., Richland, WA). Transactions of the American Nuclear Society ; 30: 292-293(1978). (CONF-781109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
SPENT FUELS;TRANSPORT;SPENT FUEL STORAGE; RADIOACTIVE WASTE DISPOSAL;MATHEMATICAL MODELS

Marketing and Economics

41 (PNL--3317-2) COMMERCIAL US NUCLEAR REACTORS AND WASTE: THE CURRENT STATUS. Platt, A.M.; Robinson, J.V. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Sep 1980. Contract AC06-76RL01830. 53p. NTIS, PC A04/MF A01.
Between March 1 and June 15, 1980, the declared size of the commercial light waste reactor (LWR) nuclear power industry in the US has decreased another 9 GWe. For the presently declared size: the 165 declared reactors will peak at a capacity of 153 GWe in 2001 and will consume about 870,000 MTU as enrichment feed; the theoretical rate of enrichment requirements will peak at about 19,000,000 SWUs/y in the year 2014; as few as two repositories each with capacity equivalent to 100,000 MTU would hold the waste; and predisposal storage reactor basins and AFRs (away-from-reactor basins) would peak at <85,000 MTU in the year 2020 if the two respositories were commissioned in the years 1997 and 2020. It should be noted that the number of declared LWRs has dropped from 226 on December 31, 1974 to 165 as of this writing. The oil equivalent of the energy loss, assuming a 50% efficiency in use as in cars, is 17,000 million barrels. This is about 10 years of the current rate of US consumption of OPEC oil.

42 (PNL-SA--9520) USE OF ASPHALT EMULSION SEALANTS IN DISPOSAL OF URANIUM MILL TAILINGS. Hartley, J.N.; Freeman, H.D.; Elmore, M.R. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jul 1981. Contract AC06-76RL01830. 10p. (CONF-810730--2). NTIS, PC A02/MF A01. Order Number DE81028019.
From 1981 national conference on environmental engineering; Atlanta, GA, USA (8 Jul 1981).
Studies of asphalt emulsion sealants conducted by the Pacific Northwest Laboratory have demonstrated that the sealants are effective in containing radon within uranium tailings. The laboratory and field studies have further demonstrated that radon exhalation from uranium tailings piles can be reduced by greater than 99% to near background levels. Field tests at the tailings pile in Grand Junction, Colorado confirmed that an 8-cm admix seal containing 22 wt % asphalt could be effectively applied with a cold-mix paver. Other techniques were successfully tested, including a soil stabilizer and a hot, rubberized asphalt seal that was applied with a distributor truck. After the seals were applied

and compacted, overburden was applied over the seal to protect the seal from ultraviolet degradation.

Waste Management

43 (BNWL--2117) PROCEEDINGS OF AN ACTINIDE-SEDIMENT REACTIONS WORKING MEETING. Ames, L.L. (ed.). (Battelle Pacific Northwest Labs., Richland, WA (USA)). 1976. Contract EY-76-C-06-1830. 575p. (CONF-760214--). Dep. NTIS, PC A99/MF A01.

From Conference on actinide sediment reaction; Seattle, WA, USA (10 Feb 1976).

Separate abstracts were prepared for seven sections of this report. Abstracts of two papers appeared previously in ERA.

44 (BNWL--2144) ACORN: A COMPUTER PROGRAM FOR PLOTTING FAULT TREES. Carter, J.L. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Nov 1977. Contract EY-76-C-06-1830. 48p. Dep. NTIS, MF A01.

Portions of document are illegible.

A description and user instructions are presented for ACORN, a FORTRAN computer program for drawing fault trees. ACORN analyzes the input logical structure of a fault tree and provides data for CalComp plot of the tree. AND, OR, and INHIBIT gates are permitted, and basic events are drawn as diamonds, circles, or houses. Each component (gate or basic event) can have a descriptive label within a rectangle attached to the top of its respective symbol. Tree logic is input as a set of FORTRAN statements, each defining a gate in terms of logical operations of the components input to it. ACORN develops the logical structure of the tree from the input statements. The tree's physical structure is developed by assigning relative spatial coordinates to the logical relationships between a gate and its inputs. ACORN provides input data checking, a printer plot of the fault tree, and plotting data for a CalComp model 763 plotter. The program is operational on a CONTROL DATA CYBER 74 computer. 2 figures, 1 table.

45 (BNWL--2145) MFAULT: A COMPUTER PROGRAM FOR ANALYZING FAULT TREES. Pelto, P.J.; Purcell, W.L. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Nov 1977. Contract EY-76-C-06-1830. 58p. Dep. NTIS, PC A04/MF A01.

A description and user instructions are presented for MFAULT, a FORTRAN computer program for fault tree analysis. MFAULT identifies the cut sets of a fault tree, calculates their probabilities, and screens the cut sets on the basis of specified cut-offs on probability and/or cut set length. MFAULT is based on an efficient upward-working algorithm for cut set identification. The probability calculations are based on the assumption of small probabilities and constant hazard rates (i.e., exponential failure distributions). Cut sets consisting of repairable components (basic events) only, non-repairable components only, or mixtures of both types can be evaluated. Components can be on-line or standby. Unavailability contributions from pre-existing failures, failures on demand, and testing and maintenance down-time can be handled. MFAULT can analyze fault trees with AND gates, OR gates, inhibit gates, on switches (houses) and off switches. The code is presently capable of finding up to ten event cut sets from a fault tree with up to 512 basic events and 400 gates. It is operational on the CONTROL DATA CYBER 74 computer. 11 figures.

46 (BNWL--2146) RAFT: A COMPUTER PROGRAM FOR FAULT TREE RISK CALCULATIONS. Seybold, G.D. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Nov 1977. Contract EY-76-C-06-1830. 67p. Dep. NTIS, MF A01.

Portions of document are illegible.

A description and user instructions are presented for RAFT, a FORTRAN computer code for calculation of a risk measure for fault tree cut sets. RAFT calculates release quantities and a risk measure based on the product of probability and release quantity for cut sets of fault trees modeling the accidental release of radioactive material from a nuclear fuel cycle facility. Cut sets and their probabilities are supplied as input to RAFT from an external fault tree analysis code. Using the total inventory available of radioactive material, along with release fractions for each event in a cut set, the release terms are calculated for each cut set. Each release term is multiplied by the cut set probability to yield the cut set risk measure. RAFT orders the dominant cut sets on the risk measure. The total risk measure of processed cut sets and their fractional contributions are supplied as output. Input options are available to eliminate redundant cut sets, apply threshold values on cut set probability and risk, and control the total number of cut sets output. Hash addressing is used to remove redundant cut sets from the analysis. Computer hardware and software restrictions are given along with a sample problem and cross-reference table of the code. Except for the use of file management utilities, RAFT is written exclusively in FORTRAN language and is operational on a Control Data, CYBER 74-18--series computer system. 4 figures.

47 (BNWL-SA--6064) RISK ASSESSMENT METHOD FOR NUCLEAR FUEL CYCLE OPERATIONS. Pelto, P.J.; Winegardner, W.K. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Mar 1977. Contract EY-76-C-06-1830. 20p. (CONF-770611--39). Dep. NTIS, PC A02/MF A01.

From American Nuclear Society annual meeting; New York, NY, USA (12 Jun 1977).

A method is described for the identification and preliminary evaluation of potential accidents (release sequences) which could lead to the release of radioactive material from nuclear fuel cycle operations. Potential accident sequences are evaluated on the basis of risk. The basic elements of this method are presented along with its application to a conceptual high-level radioactive waste management.

48 (BNWL-SA--6450) PLANNING FOR DECOMMISSIONING AND DECONTAMINATION OF HANFORD NUCLEAR FACILITIES. Litchfield, J.W.; King, J.C. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Sep 1977. Contract EY-76-C-06-1830. 47p. (CONF-771102--26). Dep. NTIS, PC A03/MF A01.

From 70. annual AIChE meeting; New York, NY, USA (13 Nov 1977).

The 570-square mile Hanford Project contains facilities with varying degrees of radioactive contamination as a result of plutonium production operations. With the evolution of production requirements and technology, many of these have been retired and will be decommissioned and decontaminated (D and D). Planning for D and D at Hanford requires identification and characterization of contaminated facilities, prioritization of facilities for decommissioning, selection of D and D modes, estimating costs and other characteristics of D and D activities, definition of future scenarios at Hanford, and preparation and assessment of plans to achieve defined scenarios. A multiattributed decision model using four criteria was used to prioritize facilities for decommissioning. A computer-based interactive planning system was developed to facilitate preparation and assessment of D and D plans.

49 (CONF-790420--) CERAMICS IN NUCLEAR WASTE MANAGEMENT. Chikalla, T.D.; Mendel, J.E. (eds.). (Battelle Pacific Northwest Labs., Richland, WA (USA)). May 1979. Contract EY-76-C-06-1830. 385p. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

Seventy-three papers are included, arranged under the following section headings: national programs for the disposal of radioactive wastes, waste from stability and characterization, glass processing, ceramic processing, ceramic and glass processing, leaching of waste materials, properties of nuclear waste forms, and immobilization of special radioactive wastes. Separate abstracts were prepared for all the papers. (DLC)

50 (CONF-790420--, pp 210-212) VISCOSITY AND ELECTRICAL CONDUCTIVITY OF GLASS MELTS AS A FUNCTION OF WASTE COMPOSITION. Plodinec, M.J.; Wiley, J.R. (E.I. du Pont de Nemours and Co., Aiken, SC). May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

Radioactive waste at the Savannah River Plant contains high concentrations of nonradioactive compounds of iron and aluminum. Simulated waste compositions containing varying ratios of iron to aluminum were added to glass melts to determine the effect on the melt properties. Waste containing high aluminum increased the melt viscosity, but waste containing high iron reduced the melt viscosity. Aluminum and iron both reduced the melt conductivity.

51 (CONF-790420--, pp 198-202) COMPATIBILITY OF ACTINIDES WITH HLW BOROSILICATE GLASS: SOLUBILITY AND PHASE FORMATION. Walker, C.T. (European Inst. for Transuranium Elements, Karlsruhe, Germany); Riege, U. May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

Actinide oxides in the amounts expected in HLW glass are completely soluble in the glass. At 1400°K, the solubility of PuO_2 in borosilicate glass is about 4.5 wt % and is uneffected by the addition of 20 wt % HLW oxides. The presence of 10 wt % Gd_2O_3 gives rise to a segregate phase of the type $Pu_{0.7}Ce_{0.2}Gd_{0.1}O_2$ and hence decreases the solubility of PuO_2 to 1.3 wt %.

52 (CONF-790420--, pp 36-40) DEVELOPMENT OF COMPREHENSIVE WASTE ACCEPTANCE CRITERIA FOR COMMERCIAL NUCLEAR WASTE. O'Hara, F.A.; Miller, N.E.; Ausmus, S.S.; Yates, K.R.; Means, J.L.; Christensen, R.N.; Kulacki, F.A. (Battelle Columbus Labs., OH). May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

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

53 (CONF-790420--, pp 41-46) APPROACH TOWARDS LONG TERM PREDICTION OF STABILITY OF NUCLEAR WASTE FORMS. Yen-Bower, E.L.; Clark, D.E.; Hench, L.L. (Univ. of Florida, Gainesville). May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

An 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 leacning indices and exposure times, illustrates how predictions can be made based on laboratory experiments, field trials, and museum artifacts.

54 (CONF-790420--, pp 52-56) COMPARISON OF GLASS AND CRYSTALLINE WASTE MATERIALS. Ross, W.A.; Turcotte, R.P.; Mendel, J.E.; Rusin, J.M. (Battelle Pacific Northwest Labs., Richland, WA). May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

Either glass or crystalline materials can be shown to have some advantage when a single property is considered. However, the differences are small and each material has both assests and liabilities. With proper design, either type of waste form can almost certainly be utilized to solidify and contain radioactive waste.

55 (CONF-790420--, pp 57-61) GLASSES AS MATERIALS USED IN FRANCE FOR MANAGEMENT OF HIGH-LEVEL WASTES. Bonniaud, R.A.; Jacquet Francillon, N.R.; Laude, F.L.; Sombret, C.G. (Commissariat a l'Energie Atomique, Marcoule, France). May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

Sodium borosilicate and sodium aluminoborosilicate glasses have been selected as final materials in the field of the solidification of high-level radioactive wastes. The properties related to long-term storage are reviewed. The effects of alpha emission upon the structure and of aqueous leaching upon radioactive release are emphasized.

56 (CONF-790420--, pp 62-65) CHARACTERIZATION OF BOROSILICATE GLASSES CONTAINING SIMULATED HIGH-LEVEL RADIOACTIVE WASTES FROM PNC. Terai, R.; Eguchi, K.; Yamanaka, H. (Government Industrial Research Inst., Osaka, Japan). May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

The characterization of borosilicate glasses containing simulated HLW from PNC has been carried out. Phase separation of molybdates, volatilization, viscosity, electrical resistivity, thermal conductivity, elastic modulus, chemical durability, and devitrification of these glasses have been measured, and the suitability of the glasses for the vitrified solidification processes is discussed from the viewpoint of safety.

57 (CONF-790420--, pp 203-209) INVESTIGATION OF CRYSTALLIZATION IN GLASSES CONTAINING FISSION PRODUCTS. Malow, G. (Hahn-Meitner-Institut fuer Kernforschung, Berlin, Germany). May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

Five potential solidification products for high-level waste (four borosilicate glasses and one celsian glass ceramic) have been investigated in terms of crystallization. In all glasses and in the glass ceramic, crystallization, and recrystallization, respectively, were observed by heating above 773°K, however, at very different periods of time (0.1d greater than or equal to 100d). The noble metals precipitated into various phases. Crystal growth proceeded at the phase boundary glass-noble metal. In all products rare earth phases crystallized. Silicate phases rarely formed. The leach resistance (by the grain titration and Soxhlet tests) decreased after heat treatment in all cases. The changes were found to be within one order of magnitude for all products. 2 figures, 4 tables.

58 (CONF-790420--, pp 213-217) EFFECTS OF COMPOSITION ON WASTE GLASS PROPERTIES. Mellinger, G.B.; Chick, L.A. (Battelle Pacific Northwest Labs., Richland, WA). May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

The electrical conductivity, viscosity, chemical durability, devitrification, and crystallinity of a defense waste glass were measured. Each oxide component in the glass was varied to determine its effect on these properties. A generic study is being developed which will determine the effects of 76 oxides on the above and additional properties of a wide field of possible waste glasses. 5 figures, 2 tables.

59 (CONF-790420--, pp 224-228) PELLETED WASTE FORM FOR HIGH-LEVEL ICPP WASTES. Lamb, K.M.; Priebe, S.J.; Cole, H.S.; Taki, B.d. (Allied Chemical Corp., Idaho Falls, ID). May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

Simulated zirconia-type calcined waste is pelletized on a 41-cm diameter disc pelletizer using 5% bentonite, 2% metakaolin, and 2% boric acid as a solid binder and 7M phosphoric plus 4M nitric acid as a liquid binder. After heat treatment at 800°C for 2 hours the pellets are impact resistant and have a leach resistance of 10^{-4} g/cm_2 . day, based on Soxhlet leaching for 100 hours at 95°C with distilled water. An integrated pilot plant is being fabricated to verify the process. 1 figure, 4 tables.

60 (CONF-790420--, pp 233-237) RELATIVE LEACH BEHAVIOR OF WASTE GLASSES AND NATURALLY OCCURRING GLASSES. Adams, P.B. (Corning Glass Works, NY). May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

Simulated nuclear waste glasses of the sodium-borosilicate type with a low waste loading and of the zinc-borosilicate type with a high waste loading have been compared with obsidians. The resuls indicate that the waste glasses would corrode in normal natural environments at a rate of about 0.1 µm per year at 30°C and about 5 µm per year at 90°C, compared with obsidians which seem to corrode at, or less than, about 0.01 µm per year at 30°C and less than 1 µm per year at 90°C. Activation energies for reactions of the two waste glasses with pure water are about 20 kcal/g-mol. 3 figures, 7 tables.

61 (CONF-790420--, pp 289-293) SIMULATIONS OF RADIATION DAMAGE IN GLASSES. Antonini, M.; Lanza, F.; Manara, A. May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

Simulation of the damage produced in glasses by alpha emitters has been performed using fission fragments and accelerated Ni ions. As a unit of damage the number of displaced atoms has been assumed. Stored energy was considered as a basic quantity for verification. The results obtained compare very well with the existing data. 2 figures, 4 tables.

62 (CONF-790420--, pp 294-299) RADIATION EFFECTS IN VITREOUS AND DEVITRIFIED SIMULATED WASTE GLASS. Weber, W.J.; Turcotte, R.P.; Bunnell, L.R.; Roberts, F.P.; Westsik, J.H. Jr. (Battelle Pacific Northwest Labs., Richland, WA). May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

The long-term radiation stability of vitreous and partially devitrified forms of high-level waste glass was investigated in accelerated experiments by ^{244}Cm doping. The effects of radiation on microstructure, phase behavior, density, impact strength, stored energy, and leachability are reported to a cumulative radiation dose of 5 x 10^{18} α decays/cm^3. This dose produces saturation of radiation effects in most properties. 4 figures.

63 (CONF-790420--, pp 300-304) EFFECTS OF IRRADIATION ON STRUCTURAL PROPERTIES OF CRYSTALLINE CERAMICS. Clinard, F.W. Jr.; Hurley, G.F. (Los Alamos Scientific Lab., NM). May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

Stability of crystalline ceramic nuclear waste may be degraded by self-irradiation damage. Changes in density, strength, thermal conductivity, and lattice structure are of concern. In this paper, structural damage of ceramics under various radiation conditions is discussed and related to possible effects in nuclear waste.

64 (CONF-790420--, pp 315-320) CRYSTAL CHEMISTRY OF THE SYNTHETIC MINERALS IN CURRENT SUPERCALCINE-CERAMICS. McCarthy, G.J. (Pennsylvania State Univ., University Park); Pepin, J.G.; Pfoertsch, D.E.; Clarke, D.R. May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

Crystal chemistry of the synthetic mineral phase assemblages in current supercalcine-ceramics has been investigated. X-ray diffraction and a variety of electron optical tools (SEM, STEM, electron microprobe) have been used to characterize the crystal structure types and solid solution substitutions.

65 (CONF-790420--, pp 327-332) EFFECT OF IMPACT ENERGY ON SOLID-WASTE COMPOSITES OF BRITTLE AND DUCTILE MATERIALS. Mecham, W.J.; Jardine, L.J.; Steindler, M.J. (Argonne National Lab., IL). May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

A method 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. This latter form is also referred to as a metal-matrix waste form. Relevant methods, models, and methodologies reported in the literature for impact analyses of solid radioactive waste forms are summarized. Preliminary descriptive and calculational models are outlined for estimating and for comparing the impact deformations. The models are based on energy balances and are proposed to serve as a basis for design and testing of solid waste forms.

66 (CONF-790420--, pp 305-309) METAMICT STATE RADIATION DAMAGE IN CRYSTALLINE MATERIALS. Haaker, R.F.; Ewing, R.C. (Univ. of New Mexico, Albuquerque). May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

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 summarizes pertinent literature on metamictization and proposes experiments that are critical to the elucidation of structural controls on radiation damage in crystalline phases.

67 (CONF-790420--, pp 310-314) RADIATION DAMAGE STUDIES ON SYNTHETIC NACL CRYSTALS AND NATURAL ROCK SALT FOR WASTE DISPOSAL APPLICATIONS. Klaffky, R.W. (Brookhaven National Lab., Upton, NY); Swyler, K.J.; Levy, P.W. May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

Radiation damage studies are being made on synthetic NaCl and natural rock salt crystals from various localities, including potential repository sites. Measurements are being made with equipment for recording the radiation-induced F-center and colloid particle absorption bands during irradiation with 1.5 MeV electrons at various temperatures. A technique has been develcped to resolve the overlapping F-center and colloid bands. The resulting spectra and curves of absorption verus dose provide information on colloid particle size and concentration and activation energies for processes occurring during colloid formation; additional data suggest that both strain and radiation-induced dislocations contribute to the colloid formation process. 5 figures, 1 table.

68 (CONF-790420--, pp 13-16) CONCEPT FOR THE TREATMENT AND DISPOSAL OF RADIOACTIVE WASTES IN THE GERMAN ENTSORGUNGSZENTRUM. Selander, C.; Zuehlke, P. (Deutsche Gesellschaft fuer Wiederaufarbeitung von Kernbrennstoffen mbH, Hannover, Germany). May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

The concept for treatment and disposal of radioactive wastes in the planned German Entsorgungszentrum is presented. HLLW will be converted to borosilicate glass. Medium- and low-level-liquid waste concentrates will be fixed by cementation. The products will be finally disposed of in a salt dome just below the Entsorgungszentrum.

69 (CONF-790420--, pp 27-32) INFLUENCE OF SYSTEM CONSIDERATIONS ON WASTE FORM DESIGN. Bauer, A.A.; Matthews, S.C.; Peterson, R.W. (Office of Nuclear Waste Isolation, Columbus, OH). May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

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.

70 (CONF-790420--, pp 33-35) IAEA COORDINATED RESEARCH PROGRAM ON THE EVALUATION OF SOLIDIFIED HIGH-LEVEL RADIOACTIVE WASTE PRODUCTS. Grover, J.R.; Schneider, K.J. (International Atomic Energy Agency, Vienna, Austria). May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

A coordinated research program on the evaluation of solidified high-level radioactive waste products has been active with the IAEA since 1976. The program's objectives are to integrate research and to provide a data bank on an international basis in this subject area. Results and considerations to date are presented.

71 (CONF-790420--, pp 218-223) PHASE BOUNDARY EFFECTS IN METAL MATRIX EMBEDDED GLASSES. Schiewer, E. (Hahn-Meitner-Institut fuer Kernforschung GmbH, Berlin, Germany). May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

An investigation was performed to study reactions at the phase bounderies of glass-lead composites at temperatures up to the softening point of the glass. Some metal was oxidized at the boundary and penetrated into the glass. Solid-state diffusion was rate controlling. In the case of a phosphate glass, fission products were depleted in the boundary area. Molybdenum migrated into the lead, and cesium migrated into the glass core. 2 figures, 3 tables.

72 (CONF-790420--, pp 333-337) STRESS

ANALYSIS OF GLASS-CANISTER INTERACTION: A STUDY OF RESIDUAL STRESSES AND FRACTURING. Simonen, F.A.; Friley, J.R. (Battelle Pacific Northwest Labs., Richland, WA). May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

Residual stresses and cracking in canisters filled with vitrified nuclear waste are simulated using finite element computer calculations. Cooling rates, internal 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 Sartenev.

73 (CONF-790420--, pp 365-369) CHARACTERISTICS OF STORED HIGH-LEVEL ICPP WASTE CALCINE. Staples, B.A.; Pomiak, G.S.; Wade, E.L. (Allied Chemical Corp., Idaho Falls, ID). May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

Calcined high-level radioactive defense waste stored at the Idaho Chemical Processing Plant for up to twelve years was sampled for retrievability. The calcine was determined to be retrievable as expected. A program to determine physical and chemical characterisitcs for future retrieval and alternative waste treatment processes has been completed.

74 (EPRI-NP--1087) NUCLEAR WASTE MANAGEMENT STATUS AND RECENT ACCOMPLISHMENTS. FINAL REPORT. McElroy, J.L.; Burns, R.E. (Battelle Pacific Northwest Labs., Richland, WA (USA)). May 1979. Contract EY-76-C-06-1830. 86p. Dep. NTIS, PC A05/MF A01.

The status and technical progress of nuclear waste disposal is reviewed with emphasis on technical and programmatic progress in High Level Nuclear Waste disposal technology during the 1976 to 1978 time period. Process steps in the waste solidification and geologic disposal system are described emphasizing processes and systems that are more advanced in terms of readiness for full-scale U.S. demonstration. Worldwide technical accomplishments in support of the reference US waste vitrification and geologic disposal approach are highlighted.

75 (NUREG/CR--0129(Vol.1)) TECHNOLOGY, SAFETY AND COSTS OF DECOMMISSIONING A REFERENCE SMALL MIXED OXIDE FUEL FABRICATION PLANT. VOLUME 1. MAIN REPORT. Jenkins, C.E.; Murphy, E.S.; Schneider, K.J. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jan 1979. Contract EY-76-C-06-1830. 350p. Dep. NTIS, PC A15/MF A01.

Detailed technology, safety and cost information are presented for the conceptual decommissioning of a reference small mixed oxide fuel fabrication plant. Alternate methods of decommissioning are described including immediate dismantlement, safe storage for a period of time followed by dismantlement and entombment. Safety analyses, both occupational and public, and cost evaluations were conducted for each mode.

76 (NUREG/CR--0129(Vol.2)(App.)) TECHNOLOGY, SAFETY AND COSTS OF DECOMMISSIONING A REFERENCE SMALL MIXED OXIDE FUEL FABRICATION PLANT. VOLUME 2. APPENDICES. Jenkins, C.E.; Murphy, E.S.;Schneider, K.J. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jan 1979. Contract EY-76-C-06-1830. 375p. Dep. NTIS, PC A16/MF A01.

Volume 2 contains appendixes on small MOX fuel fabrication facility description, site description, residual radionuclide inventory estimates, decommissioning, financing, radiation dose methodology, general considerations, packaging and shipping of radioactive materials, cost assessment, and safety (JRD)

77 (ORP/CSD--77-2, pp 1.47-1.49) ENVIRONMENTAL IMPACT STATEMENT, COMMERCIAL RADIOACTIVE WASTE MANAGEMENT. Unruh, C.M. (Battelle Pacific Northwest Labs., Richland, WA). 1977.

From Workshop on policy and technical issues pertinent to the development of environmental protection criteria for radioactive wastes; Albuquerque, NM, USA (12 Apr 1977).

A brief description is given of the ERDA's preparation of a generic environmental impact statement for waste from the LWR fuel cycle. (DLC)

78 (PNL--2378-3) NUCLEAR WASTE MANAGEMENT. QUARTERLY PROGRESS REPORT, JULY--SEPTEMBER 1978. Platt, A.M.; Powell, J.A. (comps.). (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jan 1979. Contract EY-76-C-06-1830. 88p. Dep. NTIS, PC A05/MF A01.

Research on the following is reported: decontamination and densification of chop-leach cladding residues; monitoring methods for particulate and gaseous effluents from waste solidification processes; TRU waste immobilization; krypton solidification; ^{14}C and ^{129}I fixation; waste management system studies; waste management safety studies; waste isolation safety assessment; well logging instrumentation for shallow land burial; monitoring and physical characteriztion of unsaturated zone transport; detection and characterization of mobile organic complexes of fission products; and electropolishing for surface decontamination of metals. 21 figures, 17 tables.

79 (PNL--2400) NONTECHNICAL ISSUES IN WASTE MANAGEMENT: ETHICAL, INSTITUTIONAL, AND POLITICAL CONCERNS. Hebert, J.A.; Rankin, W.L.; Brown, P.G.; Schuller, C.R; Smith, R.F.; Goodnight, J.A.; Lippek, H.E. (Battelle Pacific Northwest Labs., Richland, WA (USA)). May 1978. Contract EY-76-C-06-1830. 116p. Dep. NTIS, PC A06/MF A01.

The report consists of a presentation and distillation of major nontechnical issues surrounding commercial waste management, followed by ethical, institutional, and political analyses of these issues. The ethical analysis consists of a discusson of what is meant by ''ethics'' and ''morality'' in the waste management context and an illustrative attempt at an ethical analysis of the commercial nuclear waste problem. Two institutional analyses are presented: one is an analysis of the possible problems of long-term human institutions in waste management; the other is a presentation of institutional arrangements for the short term. A final chapter discusses issues and concerns involving intergovernmental relations--that is, local, state, and federal interface problems in waste management.

80 (PNL--2598) SUMMARY OF NATIONAL AND INTERNATIONAL RADIOACTIVE WASTE MANAGEMENT PROGRAMS (EXCLUDING UNITED STATES). Harmon, K.M. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Mar 1978. Contract EY-

76-C-06-1830. 54p. Dep. NTIS, PC A04/MF A01.
This report is divided into two parts. Various national fuel cycle and waste management programs are summarized in tabular form in the first part. The second part of the report gives a nation-by-nation overview of fuel cycle and waste management technologies. Brief summaries on the activities of several international organizations are included. (LK)

81 (PNL--2668-2) MULTIBARRIER WASTE FORMS. PART II. CHARACTERIZATION AND EVALUATION. Rusin, J.M.; Grey, W.J.; Wald, J.W. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Aug 1979. Contract EY-76-C-06-1830. 66p. Dep. NTIS, PC A04/MF A01.
The multibarrier concept for the storage of radioactive waste is to use up to three barriers to isolate radionuclides from the environment: a solidified waste inner core, an impervious coating, and a matal matrix. The four multibarrier waste forms were evaluated for thermal stability (volatility), mechanical strength (impact resistance), and leach resistance. This report discusses the characterization of the multibarrier waste forms and compares them to reference calcine and glass waste forms. The weight loss of supercalcine-ceramics after 4 h in dry air ranges between 0.01 and 1.6 wt % from 1000 to 1200°C and is dependent upon composition. Glass marbles in a cast lead alloy offer approximately an order of magnitude decease in the wt % fines < 37 μm released after impact as compared to a glass monolith. CVD-coated supercalcine in a sintered 410 SS matrix offers up to two orders of magnitude decrease. Hot-pressed supercalcine ceramics may offer no increase in impact resistance or leach resistance over that of a glass monolith. Supercalcine may offer no advantage over waste glasses in leach resistance. Glass and PyC/Al_2O_3 coatings provide effective inert leaching barriers.

82 (PNL--2672) ISOLATION OF PLUTONIUM PHYSICAL--CHEMICAL STATES FROM NATURAL WATERS. Weimer, W.C. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Aug 1978. Contract EY-77-C-06-1030;EY-76-C-06-1830. 29p. Dep. NTIS, PC A03/MF A01.
The purpose of this research program was to evaluate the feasibility, on a bench scale, of methods for preconcentrating selectively individual plutonium forms from very dilute natural water samples, and to apply these results to use with the Battelle large volume water sampler. From the results of the current investigations, several alternative water sampling strategies have been recommended. The preferred water sampling technique has been field tested at several groundwater wells in the 200 East and 200 West areas of the U.S. Department of Energy Hanford Reservation. These laboratory investigations, in combination with field testing of the proposed water sampling techniques, have yielded the following conclusions: (1) The use of polypropylene microporous filters (0.04μ pore size) in conjunction with glass fiber filters (3.0μ pore size) enables the characterization of two size fractions of particulate plutonium forms in groundwater samples. Those species which pass the microporous polypropylene filters are considered to be in solution. (2) The sorption and ion exchange media evaluated do not show the selectivity necessary to allow preconcentration of individual plutonium forms from natural water samples by any of these media beds under the conditions evaluated. (3) Al_2O_3 is the most effective sorption media that was examined for removing any plutonium species from natural water samples at neutral pH values. On the basis of these investigations, a standard field testing methodology has been proposed for sampling ground waters near nuclear waste management areas. Additional laboratory evaluations of plutonium species interactions with sorption and ion exchange media have also been recommended.

83 (PNL--2727) SAFETY INDICES AND THEIR APPLICATION TO NUCLEAR WASTE MANAGEMENT SAFETY ASSESSMENT. Voss, J.W. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Apr 1979. Contract EY-76-C-06-1830. 87p. Dep. NTIS, PC A05/MF A01.
Thirteen indices have been examined to determine their potential applicability to operational waste management safety assessment. Two waste streams are presented on a normalized basis. These are the packaged spent fuel from the once-through fuel cycle and solidified high-level waste from a coprocessed UO_2--PuO_2 fuel cycle. Seven of the indices are then calculated for a hypothetical surface storage scenario of the two wastes. The indices are examined on a consistent basis to identify any biases built into them, and to determine the sensitivities of each to various waste situations. The two waste streams are then compared on the basis of the indices to extend the understanding of the analysis techniques. The results of the analysis fall into two categories, index evaluation and fuel cycle waste comparison. Only five of the indices are determined to be applicable to operational waste management safety assessment. The remainder are rejected either because they require very detailed input data; they are specifically designed for geologic isolation; they are extremely controversial in their application; or because they are particularly sensitive to a few specific radionuclides. The waste stream comparison yields three results: (1) the solidified high-level waste from the coprocessed UO_2--PuO_2 fuel cycle may be potentially less hazardous than the packaged spent fuel from the once-through fuel cycle; (2) the removal of actinides, and especially plutonium, from spent fuel may reduce the potent hazard associated with the waste; and (3) after one million years of decay, the packaged spent fuel and solidified high-level waste are nearly the same on a hazard potential basis.

84 (PNL--2751) COMPATIBILITY OF TWO IDAHO CHEMICAL PROCESSING PLANT GLASSES WITH ELECTRIC MELTING PROCESSES. Lukacs, J.M.; Bates, J.L.; Devine, J.R.; Gray, W.J.; Weber, W.J. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Dec 1978. Contract EY-76-C-06-1830. 49p. Dep. NTIS, PC A03/MF A01.
High-level liquid nuclear wastes produced at the Idaho Chemical Processing Plant (ICPP) was converted to a dry calcine powder. The feasibility of converting this calcine to a durable waste glass is being evaluated at ICPP. Candidate waste glass compositions were developed and plans were made to construct and operate a laboratory electric glass melter at INEL. The Pacific Northwest Laboratory (PNL) examined the electric melter processing characteristics of two Idaho developed waste glasses, ICPP Zr-13 and ICPP Zr-51. This preliminary evaluation found both borosilicate glasses acceptable candidates for electric melting trials. The high temperature resistivity, viscosity, and corrosion behaviors are consistent with PNL melting experience. The ICPP calcine characteristics, waste glass volatility, and waste glass devitrification

behavior have not been previously encountered at PNL. Several melter operation and design options are available to deal with all of these conditions. Additional work in three basic areas is, however, suggested for the final candidate waste glass composition: volatility - species identification; corrosion behavior of electrodes at operating temperatures; and means of improving melting rates and glass homogeneity. Reducing the devitrification tendency of the candidate waste glass will significantly improve the stability and control of an electric glass melting process.

85 (PNL--2764) STORAGE AND DISPOSAL OF RADIOACTIVE WASTE AS GLASS IN CANISTERS. Mendel, J.E. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Dec 1978. Contract EY-76-C-06-1830. 97p. Dep. NTIS, PC A05/MF A01.

A review of the use of waste glass for the immobilization of high-level radioactive waste glass is presented. Typical properties of the canisters used to contain the glass, and the waste glass, are described. Those properties are used to project the stability of canisterized waste glass through interim storage, transportation, and geologic disposal.

86 (PNL--2826) COMPARISON OF SPENT FUEL MANAGEMENT FEE COLLECTION ALTERNATIVES. White, M.K.; Engel, R.L. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jan 1979. Contract EY-76-C-06-1830. 52p. Dep. NTIS, PC A04/MF A01.

Five alternative methods for recovering the costs of spent fuel management were evaluated. These alternatives consist of collecting the fee for various components of spent fuel management cost (AFR basin storage, transportation from AFR basin to the repository, packaging, repository, R and D, and government overhead) at times ranging from generation of power to delivery of the spent fuel to the government. The five fee collection mechanisms were analyzed to determine how well they serve the interests of the public and the electricity ratepayer. (DLC)

87 (PNL--2941) SUMMARY OF NATIONAL AND INTERNATIONAL RADIOACTIVE WASTE MANAGEMENT PROGRAMS. Harmon, K.M. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Mar 1979. Contract EY-76-C-06-1830. 84p. Dep. NTIS, PC A05/MF A01.

This report summarizes information collected on the status of fuel cycle and waste management programs in Argentina, Australia, Austria, Belgium, Brazil, Canada, China, Denmark, Finland, France, Democratic Republic of Germany, Federal Republic of Germany, India, Iran, Italy, Japan, Mexico, the Netherlands, Pakistan, Spain, Sweden, Switzerland, United Kingdom, United States, and USSR. This compilation attempts to provide current information as of the end of January 1979.

88 (PNL--3036) STRESS ANALYSIS OF HIGH-LEVEL WASTE CANISTERS: METHODS, APPLICATIONS, AND DESIGN DATA. Simonen, F.A.; Slate, S.C. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Oct 1979. Contract EY-76-C-06-1830. 156p. Dep. NTIS, PC A08/MF A01.

An overview of stress analysis methods, structural design procedures, and design data is presented for canisters used to package solidified wastes, particularly borosilicate glass. In addition, 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. Existing sections of ASME Boiler and Pressure Vessel Code are applicable to canister fabrication, but the code does not cover many aspects of canister service loadings. 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. Methods to predict residual stresses in the walls of waste canisters are described; predicted residual stress levels agree with measured stress levels. The consequences of these 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. It is shown that the methods of fracture mechanics and fatigue-crack-growth analyses are valuable tools for evaluating the performance of closure welds in the presence of crack-like defects. Canister performance in process trials at PNL shows the ability of canisters to survive high temperatures and loadings during processing. Impact tests show that a suitably designed canister can sustain severe impacts without loss of intergrity.

89 (PNL--3078) NUCLEAR WASTE MANAGEMENT ISSUES: A MULTIDISCIPLINARY EVALUATION FRAMEWORK. TEKNEKRON B-52865-A-L. Hoffman, M. (Teknekron, Inc., Berkeley, CA (USA)). Feb 1980. Contract EY-76-C-06-1830. 90p. Dep. NTIS, PC A05/MF A01.

Initially, this paper characterizes the nuclear waste problem that requires analysis to establish the rationale for an interdisciplinary approach to resolve it. The problem characterization also explains why the specific concern with contaminated groundwater and intrusion through drilling has been selected for the focus of the panel meeting. The Nominal Group Technique (NGT), the group process format chosen for the experts' deliberations, is explained in some detail and its value in facilitating the desired dialogue is described. The dialogue is organized around the various issue areas that would be of concern to a program manager dealing with the potential problem of radioactivity escaping to the biosphere through human intrusion into contaminated groundwater. The participants are identified by professional discipline so that the dialogue can be presented in a realistic fashion. Both the content of the dialogue and its format are evaluated. Particular attention is given to their usefulness in generating a cross-section of subissues and factors that should be addressed when analyzing the waste disposal system's adequacy to prevent contaminated groundwater escaping to the biosphere.

90 (PNL--3152) LEACHING OF ACTINIDES AND TECHNETIUM FROM SIMULATED HIGH-LEVEL WASTE GLASS. Bradley, D.J.; Harvey, C.O.; Turcotte, R.P. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Aug 1979. Contract EY-76-C-06-1830. 104p. Dep. NTIS, PC A06/MF A01.

Leach tests were conducted using a modified version of the IAEA procedure to study the behavior of glass waste-solution interactions. Release rates were determined for Tc, U, Np, Pu, Am, Cm, and Si in the following solutions: WIPP B salt brine, NaCl (287 g/l), NaCl (1.76 g/l), $CaCl_2$ (1.66 g/l), $NaHCO_3$ (2.52 g/l), and deionized water. The leach rates for all elements decreased an order of magnitude from their initial values during the first 20 to 30 days leaching time. The sodium bicarbonate solution produced the highest elemental release rates, while the saturated salt brine and deionized water in general gave the lowest release. Technetium has the highest initial release of all elements studied. The technetium release rates, however, decreased by over four orders of magnitude in 150 days of leaching time. In the prepared glass, technetium was phase separated, concentrating on internal pore surfaces. Neptunium, in all cases except $CaCl_2$ solution, shows the highest actinide release rate. In general, curium and uranium have the lowest release rates. The range of actinide release rates is from 10^{-5} to 10^{-8} g/cm²/day. 25 figures, 7 tables.

91 (PNL--3176) ^{85}KR MANAGEMENT TRADE-OFFS: A PERSPECTIVE TO TOTAL RADIATION DOSE COMMITMENT. Mellinger, P.J.; Hoenes, G.R.; Brackenbush, L.W.; Greenborg, J. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jan 1980. Contract EY-76-C-06-1830. 126p. Dep. NTIS, PC A07/MF A01.

Radiological consequences arising from the trade-offs for ^{85}Kr waste management from possible nuclear fuel resource recovery activities have been investigated. The reference management technique is to release all the waste gas to the atmosphere where it is diluted and dispersed. A potential alternative is to collect, concentrate, package and submit the gas to long-term storage. This study compares the radiation dose commitment to the public and to the occupationally exposed work force from these alternatives. The results indicate that it makes little difference to the magnitude of the world population dose whether ^{85}Kr is captured and stored or chronically released to the environment. Further, comparisons of radiation exposures (for the purpose of estimating health effects) at very low dose rates to very large populations with exposures to a small number of occupationally exposed workers who each receive much higher dose rates may be misleading. Finally, cost studies (EPA 1976 and DOE 1979a) show that inordinate amounts of money will be required to lower this already extremely small 80-year cumulative world population dose of 0.05 mrem/person (<0.001% of natural background radiation for the same time period).

92 (PNL--3254) MATERIALS CHARACTERIZATION CENTER PROGRAM PLAN. Nelson, R.D.; Ross, W.A.; Hill, O.F.; Mendel, J.E.; Merz, M.D.; Turcotte, R.P. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Mar 1980. Contract EY-76-C-06-1830. 82p. Dep. NTIS, PC A05/MF A01.

The Materials Characterization Center (MCC) has been established at Pacific Northwest Laboratory as part of the Materials Characterization Organization for providing an authoritative, referenceable basis for establishing nuclear waste material properties and test methods. The MCC will provide a data base that will include information on the components of the waste emplacement package - the spent fuel or processed waste form and the engineered barriers - and their interaction with each other and as affected by the environment. The MCC will plan materials testing, develop and document procedures, collect and analyze existing materials data, and conduct tests as necessary.

93 (PNL--3300(Pt.2), pp 161) DECOMMISSIONING AND DECONTAMINATION (BURIAL GROUND STABILIZATION) STUDIES. Cline, J.F. Feb 1980. Dep. NTIS, PC A11/MF A01.

Pacific Northwest Laboratory annual report for 1979 to the DOE Assistant Secretary for Environment. Part 2. Ecological sciences.

The decommissioning and decontamination of retired Hanford facilities and the future use of surrounding landscapes require isolation of contaminated wastes from the biosphere. Burial ground stabilization studies were conducted to determine the effectiveness of physical barriers for isolating contaminated wastes in shallow-land burial sites from plants and animals. This study was undertaken to determine the effectiveness of using a layer of loose rock between the waste and the surface soil covering to prevent both plant root and animal penetrations.

94 (PNL--3317-1) COMMERCIAL NUCLEAR REACTORS AND WASTE: THE CURRENT STATUS. Platt, A.M.; Robinson, J.V. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Apr 1980. Contract EY-76-C-06-1830. 55p. Dep. NTIS, PC A04/MF A01.

During the last five years, the declared size of the commercial light water reactor (LWR) nuclear power industry in the US has steadily decreased. As of January 1980, the total number of power plants had dropped to 191 from the 226 in December 31, 1974. At least another nine were cancelled in the last few months. This report was developed as the first of a series to track implications to waste management due to such changes in the declared size of the industry. For the presently declared size, key conclusions are: the declared reactors will peak at a capacity of 162 GWe and consume about 10^6 MTU as enrichment feed. As few as two repositories of about 100,000 MTHM capacity each would hold the waste. Predisposal storage (reactor basins and AFRs) would peak at less than 100,000 MTHM (in the year 2020) with one repository opening in the year 1997 and the other in the year 2020. Most of the 100,000 MTHM would have to be in AFR storage unless current practice regarding reactor basin size was radically changed.

95 (PNL--3401) SOCIAL ISSUES AND ENERGY ALTERNATIVES: THE CONTEXT OF CONFLICT OVER NUCLEAR WASTE. FINAL REPORT. Lindell, M.K.; Earle, T.C.; Perry, R.W. (Battelle Human Affairs Research Center, Seattle, WA (USA)). Jun 1980. Contract AC06-76RL01830. 67p. (B-HARC--411/80/033). NTIS, PC A04/MF A01.

The perceived risks and benefits of electric power alternatives were used to explore the context of attitudes toward nuclear power. Supporters and opponents of nuclear power responded to thirty-three items which referred to five categories of energy issue: the production potential of electric, risks of those technologies, power generation technologies, energy conservation, comparisons of risks among technologies and comparisons between risks and benefits of each technology. The results are summarized. The nuclear supporters studied here do favor nuclear power. However, they believe that there are limited prospects for contributions from solar, wind and hydroelectric technologies. They also believe that there are serious disadvantages to conservation. Nuclear opponents, on the other hand, disagree that there are such limited prospects for solar and wind, although they are neutral on the prospects for increased hydro capacity. They also do not believe that conservation necessarily poses serious adverse consequences either for themselves or others.

96 (PNL--3404) ATTUTUDE-ACTION CONSISTENCY AND SOCIAL POLICY RELATED TO NUCLEAR TECHNOLOGY. Lindell, M.K.; Perry, R.W.; Greene, M. (Battelle Human Affairs Research Center, Seattle, WA (USA)). Jun 1980. Contract AC06-76RL01830. 27p. (B-HARC--411/80/030). NTIS, PC A03/MF A01.

This study reports the results of a further analysis of questionnaire data--parts of which have been previously reported by Lindell, Earle, Hebert and Perry (1978)--that are related to the issue of consistency of attitudes and behavior toward nuclear power and nuclear waste management. Three factors are considered that might be expected to have a significant bearing on attitude-action consistency: social support, attitude object importance and past activism. Analysis of the data indicated that pronuclear respondents were more likely to show consistency of attitudes and actions (66%) than were antinuclear respondents (51%) although the difference in proportions is not statistically significant. Further analyses showed a strong positive relation between attitude-action consistency and perceived social support, measured by the degree to which the respondent believed that close friends and work associated agreed with his attitude. This relationship held up even when controls for attitude object importance and past activism were introduced. Attitude object importance--the salience of the issue of energy shortage--had a statistically significant effect only when perceived social support was low. Past activism had no significant relation to attitude-action consistency. These data suggest that the level of active support for or opposition to nuclear technology will be affected by the distribution of favorable and unfavorable attitudes among residents of an area. Situations in which pro- and antinuclear attitudes are concentrated among members of interacting groups, rather than distributed randomly, are more likely to produce high levels of polarization.

97 (PNL--3425) WASTES FROM SELECTED ACTIVITIES IN TWO LIGHT-WATER REACTOR FUEL CYCLES. Palmer, C.R.; Hill, O.F. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jul 1980. Contract AC06-76RL01830. 45p. NTIS, PC A03/MF A01.

This report presents projected volumes and radioactivities of wastes from the production of electrical energy using light-water reactors (LWR). The projections are based upon data developed for a recent environmental impact statement in which the transuranic wastes (i.e., those wastes containing certain long-lived alpha emitters at concentrations of at least 370 becquerels, or 10 nCi, per gram of waste) from fuel cycle activities were characterized. In addition, since the WG.7 assumed that all fuel cycle wastes except mill tailings are placed in a mined geologic repository, the nontransuranic wastes from several activities are included in the projections reported. The LWR fuel cycles considered are the LWR, once-through fuel cycle (Strategy 1), in which spent fuel is packaged in metal canisters and then isolated in geologic formations; and the LWR U/Pu recycle fuel cycle (Strategy 2), wherein spent fuel is reprocessed for recovery and recycle of uranium and plutonium in LWRs. The wastes projected for the two LWR fuel cycles are summarized. The reactor operations and decommissioning were found to dominate the rate of waste generation in each cycle. These activities account for at least 85% of the fuel cycle waste volume (not including head-end wastes) when normalized to per unit electrical energy generated. At 10 years out of reactor, however, spent fuel elements in Strategy 1 represent 98% of the fuel cycle activity but only 4% of the volume. Similarly, the packaged high-level waste, fuel hulls and hardware in Strategy 2 concentrate greater than 95% of the activity in 2% of the waste volume.

98 (PNL--3522) LOW LEACH RATE GLASSES FOR IMMOBILIZATION OF NUCLEAR WASTES. Chick, L.A.; Buckwalter, C.Q. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Oct 1980. Contract AC06-76RL01830. 27p. NTIS, PC A03/MF A01.

Improved defense and commercial waste glass have about one order of magnitude lower leach rates at 90°C in static deionized water than reference glasses. This durability difference diminishes as the leaching temperature is raised, but at repository temperature less than 150°C, the improved compositions would have considerable advantages over reference glasses. At the melting temperatures necessary for most of the high-durability glasses, volatility was found to be higher than that experienced in processing current reference glases. Higher volatilities might be compensated for by specific design of the off-gas system for improved off-gas treatment and volatile materials recovery. 6 figures, 2 tables.

99 (PNL--3564) ISSUES AND SCENARIOS FOR NUCLEAR WASTE MANAGEMENT SYSTEMS ANALYSIS. Mendel, J.E. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Nov 1980. Contract AC06-76RL01830. 18p. NTIS, PC A02/MF A01.

The Planning and Analysis Branch of the Department of Energy's Nuclear Waste Management Programs is developing a new systems integration program. The Pacific Northwest Laboratory was requested to perform a brief scoping analysis of what scenarios, questions, and issues should be addressed by the systems integration program. This document reports on that scoping analysis.

100 (PNL--3588) MATERIALS CHARACTERIZATION CENTER WORKSHOP ON THE IRRADIATION EFFECTS IN NUCLEAR WASTE FORMS. Roberts, F.P.; Turcotte, R.P.; Weber, W.J. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jan 1981. Contract AC06-76RL01830. 77p. NTIS, PC A05/MF A01.

The workshop on Irradiation Effects in Nuclear Waste Forms sponsored by the Materials Characterization Center (MCC) brought together experts in radiation damage in materials and waste-management technology to review the problems associated with irradiation effects on waste-form integrity and to evaluate standard methods for generating data to be included in the Nuclear Waste Materials Handbook. The workshop reached the following conclusions: the concept of Standard Test for the Effects of Alpha-Decay in Nuclear Waste Solids, (MCC-6) for evaluating the effects of alpha decay is valid and useful, and as a result of the workshop, modifications to the proposed procedure will be incorpoated in a revised version of MCC-6; the MCC-6 test is not applicable to the evaluation of radiation damage in spent fuel; plutonium-238 is recommended as the dopant for transuranic and defense high-level waste forms, and when high doses are required, as in the case of commercial high-level waste forms, ^{244}Cm can be used; among the important property changes caused by irradiation are those that lead to greater leachability, and additionally, radiolysis of the leachant may increase leach rates; research is needed in this area; ionization-induced changes in physical properties can be as important as displacement damage in some materials, and a synergism is also likely to exist from the combined effects of ionization and displacement damage; and the effect of changing the temperature and dose rates on property changes induced by radiation damage needs to be determined.

101 (PNL--3700(Pt.2), pp 21-23) APPLICATION OF LONG-TERM CHEMICAL BIOBARRIERS FOR U-TAILINGS. Cline, J.F.; Cataldo, D.A.; Burton, F.G.; Skiens, W.E. Feb 1981. NTIS, PC A09/MF A01.

Pacific Northwest Laboratory annual report for 1980 to the DOE Assistant Secretary for Environment. Part 2. Ecological sciences.

The objective of this project is to develop and evaluate the effectiveness of physical and chemical barriers that are designed to prevent plant and animal breachment of uranium tailings containment systems for extended periods of time. The development of a polymeric carrier/delivery system and the construction of rock asphalt barriers to prevent plant or animal intrusion are discussed.

102 (PNL--3700(Pt.2), pp 25-26) REVEGETATION OF INACTIVE U-TAILING SITES. Cadwell, L.L.; Hinds, N.R.; McShane, M.C.; Sauer, R.H.; Skalski, J.R. Feb 1981. NTIS, PC A09/MF A01.

Pacific Northwest Laboratory annual report for 1980 to the DOE Assistant Secretary for Environment. Part 2. Ecological sciences.

Soil placed over any sealant/barrier system can provide a protective mantle if the soil is not lost by erosion. Vegetation is an attractive choice for controlling erosion because it can provide an economical self-renewing cover that serves to reduce erosion by both wind and water. The objective of this research and development effort is to select and test vegetation strategies, including the choice of species and methods for revegetation that are compatible with sealant/barrier systems and are suited to soils and climates at inactive uranium mill tailings sites.

103 (PNL--3774) INFX GUIDE: SUMMARY OF US DOE PLANS AND POLICIES FOR INTERNATIONAL COOPERATION IN THE FIELD OF RADIOACTIVE WASTE MANAGEMENT. Harmon, K.M.; Kelman, J.A. (comps.). (Battelle Pacific Northwest Labs., Richland, WA (USA)). 1981. Contract AC06-76RL01830. 151p. NTIS, PC A08/MF A01. Order Number DE81028208.

The purpose of this document is to provide, in one source, an overview and summary of major international cooperative activities such as long-term personnel exchanges, planning of complementary R and D programs, and testing programs like the one at Stripa for use by DOE and DOE contractor personnel responsible for planning such programs. The contents are as follows: waste management-general, high-level waste immobilization; transuranic wastes; low-level radioactive waste; airborne wastes; waste isolation in geologic repositories; marine disposal; spent fuel storage; transportation; uranium mill tailings; decontamination and decommissioning; and appendices which are for bilaterial waste management agreements; INFX policies and procedures, DOE contractor personnel and international agencies.

104 (PNL--3797) PUBLIC CONCERNS AND CHOICES REGARDING NUCLEAR-WASTE REPOSITORIES. Rankin, W.L.; Nealey, S.M. (Battelle Pacific Northwest Labs., Richland, WA (USA); Battelle Human Affairs Research Center, Seattle, WA (USA)). Jun 1981. Contract AC06-76RL01830. 43p. (BHARC--411/81/003). NTIS, PC A03/MF A01. Order Number DE81027835.

Survey research on nuclear power issues conducted in the late 1970's has determined that nuclear waste management is now considered to be one of the most important nuclear power issues both by the US public and by key leadership groups. The purpose of this research was to determine the importance placed on specific issues associated with high-level waste disposal. In addition, policy option choices were asked regarding the siting of both low-level and high-level nuclear waste repositories. A purposive sampling strategy was used to select six groups of respondents. Averaged across the six respondent groups, the leakage of liquid wastes from storage tanks was seen as the most important high-level waste issue. There was also general agreement that the issue regarding water entering the final repository and carrying radioactive wastes away was second in importance. Overall, the third most important issue was the corrosion of the metal containers used in the high-level waste repository. There was general agreement among groups that the fourth most important issue was reducing safety to cut costs. The fifth most important issue was radioactive waste transportation accidents. Overall, the issues ranked sixth and seventh were, respectively, workers' safety and earthquakes damaging the repository and releasing radioactivity. The eighth most important issue, overall, was regarding explosions in the repository from too much radioactivity, which is something that is not possible. There was general agreement across all six respondent groups that the two least important issues involved people accidentally digging into the site and the issue that the repository might cost too much and would therefore raise electricity bills. These data indicate that the concerns of nuclear waste technologists and other public groups do not always overlap.

105 (PNL--3803) SELECTION OF CONTAINMENT SYSTEMS FOR COMMERCIAL HIGH-LEVEL RADIOACTIVE WASTE MANAGEMENT. Kaplan, M.F.; Giuffre, M.S.; Bartlett, J.W. (Battelle Memorial Inst., Columbus, OH (USA)). May 1981. Contract AC06-76RL01830. 49p. NTIS, PC A03/MF A01.

This document reports the results of a study aimed at determining the best strategy for providing containment during management of commercial high-level radioactive wastes. Containment to assure public and worker safety is needed for all storage, transport, handling, and disposal operations. There are several thousand containment system options; this work determined, in overview rather than detail, which options should be pursued. This work shows that the geologic and engineered barriers in repositories in different geologic media, such as salt and granite, play very different roles in preserving long-term containment. In sum, there is no common engineered waste package that is suitable for disposal in all geologic media, each package must be tailored to the specific repository system. The need to make the waste package specific to the repository system leads to the key elements of waste management containment strategy: perform final packaging at the disposal site, and deliver to the site a waste that is in a form suitable for disposal and in a container that is (a) appropriate for the process that produced the waste form, (b) satisfactory for transport, and (c) suitable as the common basis for custom tailoring the waste package for any repository system. As described in this report, mild carbon steel is a container material that can be expected to meet these requirements.

106 (PNL--3881) SUMMARY OF NON-US NATIONAL AND INTERNATIONAL RADIOACTIVE WASTE MANAGEMENT PROGRAMS 1981. Harmon, K.M.; Kelman, J.A. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jun 1981. Contract AC06-76RL01830. 78p. NTIS, PC A05/MF A01. Order Number DE81026978.

Many nations and international agencies are working to develop improved technology and industrial capability for nuclear fuel cycle and waste management operations. The effort in some countries is limited to research in university laboratories on treating low-level waste from reactor plant operations. In other countries, national nuclear research institutes are engaged in major programs in all phases of the fuel cycle and waste management, and there is a national effort to commercialize fuel cycle operations. Since late 1976, staff members of Pacific Northwest Laboratory have been working under US Department of Energy sponsorship to assemble and consolidate openly available information on foreign and international nuclear waste management programs and technology. This report summarizes the information collected on the status of fuel cycle and waste management programs in selected countries making major efforts in these fields as of the end of May 1981.

107 (PNL--4015) HYDRAULIC AND THERMAL PROPERTIES OF SOIL SAMPLES FROM THE BURIED WASTE TEST FACILITY. Cass, A.; Campbell, G.S.; Jones, T.L. (Pacific Northwest Lab., Richland, WA (USA)). Oct 1981. Contract AC06-76RL01830. 59p. NTIS, PC A04/MF A01. Order Number DE82002649.

In shallow land burial, the most common disposal method for low-level waste, waste containers are placed in shallow trenches and covered with natural sediment material. To design such a facility requires an in-depth understanding of the infiltration and evaporation processes taking place at the soil surface and the effect these processes have on the amount of water cycling through a burial zone. At the DOE Hanford Site in Richland, Washington, a field installation called the Buried Waste Test Facility (BWTF) has been constructed to study unsaturated soil water and contaminant transport. PNL is collecting data at the BWTF to help explain soil water movement at shallow depths, and specifically evaporation from bare sols. The data presented here represent the initial phase of a cooperative effort between PNL and Washington State University to use data collected at the BWFT.

108 (PNL--4050) SUBJECT BIBLIOGRAPHY OF RADIOACTIVE WASTE MANAGEMENT PUBLICATIONS AT PACIFIC NORTHWEST LABORATORY, 1975-1978. Powell, J.A. (comp.). (Pacific Northwest Lab., Richland, WA (USA)). Oct 1981. Contract AC06-76RL01830. 62p. NTIS, PC A04/MF A01. Order Number DE82002525.

This bibliography contains publications from 1975 to 1978 written by PNL staff. PNL translations are also announced in this document. The following areas are covered: actinides; airborne wastes; alternative waste forms; calcination; characterization; containers; decontamination; disposal; high-level wastes; liquid wastes; radionuclide migration; safety; separation processes; soils; solidification; storage; transport; transuranic waste; and vitrification.

109 (PNL-SA--6707) TECHNIQUES FOR SAMPLING NUCLEAR WASTE TANK CONTENTS AND IN SITU MEASUREMENT OF ACTIVITY. Lawrence, R.C. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Apr 1978. Contract EY-76-C-06-1830. 11p. (CONF-781004--1). Dep. NTIS, MF A01.

From Nuclear meeting; Basel, Switzerland (3 Oct 1978).

Portions of document are illegible.

A study was conducted to develop suitable sampling equipment and techniques for characterizing the mechanical properties of nuclear wastes; identifying effective means of measuring radiation levels, temperatures, and neutron fluxes in situ in wastes; and developing a waste core sampler. A portable, stainless steel probe was developed which is placed in the tank through a riser. This probe is built for the insertion of instrumentation that can measure the contents of the tank at any level and take temperature, radiation, and neutron activation readings with reliable accuracy. A simple and reliable instrument for the in situ extraction of waste materials ranging from liquid to concrete-like substances was also developed. This portable, stainless steel waste core sampler can remove up to one liter of radioactive waste from tanks for transportation to hot cell laboratories for analysis of hardness, chemical form, and isotopic content. A cask for transporting the waste samples from the tanks to the laboratory under radiation-protected conditions was also fabricated. This cask was designed with a ''boot'' or inner-seal liner to contain any radioactive wastes that might remain on the outside of the waste core sampling device.

110 (PNL-SA--6729) APPROACH TO RADIOLOGICAL ASSESSMENT FOR COMMERCIAL WASTE MANAGEMENT. Shipler, D.B.; Nelson, I.C. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 1978. Contract EY-76-C-06-1830. 9p. (CONF-780316--15). Dep. NTIS, PC A02/MF A01.

From Waste management fuel cycles; Tucson, AZ, USA (5 Mar 1978).

A radiological assessment was conducted in support of the Generic Environmental Impact Statement for Management of Commercially Generated Radioactive Waste. The assessment considered for individuals and populations the effects of airborne releases of radioactive materials from planned operations, decommissioning, and postulated accidents. Doses to work forces from direct radiation were estimated and health effects in populations were calculated. The preliminary drafts of the statement and the detailed environmental assessment are being reviewed by DOE; the results of this work will be presented after the documents are made public.

111 (PNL-SA--6839) DE MINIMIS LEVELS OF RADIOACTIVITY IN WASTE MANAGEMENT. Corley, J.P.; Schiager, K.J. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Feb 1978. Contract EY-76-C-06-1830. 10p. (CONF-780316--14). Dep. NTIS, PC A02/MF A01.

From Waste management fuel cycles; Tucson, AZ, USA (5 Mar 1978).

The authors define and discuss several approaches to establishing de minimis levels, with examples. In order to avoid creating still another constraint on waste facilities or controls, they suggest the term de minimis be applied only when based on environmental radioactivity, dose, or health effect. They further suggest that such usage implies case-by-case numerical evaluation for application to release rates or effluent concentrations.

112 (PNL-SA--7072) MANAGEMENT OF HIGH-LEVEL NUCLEAR WASTES. Platt, A.M.; McElroy, J.L. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jul 1978. Contract EY-76-C-06-1830. 28p. (CONF-780904--2). Dep. NTIS, PC A03/MF A01.

From 2. Pacific Basin conference; Tokyo, Japan (24 Sep 1978).

A brief review is given of significant developments in the management of high-level nuclear wastes since the Oct. 1976 first Pacific Basin Conference on Nuclear Power Development and the Fuel Cycle. Emphasis is on policy and technical developments in the U.S., with some attention paid to developments in other countries that have impacted technical direction in the U.S. Spent fuel and its packaging, vitrification, high-level waste glasses, and repositories are discussed. It is concluded that predisposal technology for processing high-level wastes is well developed and that geologic media can be used for disposal of nuclear wastes without significant risk. 19 figures. (DLC)

113 (PNL-SA--7072/T1) MANAGEMENT OF HIGH-LEVEL NUCLEAR WASTES. Platt, A.M.; McElroy, J.L. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Dec 1978. Contract AC06-76RL01830. 33p. NTIS, PC A03/MF A01.

Significant developments in the management of high-level nuclear wastes that have occurred in the last 2 years are reviewed. The principal thrust is on the policy and technical developments in the United States, but some attention is paid to major developments in other countries which have impacted technical direction in the US.

114 (PNL-SA--7540) CHARACTERIZATION OF NUCLEAR WASTE. Platt, A.M. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 14 Feb 1979. Contract EY-76-C-06-1830. 6p. (CONF-790304--9). Dep. NTIS, PC A02/MF A01.

From 19. annual ASME symposium, geological disposal of nuclear waste; Albuquerque, NM, USA (15 Mar 1979).

Nuclear wastes which are logical candidates for deep geologic disposal include commercial (spent fuel, reprocessing) and defense wastes. It is expected that the 5250 metric tons of spent fuel discharged through the end of 1978 would increase to about 100,000 tons by the end of 2000. The individual characteristics of each waste type (spent fuel, solidified waste, defense wastes) are described in turn. (DLC)

115 (PNL-SA--8829) APPLICATION OF ASPHALT EMULSION SEALS TO URANIUM MILL TAILINGS. Hartley, J.N.; Koehmstedt, P.L.; Esterl, D.J.; Freeman, H.D.; Clark, R.L. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Nov 1980. Contract AC06-76RL01830. 19p. (CONF-801155--4). NTIS, PC A02/MF A01.

From 3. annual symposium on uranium mill tailings management; Ft Collins, CO, USA (24 Nov 1980).

Studies of asphalt emulsion sealants have demonstrated that the sealants are effective in containing radon and other potentially hazardous material within uranium tailings. The laboratory and field studies have further demonstrated that radon exhalation from uranium tailings piles can be reduced by greater than 99% to less than background levels. Field tests at the tailings pile in Grand Junction, Colorado confirmed that an 6-cm admix seal containing 22 wt % asphalt could be effectively applied with a cold-mix paver. Other techniques were successfully tested, including a soil stabilizer and a hot, rubberized asphalt seal that was applied with a distributor truck. After the seals were applied and compacted, overburden was applied over the seal to protect the seal from ultraviolet degradation. 14 figures.

116 (PNL-SA--8872) APPLICATION OF CONTROLLED RELEASE TECHNOLOGY TO URANIUM MILL TAILINGS STABILIZATION. Burton, F.G.; Cataldo, D.A.; Cline, J.F.; Skiens, W.E. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 1981. Contract AC06-76RL01830. 18p. (CONF-810217--8). NTIS, PC A02/MF A01.

From ANS waste management conference; Tucson, AZ, USA (23 Feb 1981).

A trifluralin (herbicide) releasing device was developed with a theoretical effective lifetime in excess of 100 years. When placed in a layer in soil, the PCD system will prevent root penetration through that layer without harming the overlying vegetation. Equilibrium concentrations of trifluralin in soil can be adjusted (along with the theoretical life of the device) to suit specific needs. The present system was designed specifically to protect the asphalt layer or clay/aggregate barriers on uranium mill tailings piles; PCD devices composed of pellets could also be implanted over burial sites for radioactive and/or toxic materials, preventing translocation of those materials to plant shoots, and thence into the biosphere.

117 (PNL-SA--9231) COMMERCIAL US NUCLEAR REACTORS AND WASTE: THE STATUS, DECEMBER 1980. Platt, A.M.; Robinson, J.V. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jan 1981. Contract AC06-76RL01830. 12p. (CONF-810217--16). NTIS, PC A02/MF A01.

From ANS waste management conference; Tucson, AZ, USA (23 Feb 1981).

Between March 1 and December 15, 1980, the declared size of the commercial light water reactor (LWR) nuclear power industry in the US decreased another 9 GW(e). For the presently declared size: the 165 declared reactors will peak at a capacity of 153 GWe in 2001 and will consume about 820,000 MTU as enrichment feed; the theoretical rate of enrichment requirements will peak at about 19,000,000 SWUs/y in the year 2014; as few as two repositories each with capacity equivalent to 77,000 MTHM would hold the waste; and predisposal storage reactor basins and AFRs (away-from-reactor basins) would peak at < 65,000 MTU in the year 2016 if the two repositories were commissioned in the years 1997 and 2023.

118 (PNL-SA--9543) DEVELOPMENT OF ENGINEERED STRUCTURAL BARRIERS FOR NUCLEAR-WASTE PACKAGES. Westerman, R.E.; Elmore, R.P.; Pitman, S.G.; Nelson, J.L. (Pacific Northwest Lab., Richland, WA (USA)). Sep 1981. Contract AC06-76RL01830. 15p. (CONF-811122--47). NTIS, PC A02/MF A01. Order Number DE82005924.

From Annual meeting of the Materials Research Society; Boston, MA, USA (16 Nov 1981).

The development of structural barriers for nuclear waste packages involves selection of candidate materials, their screening by mechanical and corrosion testing, rigorous accelerated testing, and evaluation and comparison with other package elements. This document presents results from work conducted on titanium and ferrous alloys.

119 (PNL-SA--9676) DEVELOPMENT OF STANDARD TESTING METHODS FOR NUCLEAR-WASTE FORMS. Mendel, J.E.; Nelson, P.D. (Pacific Northwest Lab., Richland, WA (USA)). Nov 1981. Contract AC06-76RL01830. 26p. (CONF-811103--67). NTIS, PC A03/MF A01. Order Number DE82005920.

From ANS winter meeting; San Francisco, CA, USA (29 Nov 1981).

Standard test methods for waste package component development and design, safety analyses, and licensing are being developed for the Nuclear Waste Materials Handbook. This paper describes mainly the testing methods for obtaining waste form materials data. (DLC)

120 (PNL-SA--9823) GROUNDWATER LEACHING OF NEUTRALIZED AND UNTREATED ACID-LEACHED URANIUM-MILL TAILINGS. Gee, G.W. (Pacific Northwest Lab., Richland, WA); Begej, C.W.; Campbell, A.C.; Sauter, N.N.; Opitz, B.E.; Sherwood, D.R. (Pacific Northwest Lab., Richland, WA (USA)). 1981. Contract AC06-76RL01830. 16p. (CONF-811049--9). NTIS, PC A02/MF A01. Order Number DE82003858.

From 4. symposium on uranium mill tailings management; Fort Collins, CO, USA (26 Oct 1981).

Tailings neutralization was examined to determine the effect of neutralization on contaminant release. Column leaching of acid extracted uranium mill tailings from Exxon Highland Mill, Wyoming, Pathfinder Gas Hills Mill, Wyoming, and the Dawn Midnite Mill, Washington, resulted in the flushing of high concentrations of salts in the first four pore volumes of leachate, followed by a steady decrease to the original groundwater salt concentrations. Neutralization decreased the concentration of salts and radionuclides leaching from the tailings and decreased the volume of solution required to return the solution to the groundwater pH and EC. Radium-226 and uranium-238 leached quickly from the tailings in the initial pore volumes of both neutralized and unneutralized tailings, and then decreased significantly. 6 figures, 5 tables.

121 (PNL-SA--9824) CONTACT OF CLAY-LINER MATERIALS WITH ACIDIC TAILINGS. II. CHEMICAL MODELING. Peterson, S.R.; Krupka, K.M. (Pacific Northwest Lab., Richland, WA (USA)). Sep 1981. Contract AC06-76RL01830. 19p. (CONF-811049--12). NTIS, PC A02/MF A01. Order Number DE82005897.

From 4. symposium on uranium mill tailings management; Fort Collins, CO, USA (26 Oct 1981).

The ion speciation-solubility model WATEQ3 was used to model original aqueous solutions and solutions resulting from liner materials contacted with uranium mill tailings, synthetic mill tailings or H_2SO_4. The modeling results indicate solution species which are in apparent equilibrium with respect to particular solids. These solids provide potential solubility controls for their corresponding dissolved constituents. The disequilibrium indices computed by WATEQ3 indicate amorphic $Fe(OH)_3(A)$, $AlOHO_4$, alunite $[KAl_3(SO_4)_2(OH)_6]$, gypsum ($CaSO_4 \cdot 2H_2O$), celestite ($SrSO_4$), anglesite ($PbSO_4$) and $MnHPO_4$ may have precipitated in the contacted liner materials and may also provide solubility controls for their dissolved constituents. The disequilibrium indices also show that the solutions resulting from the interaction of Highland Mill tailings are oversaturated with K-, H-, and Na-jarosites $[(K,H,Na)Fe_3(SO_4)_2(OH)_6]$. Because jarosite has been identified by x-ray diffraction as a precipitate in these reacted liner materials, it would appear that there is a kinetic barrier which prohibits jarosite from being an effective solubility control. Results of this study also show that the solubilities of many solid phases were pH dependent. This exploratory use of geochemical modeling has demonstrated its capability to test solubility hypotheses for clay liners reacted with tailings solutions and to guide the analyses of important constituents and parameters for these solutions. Geochemical modeling can be used, in parallel with characterization techniques for the solid phases, to support the presence of the solid phase and to guide the search for further solid phases. Geochemical modeling is also an effective tool in delineating the chemical causes for changes in permeability of liner materials.

122 (PNL-SA--9898) APPROACH TO ACCELERATED TESTING. Westerman, R.E. (Pacific Northwest Lab., Richland, WA (USA)). Sep 1981. Contract AC06-76RL01830. 25p. (CONF-811119--7). NTIS, PC A02/MF A01. Order Number DE82005234.

From NWTS program information meeting; Columbus, OH, USA (17 Nov 1981).

This report reviews the materials degradation modes that may influence the longevity of spent fuel and high-level waste packages. An approach to testing is recommended that will lead to qualification of waste package materials in specific repository environments in times that are short relative to the time period over which the package and repository are expected to remain effective. (DLC)

123 (PNL-TR--355) ENVIRONMENTAL IMPACT OF WASTE MANAGEMENT AND TRANSPORT OF RADIOACTIVE MATERIALS ORIGINATING FROM THE NUCLEAR FUEL CYCLE. Sousselier, Y. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 15 Mar 1979. Contract W-7405-ENG-48. 20p. Dep. NTIS, PC A02/MF A01.

Transport of radioactive materials originating from operation of the nuclear fuel cycle represents only a limited number of transports at present. Such transports have already been subject to precise, complete, and strict rules for many years. The radioactive impact which transports of radioactive materials have will remain very limited. The impact of accidents will also be very limited. The detailed study of the various stages of the management of radioactive wastes as it is presently performed and of the techniques which are being developed and will be implemented as nuclear power continues to be developed has clearly demonstrated that the potential environmental impacts are presently, and will remain in the future, small if not negligible.

124 (SAND--80-2416C(Draft)) DEVELOPMENT OF REFERENCE CONDITIONS FOR GEOLOGIC REPOSITORIES FOR NUCLEAR WASTE IN THE USA. Raines, G.E.; Rickertsen, L.D.; Claiborne, H.C.; McElroy, J.L.; Lynch, R.W. (Battelle Memorial Inst., Columbus, OH (USA). Office of Nuclear Waste Isolation; Science Applications, Inc., Oak Ridge, TN (USA); Oak Ridge National Lab., TN (USA); Battelle Pacific Northwest Labs., Richland, WA (USA); Sandia National Labs., Albuquerque, NM (USA)). 1 Oct 1980. Contract AC04-76DP00789. 27p. (CONF-801124--10(Draft)). NTIS, PC A03/MF A01.

From 3. annual meeting of the Materials

Research Society; Boston, MA, USA (17 Nov 1980).

Portions of document are illegible.

Activities to determine interim reference conditions for temperatures, pressure, fluid, chemical, and radiation environments that are expected to exist in commercial and defense high-level nuclear waste and spent fuel repositories in salt, basalt, tuff, granite, and shale are summarized. These interim conditions are being generated by the Reference Repository Conditions Interface Working Groups (RRC-IWG), an ad hoc IWG established by the National Waste Terminal Storage Program's (NWTS) Isolation Interface Control Board (I-ICB).

125 STATISTICAL ASPECTS OF RADIOLOGICAL SITE ASSESSMENTS. Gilbert, R.O. (Battelle Northwest Labs., Richland, WA). Transactions of the American Nuclear Society ; 34: 105-106(1980). (CONF-800607--).

From American Nuclear Society annual meeting; Las Vegas, NV, USA (8 Jun 1980).

STATISTICS;ABANDONED SITES;NUCLEAR FACILITIES;DECOMMISSIONING;DECONTAMINATION; RADIOMETRIC SURVEYS;PLANNING;DATA ACQUISITION

126 NUCLEAR WASTE MATERIALS CHARACTERIZATION CENTER. Nelson, R.D.; Ross, W.A. (Battelle Northwest Labs., Richland, WA). Transactions of the American Nuclear Society ; 34: 192(1980). (CONF-800607--).

From American Nuclear Society annual meeting; Las Vegas, NV, USA (8 Jun 1980).

RADIOACTIVE WASTE MANAGEMENT;HIGH-LEVEL RADIOACTIVE WASTES;US DOE;ORGANIZING; RADIOACTIVE WASTE DISPOSAL;INFORMATION CENTERS; DATA BASE MANAGEMENT

127 COMPARATIVE TECHNIQUES FOR NUCLEAR FUEL CYCLE WASTE MANAGEMENT SYSTEMS. Pelto, P.J.; Voss, J.W. (Battelle Pacific Northwest Labs., Richland, WA). Transactions of the American Nuclear Society ; 34: 402-403(1980). (CONF-800607--).

From American Nuclear Society annual meeting; Las Vegas, NV, USA (8 Jun 1980).

Once-through and coprocessed UO_2-PuO_2 fuel cycles. RADIOACTIVE WASTE MANAGEMENT;FUEL CYCLE; SAFETY;COMPARATIVE EVALUATIONS;URANIUM DIOXIDE; PLUTONIUM DIOXIDE

128 ^{85}KR MANAGEMENT TRADE-OFFS: RADIOLOGICAL CONSEQUENCES. Mellinger, P.J.; Hoeres, G.R.; Brackenbush, L.W.; Greenborg, J. (Battelle Pacific Northwest Labs., Richland, WA). Transactions of the American Nuclear Society ; 34: 407-408(1980). (CONF-800607--).

From American Nuclear Society annual meeting; Las Vegas, NV, USA (8 Jun 1980).

RADIOACTIVE WASTE MANAGEMENT;KRYPTON 85;RISK ASSESSMENT;HUMAN POPULATIONS;RADIATION DOSES; GASEOUS WASTES;RADIOLOGICAL PERSONNEL;COST BENEFIT ANALYSIS

129 (BNWL-tr--332) ALTERNATIVE METHODS FOR ^{85}KR ULTIMATE STORAGE. Penzhorn, R.D. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Radiochemie). Jul 1978. Contract EY-76-C-06-1830. Translation of KFK--2482, December 1977 9p. Dep. NTIS, PC A02/MF A01.

Storage by ion implantation is described. Ion implantation (II) involves the penetration and retention of ions accelerated in the keV-MeV energy range in the surface layer of a solid material. Through collisions, the accelerated ions that are to be implanted give off electrical energy to the target material and heat it up. The number of ions implanted, resulting from the entire ion current, depends not on the physical properties of the implantation material but rather is determined by the external system. The gas enclosed in the matrix is in the form of small bubbles. The size of the bubbles depends on the temperature and is in the range of a few hundred Angstrom units. The bubbles are stable at least up to the temperature at which they were produced. Therefore, bombardment of the metal at higher temperatures decreases the danger of the gas being released, even at very high temperatures. Methods of producing noble gas ions are discussed along with embedding capacity, safety, recovery, and advantages and disadvantages of the method. (JRD)

130 (BNWL-tr--314) FRENCH PROGRAM AND ACHIEVEMENTS CONCERNING α WASTE. Sousselier, Y. (CEA Centre d'Etudes Nucleaires de Fontenay-aux-Roses, 92 (France)). May 1978. Contract EY-76-C-06-1830. Translated from pp 29--33 of paper presented at NEA-IAEA technical seminar on the treatment, conditioning, and storage of solid alpha-bearing waste and cladding hulls, Paris, France, December 5--7, 1977 (CONF-771208--5). 9p. Dep. NTIS, PC A02/MF A01.

From Technical seminar on the treatment, conditioning and storage of solid alpha-bearing waste and cladding hulls; Paris, France (5 Dec 1977).

Processes of encasement in asphalt or thermosetting resins continue to be developed in France for treating certain α wastes and particularly for sludge from chemical treatment by coprecipitation, evaporation sludge and used resins. French studies on long-term storage of radioactive wastes concern α waste in particular. Particularly in view of the volume involved, it is believed that the question of permanent storage of α waste must be settled sooner than the question of storing glass blocks. Three methods are being examined: storage in granitic rocks, storage in salt formations, and storage in suboceanic formations. Storage of α waste in geologic formations poses practically no problem of heat release and relatively few problems of irradiation protection during maintenance operations, but it does raise the question, to the greatest extent, of guaranteed long-term confinement. It is believed that storage in suboceanic formations may be particularly advantageous for this type of waste. With regard to granitic rocks, studies conducted have made it possible to determine a certain number of favorable mountain masses and it is proposed that the first depth drillings (1000 to 1500 meters) be made in 1978. The matter of salt formations is somewhat special, since the main problem is the long-term risk of intrusion into the storage area as a result of human activity; for example, possibly as a result of future development of the salt. This risk makes analysis very difficult. Considering the fact that in current practice the greatest amount of plutonium is found in low- and intermediate-level waste, maximum attention must be given to this type of waste, to determining the plutonium content, to recovery of the plutonium contained therein and to reduction of waste volume.

131 USE OF SAFETY INDICES IN WASTE MANAGEMENT SAFETY ASSESSMENT. Voss, J.W. (Battelle Northwest Labs., Richland, WA). Transactions of the American Nuclear Society ; 30: 286-288(1978). (CONF-781109--).

From 1978 winter meeting of American Nuclear

Society; Washington, DC, USA (12 Nov 1978).
RADIOACTIVE WASTE MANAGEMENT;SAFETY;HAZARDS

132 SECURE STORAGE OF RADIOACTIVE WASTE. Dau, G. (Battelle Pacific Northwest Labs., Richland, WA); Williams, R. Combustion; 48: No. 12, 24-32(Jun 1977).

The authors have reviewed available technology and have selected what appears to be the best of several alternatives at each step in the waste-handling process, as a technique for presenting the current status of the technology to the public and the industry. The process consists of the following sequence: fuel reprocessing; high-level liquid waste storage; two-step solidification; transportation and final disposal; and decontamination of fuel cladding and hardware residue. (auth)

133 (BNWL-tr--264) MANAGEMENT OF RADIOACTIVE WASTE IN BELGIUM. Dejonghe, P. (International Atomic Energy Agency, Vienna (Austria)). 1977. Contract EY-76-C-06-1830. Translation of Belgian report (CONF-770505--215(trans.)). 23p. Dep. NTIS, PC A02/MF A01.

From International conference on nuclear power and its fuel cycles; Salzburg, Austria (2 May 1977).

BelgoWaste Research Syndicate was formed in 1975 to carry out a technico-economic and legal study of radioactive waste in Belgium and to prepare the constitution of an organization charged with managing these wastes. This paper summarizes the results of the first phase of the study, which was devoted to an inventory of the radioactive waste for different development scenarios for the nuclear industry in Belgium. A project was established to detail the treatment and conditioning processes and the capacities of the projected installations.

134 (BNWL-tr--288) SPANISH NATIONAL RADIOACTIVE WASTE MANAGEMENT PROGRAM. Lopez Perez, B.; Ramos Salvador, L.; Martinez Martinez, A. (Junta de Energia Nuclear, Madrid (Spain)). 1977. Contract EY-76-C-06-1830. Translation of IAEA/CN--36/206 (CONF-770505--230(trans)). 16p. Dep. NTIS, PC A02/MF A01.

From International conference on nuclear power and its fuel cycles; Salzburg, Austria (2 May 1977).

The waste management for the nuclear program (nuclear power plants, fuel cycle, nuclear research centers, and radioisotope users) in Spain is discussed. Legislation affecting the above is discussed. Disposal and storage of the solid radioactive residues are considered. (DLC)

135 CHEMICAL DECONTAMINATION AND MELT DENSIFICATION OF CHOP-LEACH FUEL HULLS. Dillon, R.L.; Griggs, B.; Kemper, R.S.; Nelson, R.G. (Battelle Pacific Northwest Labs., Richland, Wash. (USA)). pp 185-197 of Management of radioactive wastes from the nuclear fuel cycle. Vol. 1. Vienna; IAEA (1976).

From IAEA symposium on the management of radioactive waste; Vienna, Austria (22 Mar 1976).

See CONF-760310--; STI/PUB--433(Vol.1).

This paper reports on decontamination and densification studies of chop-leach fuel hull residues designed to minimize the transuranic element (TRU) contaminated waste stream. Decontamination requirements have been established from studies of TRU element distribution in the fuel hull residues. Effective surface decontamination of Zircaloy requires removal of zirconium oxide corrosion products. Good decontamination factors have been achieved with aqueous solutions following high temperature HF conditioning of oxide films. Molten fluoride salt mixtures are effective decontaminants, but pose problems in metal loss and salt dragout. Molten metal decontamination methods are highly preliminary, but may be required to reduce TRU originating from tramp uranium in Zircaloy. Low melting (1300°C) alloy of Zircaloy, stainless steel, and Inconel have been prepared in induction heated graphite crucibles. High quality ingots of Zircaloy-2 have been prepared directly from short sections of descaled fuel clad tubing using the Inductoslag process. This material is readily capable of refabrication. Inductoslag melts have also been prepared from heavily oxidized Zircaloy tubing demonstrating melt densification without prior decontamination is technically feasible. Hydrogen absorption kinetics have been demonstrated with cast Zircaloy-2 and cast Zircaloy-stainless steel-Inconel alloys. Metallic fuel hull residues have been proposed as a storage medium for tritium released from fuel during reprocessing.

Waste Processing

136 (BNL--50854) SOLIDIFICATION OF SIMULATED TRANSURANIC CONTAMINATED INCINERATOR ASH WASTES USING PORTLAND TYPE I CEMENT. Neilson, R.M. Jr.; Colombo, P. (Brookhaven National Lab., Upton, NY (USA); Battelle Pacific Northwest Labs., Richland, WA (USA)). May 1978. Contract EY-76-C-02-0016. 24p. Dep. NTIS, PC A02/MF A01.

A preliminary study was performed to investigate the use of hydraulic cement for the solidification of transuranic (TRU) contaminated incinerator ash wastes. In this work, the compositional phase fields in which acceptable solidification is obtained using portland type I cement were determined for three simulated TRU incinerator ash wastes. These wastes include Battelle Northwest TRU incinerator ash, Rocky Flats plutonium recycle TRU incinerator ash, and Mound Laboratory cyclone incinerator TRU ash 11A2. The compressive strengths and set times of selected formulations were measured.

137 (BNWL--80) SOLIDIFICATION OF HIGH LEVEL WASTES. PART IV. PHOSPHATE MELTS FOR FIXATION OF RADIOACTIVE RESIDUES FROM PUREX TYPE WASTES: THREE TO FIFTY PERCENT FISSION PRODUCT OXIDES. Barton, G.B. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jul 1965. 39p. NTIS.

The possible composition range for the high-level radioactive wastes 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% of the weight of oxides in the plant waste can be successfully incorporated in a phosphate melt that can be transferred as a fluid into containers at less than 900°C.

138 (BNWL--180) ADSORPTION OF TRACE IONS FROM INTERMEDIATE-LEVEL RADIOACTIVE WASTES BY ION EXCHANGE. Mercer, B.W. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jan 1966. 33p. NTIS.

Trace cesium and strontium selectivity coefficients were determined for 16 cation exchangers. Mass action equations were derived for determining the cesium and strontium equilibrium distribution coefficients for solutions containing several competing cations. Good agreement was obtained between computed and experimental distribution coefficients. The computed 1% cesium breakthrough points were within 20% of the actual 1% cesium breakthrough points determined previously for two waste treatment pilot plant ion exchange columns with a plant condensate waste. The slopes of the computed breakthrough curves agreed well with the actual curves for the range covered. A plant condensate waste was steam stripped, filtered, and decontaminated by ion exchange on a pilot plant scale. The inadvertent introduction of bacteria into the plant condensate reretention tank complicated the decontamination of this condensate by increasing the filtration requirement. The steam stripped condensate waste containing bacterial residue was successfully filtered with a horizontal plate filter by using diatomaceous earth filter aid. The use of a fine particle zeolite as both a filtration and an ion exchange media for the horizontal plate filter was apparently unsatisfactory because only a fraction of the zeolite ion exchange capacity was used before the bed became plugged.

139 (BNWL--200) FIXATION OF HIGH-LEVEL RADIOACTIVE WASTES IN PHOSPHATE GLASS: HOT CELL GLASS EXPERIMENT. Upson, U.L. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jan 1966. 73p. (CONF-660208--9). NTIS.

From Symposium on solidification and long-term storage of highly radioactive wastes; Richland, WA, USA (14 Feb 1966).

Phosphate glass was produced from high-level fuel-reprocessing waste, demonstrating the technical feasibility of the continuous phosphate glass process. In this process, an aqueous waste is converted directly to a phosphate glass, with no intermediate drying or calcination step. Leachability of the product glass was as good as that of similar cold glasses, and has remained substantially constant for glass stored in dry air. Storage in moist air yielded 5- to 10-fold increase in leachability in the same period. An inherent instability of the glassy state, found especially at temperatures near the melting point, suggests that alternative polycrystalline forms also should be investigated.

140 (BNWL--1968) SELECTION OF A MELTING FURNACE FOR CONSOLIDATION OF NUCLEAR FUEL HULLS. Nelson, R.G.; Schlienger, M.P.; Tiesenhausen, E.V. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Dec 1976. Contract EY-76-C-06-1830. 52p. Dep. NTIS $4.50.

The selection and design criteria for a cold-crucible melting system for fuel hull consolidation are defined. Appraisals of the cold-crucible processes that are available are presented. (LK)

141 (BNWL--2063) SELECTION AND EVALUATION OF PROCESSES FOR RECOVERY OF BENEFICIAL ISOTOPES FROM COMMERICAL REACTOR WASTES. Davis, D.K.; Partridge, J.A.; Koski, O.H. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Dec 1977. Contract EY-76-C-06-1830. 32p. Dep. NTIS, PC A03/MF A01.

Conceptual processes were prepared for recovering isotopes from stored high-level liquid wastes to define equipment and facility requirements. Capital costs were estimated for each process. Four different facilities were studied for potential isotope recovery. The processes were for the recovery of: cesium only; strontium only; cesium and strontium; and cesium, strontium, promethium, americium, curium, neptunium, and the platinum metals. The selection of these recovery processes was determined by their possible effect on the waste management system. The isotope recovery processes selected tended to have a minimal effect on this system. Preliminary capital investment costs for the four types of facilities processing HLLW from a 5 MT/day reprocessing plant were estimated to be: $37 million for a cesium or a strontium recovery facility; $51 million for a facility recovering cesium and strontium; and $143 million for a multi-product facility. In all cases the recovery facility was assumed to be closely integrated with a fuel reprocessing plant and designed along with the reprocessing plant. Recovery costs were estimated for the various products, assuming 50 percent recovery from the HLLW for all of the products, except the platinum metals; 25 percent recovery was assumed for these metals. In general, product price estimates (in 1977 dollars) assumed a market for all of the recovered materials. Estimates for cesium ranged from 45 cents/Ci of ^{137}Cs (at 69 MCi/yr) in a cesium-only recovery facility to 31 cents/Ci in a plant recovering both cesium and strontium. Strontium estimates ranged from 60 cents/Ci of ^{90}Sr (at 51 MCi/yr) in a strontium-only recovery facility to 42 cents/Ci in a cesium and strontium recovery facility.

142 (BNWL--2117, pp 11-38) DISPOSAL FORMS FOR THE ACTINIDE WASTES FROM THE NUCLEAR FUEL CYCLE. Mendel, J.E. 1976.

From Conference on actinide sediment reaction; Seattle, WA, USA (10 Feb 1976).

The paper concerns the actinides produced in the generation of nuclear power, which are handled mostly by private industry. The distribution of actinides in the fuel cycle is described. Candidate solid forms for long-term immobilization are discussed. Factors potentially affecting long-term stability of HLW glasses include devitrification, radiation doses, metamictization, and leaching. Acceleration of alpha doses and extrapolation of leach rates to long times are studied. 6 figures, 6 tables, 10 references. (DLC)

143 (BNWL--2274) EVALUATION OF A PRECIPITATION-ION EXCHANGE PROCESS FOR TREATMENT OF LAUNDRY WASTE. Mercer, B.W.; Ames, L.L. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 15 Mar 1977. Contract EY-76-C-06-1830;EY-76-C-06-2130. 60p. Dep. NTIS, PC A04/MF A01.

Bench-scale pilot plant studies were conducted to evaluate chemical coagulation and ion exchange for decontamination of 2724-W laundry wastewater. Chemical coagulation is accomplished at pH 11 to avoid complexant problems and assure good transuranic radionuclide removals. Clinoptiloite is used to remove cesium and strontium. Results of the pilot plant studies are summarized as follows: Decontamination factors of 70 (strontium) and more than 100 (cesium) were achieved by chemical coagulation and ion exchange. Decontamination factors exceeding 90 were measured for europium by coagulation with a combination of ferric chloride, magnesium chloride, and calcium chloride added to the wastewater at pH 11. Coagulation with these three agents in the wastewater at pH 11 was more effective for turbidity removal than coagulation with lime. Addition of up to 1.7 lb of clinoptilolite fines per 1000 gallons of wastewater during coagulation did not substantially increase strontium and cesium removal. Filtration without chemical coagulation reduced suspended solids by only 25%. About 70% of the suspended solids remaining in the filtered wastewater were removed in the zeolite column causing plugging which could not be easily dislodged by backwashing. Plugging of the ion exchange columns by previously clarified wastewater required short periods of limited backwashing to relieve the plug. The plugging is due to $CaCO_3$ and is not expected to be a severe problem in a full-scale plant with brief detention times between filtration and ion exchange. A high pressure surface wash should be included in the columns to break up crust or plugs at the surface of the zeolite. Centrifugation of iron sludges for 2 min at 2000 g reduced the sludge volume to about 1% of the total wastewater volume. Wet iron sludges from the sludge storage tank were readily dewatered by vacuum filtration. 14 tables, 9 figures.

144 (BNWL-CC--2028(Suppl.1)) 324 BUILDING SAFETY ANALYSIS REPORT SUPPLEMENT. Dodd, A.O.; Wittenbrock, N.G. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 24 Jun 1977. Contract EY-76-C-06-1830. 63p. Dep. NTIS, PC A04/MF A01.

Process engineering designs, major equipment and plant facilities to be utilized in commercial nuclear waste preparation and vitrification in the 324 Radiochemical Engineering Building are reviewed with regard to accident potential and consequences. This Safety Analysis Report Supplement compares calculated environmental doses anticipated from the Commercial Nuclear Waste Vitrification Project (CNWVP) routine operations with the average doses from past waste management operations conducted at the Hanford Project and finds them to be significantly less. The calculated CNWVP environmental doses are found to be far below presently applicable ERDA standards and standards proposed by the EPA for nuclear power operations. (DLC)

145 (BNWL-SA--6146) PROPERTIES AND CHARACTERISTICS OF HIGH-LEVEL WASTE GLASS. Ross, W.A. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 1977. Contract EY-76-C-06-1830. 25p. (CONF-770159--1). Dep. NTIS, PC A02/MF A01.

From Properties and characteristics of high-level waste glass; Washington, DC, USA (4 Jan 1977).

This paper has briefly reviewed many of the characteristics and properties of high-level waste glasses. From this review, it can be noted that glass has many desirable properties for solidification of high-level wastes. The most important of these include: (1) its low leach rate; (2) the ability to tolerate large changes in waste composition; (3) the tolerance of anticipated storage temperatures; (4) its low surface area even after thermal shock or impact.

146 (BNWL-SA--6368) ELECTROPOLISHING AS A LARGE-SCALE DECONTAMINATION TECHNIQUE. Allen, R.P.; Arrowsmith, H.W.; Bucke, W.C. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 1977. Contract EY-76-C-06-1830. 33p. (CONF-771102--25). Dep. NTIS, PC A03/MF A01.

From 70. annual AICHE meeting; New York, NY, USA (13 Nov 1977).

Laboratory-scale studies have shown electropolishing to be a rapid and effective technique for removing plutonium and other radionuclide contamination from a variety of metal surfaces. This paper summarizes work in progress at Battelle, Pacific Northwest Laboratory, to develop electropolishing into a large-scale decontamination technique that can be used to minimize the amount of surface-contaminated metallic waste requiring geologic disposal. A 400-gal. electropolishing facility has been established to develop and demonstrate decontamination techniques for representative plutonium- and beta/gamma-contaminated nuclear industry materials and components. Initial tests using this facility have demonstrated the ability to decontaminate more than 15 sq. ft. of plutonium-contaminated stainless steel in less than 30 min. of electropolishing time. Supporting studies also are in progress to develop in situ electropolishing techniques for the decontamination of surfaces that cannot be transported to or immersed in an electropolishing cell and to develop solution treatment procedures to extend electrolyte life and minimize the amount of secondary waste generated by the decontamination process.

147 (BNWL-SA--6461) SEMIVOLATILE FISSION PRODUCT BEHAVIOR IN HIGH-LEVEL WASTE VITRIFICATION. Hanson, M.S.; McElroy, J.L. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Oct 1977. Contract EY-76-C-06-1830. 20p. (CONF-771139--3). Dep. NTIS, PC A02/MF A01.

From Radioactive effluents from nuclear fuel reprocessing plants seminar; Karlsruhe, F.R. Germany (22 Nov 1977).

The solidification of high-level wastes is being evaluated as a method for final control of high-level nuclear wastes. The high temperatures required in the solidification process cause the volatilization of small amounts of fission products, primarily ruthenium and cesium. In order to determine cleanup requirements, a good understanding must be gained as to the quantities of fission products volatilized and how these quantities relate to operating conditions. The Pacific Northwest Laboratory, operated by Battelle-Northwest for the U.S. Department of Energy, has been investigating ruthenium and cesium volatilization in various calciners. Studies have been made using both simulated (nonradioactive) and radioactive feedstock. The ongoing investigations have indicated that the semivolatile fission product losses are small enough to be controlled by off-gas cleanup treatment. During recent spray calcination of simulated high-level liquid waste, not more than 0.2 percent of the ruthenium and 0.1 percent of the cesium were lost to the off-gas system. Decontamination factors of about 600 for ruthenium and 1150 for cesium have been obtained across the spray calciner.

148 (BNWL-SA--6564) TECHNOLOGY STATUS OF SPRAY CALCINATION--VITRIFICATION OF HIGH-LEVEL LIQUID WASTE FOR FULL-SCALE APPLICATION. Keeley, R.B.; Bonner, W.F.; Larson, D.E. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 1977. Contract EY-76-C-06-1830. 27p. (CONF-771102--27). Dep. NTIS, PC A03/MF A01.

From 70. annual AICHE meeting; New York, NY,

USA (13 Nov 1977).
Spray calcination and vitrification technology for stabilization of high-level nuclear wastes has been developed to the point that initiation of technology transfer to an industrial-sized facility could begin. This report discusses current process and equipment development status together with additional R and D studies and engineering evaluations needed. Preliminary full-scale process and equipment descriptions are presented. Technology application in a full-scale plant would blend three distinct maintenance design philosophies, depending on service life anticipated: (1) totally remote maintenance with limited viewing and handling equipment, (2) totally remote maintenance with extensive viewing and handling equipment, and (3) contact maintenance.

149 (CONF-780304--, pp IX.13-IX.14) NUCLEAR WASTE VITRIFICATION. Chapman, C.C.; Buelt, J.J. (Battelle-Northwest Labs., Richland, WA). 19 Mar 1978.
From ANS: Back end of the LWR fuel cycle; Savannah, GA, USA (19 Mar 1978).
Two radioactive waste vitrification processes are presented. The first is a ceramic melter coupled to a calciner. Pilot-scale of a melter is described. The second process is the direct liquid-fed ceramic melter. Experimental results with a pilot-scale unit and a large-scale developmental unit are presented. 2 figures, 2 tables.

150 (CONF-790420--, pp 1-3) DOE MATERIALS PROGRAM SUPPORTING IMMOBILIZATION OF RADIOACTIVE WASTES. Oertel, G.K.; Scheib, W.S. Jr. (Dept. of Energy, Washington, DC). May 1979. Dep. NTIS, PC A17/MF A01.
From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).
A summary is presented of the DOE program for developing waste-form criteria, immobilization processes, and generation and evaluation of performance characterization data. Interrelationships are discussed among repository design, materials requirements, immobilization process definition, quality assurance, and risk analysis as part of the National Environmental Policy Act and regulatory processes.

151 (CONF-790420--, pp 17-24) UK PROGRAM: GLASSES AND CERAMICS FOR IMMOBILIZATION OF RADIOACTIVE WASTES FOR DISPOSAL. Johnson, K.D.B. (UKAEA, Harwell, England). May 1979. Dep. NTIS, PC A17/MF A01.
From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).
The UK Research Program 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 behavior under storage and disposal conditions have been examined. Methods for immobilizing activity from other wastes by conversion to glass or ceramic forms are described. The UK philosophy of final solutions to waste management and disposal is presented.

152 (CONF-790420--, pp 73-81) OPERATIONAL EXPERIENCE OF THE FIRST INDUSTRIAL HLW VITRIFICATION PLANT. Chotin, M.M. (Compagnie Generale des Matieres Nucleaires, Marcoule, France); Bonniaud, R.A.; Jouan, A.F.; Rabot, G.E. May 1979. Dep. NTIS, PC A17/MF A01.
From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).
Marcoule Vitrification Plant which started up in active operation in June 1978 converts high-level radioactive liquid wastes into glass. The French vitrification process is a continuous technique involving a calcination step. The glass is cast in stainless steel containers. After the lids are welded, a decontamination stage is operated prior to disposal.

153 (CONF-790420--, pp 82-85) COMPATIBILITY TESTS OF MATERIALS FOR A PROTOTYPE CERAMIC MELTER FOR DEFENSE GLASS-WASTE PRODUCTS. Wicks, G.G. (E.I. du Pont de Nemours and Co., Aiken, SC). May 1979. Dep. NTIS, PC A17/MF A01.
From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).
The corrosion--erosion resistance of potential electrode and refractory materials was evaluated by static and dynamic tests in simulated glass waste. Based on corrosion--erosion behavior, thermal and electrical properties, and cost and availability, Monofrax K3 (Carborundum Co.) and Inconel 690 (International Nickel Co.) were selected as the contact refractory and electrode materials, respectively, for a prototype ceramic melter.

154 (CONF-790420--, pp 86-92) VITRIFICATION OF HIGH-LEVEL RADIOACTIVE WASTE IN A CONTINUOUS LIQUID-FED CERAMIC MELTER. Weisenburger, S.; Gruenewald, W.; Koschorke, H. (Karlsruhe Nuclear Research Center, Germany). May 1979. Dep. NTIS, PC A17/MF A01.
From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).
The vitrification of simulated high-level liquid waste (HLLW) from the nuclear fuel cycle in a ceramic-lined waste melter is described. Particular reference is given to the present status of the melter technology. Some features are discussed arising from direct liquid feeding of the waste solution into the melter. Also presented are results concerning the entrainment of particulates as well as the loss of volatilized material into the melter off-gas. Ruthenium behavior in a liquid-fed ceramic melter is reported, and the extent of ruthenium volatility suppressed by predenitration of the waste solution is discussed. The 1500 h operation experience gained since 1976 with the pilot-scale vitrification unit is outlined, and the construction of an advanced melter having the potential for remote operation is given. The work described here is part of a German cooperative research and development program among seven institutions. One of the major objectives of this program is to provide a vitrification technology for the solidification of the LEWC waste stored at the Eurochemic site in Mol/Belgium. 3 figures, 4 tables.

155 (CONF-790420--, pp 93-96) VITRIFICATION OF HANFORD RADIOACTIVE DEFENSE WASTES. Kupfer, M.J. (Rockwell Hanford Operations, Richland, WA). May 1979. Dep. NTIS, PC A17/MF A01.
From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).
Hanford defense waste sludges and residual liquids are effectively immobilized by conversion to borosilicate glasses. The effects of composition on various properties such as

chemical durability, viscosity, and devitrification are discussed. 5 tables.

156 (CONF-790420--, pp 97-101) SOLIDIFICATION OF HLW SOLUTIONS WITH THE PAMELA PROCESS. Heimerl, W. (Deutsche Gesellschaft fuer Wiederaufarbeitung von Kernbrennstoffen mbH, Mol, Belgium). May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

A revised concept for the PAMELA process for the solidification of HLW solutions is described. The reasons for some changes, e.g., future use of borosilicate glass instead of phosphate glass, manufacture of glass blocks as well as blocks of glass beads in metal matrix, are discussed. Plans for a PAMELA demonstration plant to be built at Mol, Belgium, are presented.

157 (CONF-790420--, pp 102-106) DEVELOPMENT OF ENGINEERING SCALE HLLW VITRIFICATION TECHNIQUE AT PNC. Nagaki, H.; Oguino, N.; Tsunoda, N.; Segawa, T. (Power Reactor and Nuclear Fuel Development Corp., Ibaraki, Japan). May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

Some processes have been investigated to develop the technology of solidification of the high-level radioactive liquid waste generated from the nuclear fuel reprocessing plant operated by the Power Reactor and Nuclear Fuel Development Corporation (PNC) at Tokai-mura. This report covers the present state of development of a Joule-heated ceramic melter and a direct megahertz induction-heated melter. Engineering-scale tests have been performed with both melters. The Joule-heated melter could produce 45 kg or 16 liters of glass per hour. The direct-induction furnace was able to melt 5 kg or 1.8 liters of glass per hour. Both melters were composed of electrofused cast refractory brick. Thus it was possible to melt the glass at above 1200°C. Glass produced at higher melting temperatures is generally superior. 3 figures, 2 tables.

158 (CONF-790420--, pp 107-113) REVIEW OF CONTINUOUS CERAMIC-LINED MELTER AND ASSOCIATED EXPERIENCE AT PNL. Buelt, J.L.; Chapman, C.C.; Barnes, S.M.; Dierks, R.D. (Battelle Pacific Northwest Labs., Richland, WA). May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

Development of continuous, ceramic-lined melters applicable to immobilization of radioactive wastes began at PNL in 1973. A comprehensive program is curretly in progress. The melters constructed at PNL have incorporated remote and reliable design features necessary for radioactive use. The extensive experience with vitrification of simulated wastes has proven the continuous melter's applicability to radioactive waste immobilization.

159 (CONF-790420--, pp 114-117) PROGRESS IN FISSION PRODUCT SOLIDIFICATION AND CHARACTERIZATION UTILIZING THE JUELICH FIPS PROCESS. Halaszovich, S.; Merz, E.; Odoj, R. (Kernforschungsanlage, Juelich, Germany). May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

The FIPS process for solidification of fission product solutions is briefly presented. Its main characteristics concerning process and product control are discussed. A newly developed layout consisting of a thermobalance and a mass spectrometer is described. Possible effects of the results of the thermodynamic investigations on glass composition and processing are discussed.

160 (CONF-790420--, pp 118-121) LABORATORY-SCALE GLASS MELTER FOR TESTING DEFENSE WASTE GLASS. Gombert, D. II. (Allied Chemical Corp., Idaho Falls, ID). May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

A one-liter Joule-heated glass melter was built to test the applicability of continuous melting to simulated high-level calcined defense waste. Inconel 690 electrodes and K-3 refractory brick were chosen for their corrosion resistance to fluoride glass. The melter maintained a full melt at 1100°C using 3 kW. After approximately two months of operation the melter was dismantled for metallurgical examination. The Inconel 690 electrodes were heavily corroded. A second melter is now in operation to verify the findings of the first melter run.

161 (CONF-790420--, pp 122-126) INFLUENCE OF HEAT TRANSFER ON THE POT VITRIFICATION PROCESS. Morris, J.B. (Atomic Energy Research Establishment, Harwell, England). May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

The experimental data on the working zone in pot vitrification show that the mean wall heat flux (2 W/cm^2) is independent of vessel size. A simple model of the heat transfer processes is developed in which natural convection through the glass-forming region is the controlling mechanism. 3 figures.

162 (CONF-790420--, pp 127-131) RECENT DEVELOPMENTS IN LOW- AND INTERMEDIATE-LEVEL WASTE FIXATION BY CEMENT. Witte, H.D. (NUKEM GmbH, Hanau, Germany); Koester, R. May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

Boron-containing radwaste sludge from PWR's can be incorporated into cement by addition of slaked lime, because in the presence of sufficient calcium the formation of tri-calcium aluminate is not inhibited. Incorporation of sodium nitrate into cement at least up to 20 wt % can be achieved with product qualities within acceptable levels. 5 figures, 2 tables.

163 (CONF-790420--, pp 132-135) FUETAP (FORMED UNDER ELEVATED TEMPERATURES AND PRESSURES) CONCRETES AS HOSTS FOR RADIOACTIVE WASTES. Moore, J.G.; Rogers, G.C.; Paehler, J.H.; Devaney, H.E. (Oak Ridge National Lab., TN). May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

Cementitious solids formed at less than or equal to 250°C and less than or equal to 70.3 kg/cm^2 (1000 psi) offer excellent possibilities as hosts for radioactive wastes. Preliminary results are presented on the effect of mix composition, temperature, and pressure on the physical properties of the resulting solids. Initial leach results are encouraging in the assessment of the ability of these FUETAP concretes to isolate radionuclides from the environment. 7 tables.

164 (CONF-790420--, pp 136-142) LOW TEMPERATUE CERAMIC RADIOACTIVE WASTE FORM. Roy, D.M.; Scheetz, B.E.; Grutzeck, M.W.; Sarkar, A.K.; Atkinson, S.D. (Pennsylvania State Univ., University Park). May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

Preliminary research on low temperature ceramic waste forms based upon modified calcium silicate and aluminate cements is described. Compositional and reaction variations, modified to achieve a strongly consolidated waste form which is reistant to significant change under the anticipated ambient, have been investigated. Extending previous studies in this laboratory, the recent work has utilized combinations of calcium aluminate cements and high early strength portland cements, different supercalcine waste formulations, and other additives. Both modified conventional cement composite consolidation methods and low-temperature hot-pressing methods were used to incorporate the supercalcine into a consolidated form. Products after initial consolidation and after drastic accelerated leaching treatment (hydrothermal leaching) were characterized. Uranium and simulated transuranics were found to be highly resistant to leaching; strontium was relatively resistant in many cases; and cesium leaching was found to be compositionally dependent. 8 tables.

165 (CONF-790420--, pp 143-149) SOLIDIFICATION OF HLLW BY GLASS-CERAMIC PROCESS. Oguino N. (Power Reactor and Nuclear Fuel Development Corp., Ibaraki, Japan); Masuda, S.; Tsunoda, N.; Yamanaka, T.; Ninomiya, M.; Sakane, T.; Nakamura, S.; Kawamura, S. May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

The compositions of glass-ceramics for the solidification of HLLW were studied, and the glass-ceramics in the diopside system was concluded to be the most suitable. Compared with the properties of HLW borosilicate glasses, those of diopside glass-ceramic were thought to be almost equal in leach rate and superior in thermal stability and mechanical strength. It was also found that the glass in this system can be crystallized simply by pouring it into a thermally insulated canister and then allowing it to cool to room temperature. 2 figures, 5 tables.

166 (CONF-790420--, pp 150-154) COATING OF WASTE CONTAINING CERAMIC GRANULES. Neumann, W.; Kofler, O. (Research Centre Seibersdorf, Vienna, Austria). May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

Simulated high-level waste granules produced by fluidized-bed calcination were overcoated by chemical vapor deposition (CVD) with pyrocarbon and nickel in laboratory-scale experiments. Successful development enables pyrocrbon deposition at temperatures of 600 to 800°K. The coated granules have excellent properties for long-term waste storage.

167 (CONF-790420--, pp 155-159) SOLIDIFICATION OF HLLW INTO SINTERED CERAMICS. O-Oka, K. (Toshiba Corp., Kawasaki, Japan); Ohta, T.; Masuda, S.; Tsunoda, N. May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

Simulated HLLW from the PNC reprocessing plant at Tokai was solidified into sintered ceramics by normal sintering or hot-pressing with addition of some oxides. Among various ceramic products obtained so far, the most preferable was nepheline-type sintered solids formed with addition of SiO_2 and Al_2O_3 to the simulated waste calcine. The solid shows advantageous properties in leach rate and mechanical strength, which suggest that the ceramic solids were prepared with additions of ZrO_2 or MnO_2, and some of them showed good characteristics.

168 (CONF-790420--, pp 160-163) EMBEDDING METHODS OF SOLIDIFIED WASTE IN METAL MATRICES. Neumann, W. (Researh Center Seibersdorf, Vienna, Austria). May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

The embedding of simulated waste calcines by three different methods (vacuum-pressure casting, centrifugal casting, and metal stirred with the calcines) was investigated. The experimental performance is described and advantages and disadvantages noted. The feasibility of embedding fines by stirring in metal was shown. In addition, an estimation of the influence of porosity on the properties of composites was carried out.

169 (CONF-790420--, pp 164-168) DEVELOPMENT OF CERMETS FOR HIGH-LEVEL RADIOACTIVE WASTE FIXATION. Aaron, W.S.; Quinby, T.C.; Kobisk, E.H. (Battelle Pacific Northwest Labs., Richland, WA). May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

A method is currently under development for the solidification and fixation of commercial and defense high-level radioactive wastes in the form of ceramic particles encapsulated by metal, i.e., a cermet. The chemical and physical processing techniques which have been developed and the properties of the resulting cermet bodies are described in this paper. These cermets have the advantages of high thermal conductivity and low leach rates.

170 (CONF-790420--, pp 169-173) DENSIFICATION OF CALCINES AND DIRECT CONTAINMENT OF SPENT NUCLEAR FUEL IN CERAMICS BY HOT ISOSTATIC PRESSING. Larker, H.T. (High-Pressure Lab., Robertsfors, Sweden). May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

Calcined HLLW and many other types of radioactive waste can be incorporated in synthetic mineral blocks by hot isostatic pressing. The processing can be made relatively simple, and lasting contamination of the hot press can be effectively avoided. The technique can also be applied to make canisters of synthetic corundum for spent fuel and a development up to full scale has been made. Results from corrosion tests on the densified materials are given, and the problem of obtaining relevant long-term leech rates for crystalline materials with mechanically disturbed surface layers is discussed.

171 (CONF-790420--, pp 174-178) IMMOBILIZATION OF HIGH-LEVEL WASTES IN SYNROC TITANATE CERAMIC. Ringwood, A.E.; Kesson, S.E. (Australian National Univ., Canberra). May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

The elements occurring in high-level nuclear reactor wastes can be safely immobilized by incorporating them within the crystal lattices of the constituent phases of a titanate ceramic, SYNROC. Close structural analogs of these phases (hollandite, perovskite, and zirconolite) occur in nature and have survived for periods of 20 to 2000 million years. Accelerated leaching tests in water and saline solutions show that SYNROC is unaffected after 24 hours at extreme conditions (up to 900°C and 5000 bars), whereas borosilicate glasses decompose within a few hours under much less severe hydrothermal conditions (e.g., 350°C and 1000 bars). A variant of the SYNROC process appears to be particularly well-suited for immobilization of US military wastes. The modified ceramic waste form consists of the three major SYNROC phases (hollandite, perovskite , and zirconolite) plus a variety of refractory Fe--Al--Mn bearing titanates and spinels, which are thermodynamically compatible with the SYNROC phases. The SYNROC phases serve to immobilize fission products and actinides, while the additional refractory phases serve to immobilize the Fe--Al--Mn oxides, which constitute a very large proportion of the calcined sludges. These additional phases include pseudobrookite (AL_2TiO_5--$FeTi_2O_5$ ss), hercynite ($FeAl_2O_4$), ulvospinel (Fe_2TiO_4), ilmenite ($FeTiO_3$, and BAIT (essentially $BaAl_2Fe_8TI_{13}O_{38}$). Waste loadings about 4 times higher than those of glass could readily be achieved, offering considerable economic and safety benefits relative to glass technology.

172 (CONF-790420--, pp 179-182) USE OF NATURAL ALUMINOSILICATES AND POROUS CERAMIC MATERIALS FOR THE INCLUSION OF RADIOACTIVE WASTES. Lazarev, L.N.; Shashukov, E.A.; Kuznetsov, Yu.V.; Lyubtsev, R.I. (Radium Inst., Leningrad, USSR). May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

Data on using the porous inorganic materials, such as diatomite and shamote, for the incorporation of radioactive wastes are presented. In laboratory-scale experiments on simulated liquid wastes it has been shown that 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. This product is characterized by a sodium leaching rate of the order of 10^{-5} g/cm^2 . d and good resistance to crystallization. The content of various oxides in the wastes can attain 15 to 20 wt %. The data on the dependence of plasticity and open porosity of the clay-like products on Na_2O, SrO, ZrO_2, and MnO_2 content are also given. 3 figures, 3 tables.

173 (CONF-790420--, pp 183-187) VITRIFICATION OF ICPP HIGH-LEVEL ZIRCONIA CALCINE. Berreth, J.R.; Gombert, D. II; Cole, H.S. (Allied Chemical Corp., Idaho Falls, ID). May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

High-level radioactive calcined defense waste (approx. 50 wt % CaF_2) is vitrified using a frit containing 66% SiO_2, 24% Na_2O, 8% B_2O_3, and 2% CuO. Effects of Na, B, Li, Zn, Cu, and P on viscosity and acid leach resistance were measured. The glass contains up to 9% fluoride. Glass properties were measured. 6 tables.

174 (CONF-790420--, pp 188-192) DEVELOPMENT AND CHARACTERIZATION OF THE GLASS-CERAMIC VCP 15 FOR IMMOBILIZATION OF HIGH-LEVEL-LIQUID RADIOACTIVE WASTE. Guber, W. (Kernforschungszentrum, Karlsruhe, Germany); Hussain, M.; Kahl, L.; Mueller, W.; Saidl, J. May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, Oh, USA (30 Apr 1979).

Complementary to the glass developments for HAW solidification, a glass-ceramic matrix was synthesized (VC 15) and the derived HAW-containing product (VCP 15) was investigated on both inactive and highly radioactive scales. In comparing the glass-ceramic matrix with the glass products, an improvement in thermal properties was found, but the leaching resistance was only slightly better. The main advantage of the glass-ceramic system is its higher thermodynamic equilibrium, which increases the long-term stability and safety of solidified HAW in final storage.

175 (CONF-790420--, pp 321-326) POROUS GLASS MATRIX METHOD FOR ENCAPSULATING HIGH-LEVEL NUCLEAR WASTES. Macedo, P.B.; Tran, D.C.; Simmons, J.H.; Saleh, M.; Barkatt, A.; Simmons, C.J..; Lagakos, N.; DeWitt, E. (Catholic Univ. of America, Washington, DC). May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

A novel process which uses solidified porous high-silica glass powder to fixate radioactive high-level wastes is described. The process yields cylinders consisting of a core of high-silica glass containing the waste elements in its structure and a protective layer also of high-silica glass completely free of waste elements. The process can be applied to waste streams containing 0 to 100% solids. The core region exhibits a higher coefficient of thermal expansion and a lower glass transition temperature than the outer protective layer. This leads to mechanical strengthening of the glass and good resistance to stress corrosion by the development of a high residual compressive stress on the surface of the sample. Both the core and the protective layer exhibit extremely high chemical durability and offer an effective fixation of the radioactive waste elements, including ^{239}Pu and ^{99}Tc which have long half-lives, for calculated periods of more than 1 million years, when temperatures are not allowed to rise above 100°C.

176 (CONF-790420--, pp 333-341) UTILIZATION OF BOROSILICATE GLASS FOR TRANSURANIC WASTE IMMOBILIZATION. Ledford, J.A.; Williams, P.M. (Rockwell International, Golden, CO). May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA

(30 Apr 1979).

Incinerated transuranic waste and other low-level residues have been successfully vitrified by mixing with boric acid and sodium carbonate and heating to 1050°C in a bench-scale continuous melter. The resulting borosilicate glass demonstrates excellent mechanical durability and chemical stability.

177 (CONF-790420--, pp 342-348) CRYSTALLOCHEMICAL STABILIZATION OF RADWASTE ELEMENTS IN PORTLAND CEMENT CLINKER. Jantzen, C.M.; Glasser, F.P. (Univ. of Aberdeen, Old Aberdeen, Scotland). May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

Most of the elements present in a model PW-4b radwaste have been successfully incorporated during clinkering into the calcium silicate and ferrite phases comprising Portland cement. The bulk composition of a model waste has been systematically altered by the addition of other nonradioactive components to produce an inherently cementitious material. This has the advantage that (1) the radwaste ions are substituted directly into the crystal structure of the anhydrous cement phases, (2) the unhydrated cement continues to behave as a ceramic, (3) during hydration the radwaste ions can be more readily incorporated into the hydration products, and (4) the hydration products should represent a close approach to thermodynamic stability during storage in geological environments, e.g., low pressures and temperatures between 0°C and approx. 180°C. 4 figures, 4 tables.

178 (CONF-790420--, pp 354-358) PROCESSES FOR IMMOBILIZATION OF HIGH-LEVEL SOLID WASTES BY GLASS AND CERAMIC MATRICES. Lin, K.H.; Clark, W.E.; Howerton, W.B. (Oak Ridge National Lab., TN). May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

Processes for the immobilization of waste SiC hulls and fissile particles from reprocessing of HTGR fuels are presented. Compressed cylinders of waste-matrix material were formed under pressures of approx. 35, 71, or 142 MPa and sintered at temperatures ranging from approx. 500 to 1025°C for approx. 1 h. The cylinders containing 50% SiC-25% soft glass-25% red clay possess the most desirable properties (high mechanical strength, low surface area, and low leachability) of the several different compositions studied. Within the pressure and temperature ranges studied, the pressure of cylinder formation and the sintering temperature appear to have relatively minor effects on those properties except for the initial cesium leachability. Higher pressure and temperature would tend to reduce the initial leachability significantly. 5 tables.

179 (CONF-790420--, pp 359-364) INCORPORATION OF PRECIPITATE FROM TREATMENT OF MEDIUM-LEVEL LIQUID RADIOACTIVE WASTE INTO GLASS MATRIX OR CERAMICS TOGETHER WITH HIGH-LEVEL LIQUID WASTE. Hussain, M. (Kernforschungszentrum, Karlsruhe GmbH, Germany); Kahl, L. May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

MAW-splitting, i.e., precipitation and separation of the radioactive matter from the bulk liquid, has been developed at Kernforschungszentrum Karlsruhe (KfK) as an alternataive to direct solidification of MLLW-concentrates by bituminization or cementation. More than 90% of the total radioactivity is removed from MLLW stream by absorbing it on surface rich precipitate, which is ultimately vitrified with HLLW. An addition of MLLW-precipitate to a matrix (glass or glass-ceramic), to which HLLW has already been incorporated, showed no significant decrease in product quality or increase in product amount. 5 tables.

180 (CONF-790420--, pp 370-373) STABILIZATION OF HIGH-LEVEL WASTE FROM A CHLORIDE VOLATILITY NUCLEAR FUEL REPROCESSING SYSTEM. Smith, L.A.; Thornton, T.A. (Babcock and Wilcox Co., Lynchburg, VA). May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

Methods for stabilizing high-level waste from a chloride volatility thorium-based fuel coprocessing system have been studied. The waste, which is present as chloride salts, is combined with SiO_2 or Al_2O_3 and pyrohydrolyzed to remove the chloride ions. The resulting solid is then combined with a flux and glassified. 3 figures, 4 tables.

181 (CONF-790420--, pp 374-377) GLASS-CRYSTALLINE MATERIALS FOR ACTIVE WASTE INCORPORATION. Kulichenko, V.V.; Krylova, N.V.; Vlasov, V.I.; Polyakov, A.S. (State Committee for the Utilization of Atomic Energy, Moscow, USSR). May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

This paper presents the results of investigations into the possibility and conditions for using glass-crystalline materials for the incorporation of radionuclides. Materials of a cast pyroxene type that are obtained by smelting calcined wastes with acid blast furnace slags are described. A study was also made of materials of a basalt type prepared from wastes with and without alkali metal salt. Changes in the structure and properties of materials in the process of storage at different temperatures have been studied.

182 (CONF-790420--, pp 229-232) USE OF GLASS-CERAMIC MATERIALS FOR THE FIXATION OF RADIOACTIVE WASTES. Minaev, A.A.; Cziraner, S.N.; Prokhorova, N.P. (State Committee on Peaceful Use of Atomic Energy, Moscow, USSR). May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

This paper is concerned with the study of the crystallization of phosphate and silicate glasses. It was shown that temperature and time of storage influence considerably the crystallization of glasses and that crystallization very often increases their rates of leaching to a great extent. However, there are glasses in which crystallization does not result in leaching rate increase. It seems reasonable to use these materials for the fixation of radioactive wastes. The main reasons for the increase in the leaching rate during crystallization are the formation of porosity and soluble crystal phases.

183 (CONF-790420--, pp 25-26) PRINCIPAL TRENDS OF INVESTIGATIONS CARRIED ON IN THE USSR ON INCORPORATION OF HIGHLY ACTIVE WASTES INTO GLASS AND CERAMIC TYPE MATERIALS. Polyakov, A.S. (State Committee for the Utilization of Atomic Energy, Moscow, USSR). May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

Studies in the USSR on incorporation of high level waste into glass and ceramic materials are described.

184 (CONF-791016--, pp 7-19) COMPONENT EFFECTS IN MIXTURE EXPERIMENTS. Piepel, G.F. (Battelle Pacific Northwest Labs., Richland, WA). Sep 1980. NTIS, PC A11/MF A01.

From 1979 DOE statistical symposium; Gatlinburg, TN, USA (24 Oct 1979).

In a mixture experiment, the response to a mixture of q components is a function of the proportions x_1, x_2, ..., x/sub q/ of components in the mixture. Experimental regions for mixture experiments are often defined by constraints on the proportions of the components forming the mixture. The usual (orthogonal direction) definition of a factor effect does not apply because of the dependence imposed by the mixture restriction, /sup q/Σ/ sub i=1/ x/sub i/ = 1. A direction within the experimental region in which to compute a mixture component effect is presented and compared to previously suggested directions. This new direction has none of the inadequacies or errors of previous suggestions while having a more meaningful interpretation. The distinction between partial and total effects is made. The uses of partial and total effects (computed using the new direction) in modification and interpretation of mixture response prediction equations are considered. The suggestions of the paper are illustrated in an example from a glass development study in a waste vitrification program. 5 figures, 3 tables.

185 (CONF-801038--(Vol.2), pp 891-910) NUCLEAR WASTE VITRIFICATION EFFLUENT. Goles, R.W.; Brauer, F.P.; Hamilton, D.C.; Fager, J.E. (Battelle Pacific Northwest Labs., Richland, WA). Feb 1981. NTIS, PC A99/MF A01.

From 16. DOE nuclear air cleaning conference; San Diego, CA, USA (20 Oct 1980).

Important gaseous and airborne off-gas emissions associated with the solidification of high level liquid waste have been characterized. Effluents specifically sampled for in this off-gas monitoring study include: the gaseous chemical species of ^{3}H, ^{14}C, ^{85}Kr, ^{129}I and NO/sub x/; the semivolatile forms of ^{79}Se, ^{99}Tc, ^{106}Ru, $^{125\ 126}Sb$, /sup 125m/Te and $^{134\ 135\ 137}Cs$; and all radionuclides contained in particulate matter. The sampling and laboratory techniques employed and the off-gas analytical results obtained in this study are described herein. In addition, the magnitudes of vitrification-produced gaseous emissions are compared, whenever possible, to those associated with the complete fuel reprocessing cycle

186 (DOE/ET--0028(Vol.1)) TECHNOLOGY FOR COMMERCIAL RADIOACTIVE WASTE MANAGEMENT. (Battelle Pacific Northwest Labs., Richland, WA (USA)). May 1979. Contract EY-76-C-06-1830. 327p. Dep. NTIS, PC A15/MF A01.

The scope of this report is limited to technology for management of post-fission wastes produced in the commercial nuclear power light water reactor fuel cycle. Management of spent fuel (as a waste), high-level and other transuranic wastes, and gaseous wastes are characterized. Non-transuranic wastes are described, but management of these wastes, except for gaseous wastes, is excluded from the scope of this report. Volume 1 contains the summary and the bases and background information.

187 (DOE/ET--0028(Vol.2)) TECHNOLOGY FOR COMMERCIAL RADIOACTIVE WASTE MANAGEMENT. (Battelle Pacific Northwest Labs., Richland, WA (USA)). May 1979. Contract EY-76-C-06-1830. 564p. Dep. NTIS, PC A24/MF A01.

Conceptual processes and facilities for treating gaseous and various transuranium (TRU) wastes produced during the post fission portion of the light water reactor fuel cycle are described in volume 2. The goal of the treatment process for TRU wastes and for long-lived radionuclides removed from the gaseous waste streams is to convert these wastes to stable products suitable for placement in geologic isolation repositories. The treatment concepts are based on available technology. They do not necessarily represent an optimum design but are representative of what could be achieved with current technology. In actual applications it is reasonable to expect that there could be some improvement over these concepts that might be reflected in either lower costs or lower environmental impacts or both. These conceptual descriptions do provide a reasonable basis for cost analysis and for development of estimates of environmental impacts. The waste treatment technologies considered here include: high-level waste solidification, packaging of fuel residue, failed equipment and noncombustible waste treatment, general trash and combustible waste treatment, degraded solvent treatment, dilute aqueous waste pretreatment, immobilization of wet and solid wastes, off-gas particle removal systems, fuel reprocessing plant dissolver off-gas treatment, process off-gas treatment, and fuel reprocessing plant atmospheric protection system.

188 (DOE/ET--0028(Vol.5)) TECHNOLOGY FOR COMMERCIAL RADIOACTIVE WASTE MANAGEMENT. (Battelle Pacific Northwest Labs., Richland, WA (USA)). May 1979. Contract EY-76-C-06-1830. 642p. Dep. NTIS, PC A99/MF A01.

An analysis of the complete waste management system was developed to assess the total impact of managing radioactive wastes generated over the entire lifetime of a nuclear power system. The analysis considers the treatment and disposal of all post-fission TRU, gaseous and airborne and decommissioning wastes. Each radioactive waste stream is tracked each year from its origin through treatment, storage, transport, and accumulation in a geologic repository. The reference system is based on 400 GWe of nuclear power installed in the year 2000 and produces approximately 10,000 GWe-years of electric energy. An alternative low-growth projection based on 255 GWe in the year 2000 is also considered, but for fewer cases. This system produces approximately 6400 GWe year of electric energy. Capacity additions beyond the year 2000 are not considered a part of this system. After 40 years of operation each nuclear power plant is shut down and decommissioned. Thus, the last nuclear power plant is shut down in the year 2040. The last fuel reprocessing plant is shut down in the year 2044 and dismantled in the year 2075. Thus, the system operation encompasses a 101-year period from 1975 through 2075. In addition, the decay of radioactivity in the final repositories is followed over a million year period.

189 (DOE/TIC--11433(App.)) PRECONCEPTUAL DESIGN STUDY FOR SOLIDIFYING HIGH-LEVEL WASTE: WEST VALLEY DEMONSTRATION PROJECT. Hill, O.F. (comp.). (Battelle Pacific Northwest Labs., Richland, WA (USA)). Apr 1981. Contract AC06-76RL01830. 162p. (PNL--3608-2). NTIS, PC A08/MF A01.

This report presents a preconceptual design study for processing radioactive high-level liquid waste presently stored in underground tanks at Western New York Nuclear Service Center (WNYNSC) near West Valley, New York, and for incorporating the radionuclides in that waste into a solid. The high-level liquid waste accumulated from the operation of a chemical reprocessing plant by the Nuclear Fuel Services, Inc. from 1966 to 1972. The high-level liquid waste consists of approximately 560,000 gallons of alkaline waste from Purex process operations and 12,000 gallons of acidic (nitric acid) waste from one campaign of processing thoria fuels by a modified Thorex process (during this campaign thorium was left in the waste). The alkaline waste contains approximately 30 million curies and the acidic waste contains approximately 2.5 million curies. The reference process described in this report is concerned only with chemically processing the high-level liquid waste to remove radionuclides from the alkaline supernate and converting the radionuclide-containing nonsalt components in the waste into a borosilicate glass.

190 (DOE/TIC--11433(App.A,B,C)) PRECONCEPTUAL DESIGN STUDY FOR SOLIDIFYING HIGH-LEVEL WASTE: WEST VALLEY DEMONSTRATION PROJECT. (Battelle Pacific Northwest Labs., Richland, WA (USA); Automation Industries, Inc., Richland, WA (USA). Vitro Engineering Div.). Apr 1981. Contract AC06-76RL01830. 678p. (PNL--3608-2(App.A,B,C)). NTIS, PC A99/MF A01.

This Appendix contains the preconceptual design drawings prepared by Vitro Engineering Corporation for Pacific Northwest Laboratory. The following types of drawings are included in this Appendix: process flow diagrams; process and instrumentation diagrams; hydraulic diagrams; equipment arrangement drawings; service gallery drawings; electric power one-line diagram; equipment line lists; and outline specifications. The basic purpose of these drawings was to determine the feasibility of installing the reference solidification process in existing cells at the Western New York Nuclear Service Center. Most of the process and vitrification equipment will be installed in the former Chemical Processing Cell, while the salt solidification equipment will be housed in the former Scrap Removal Room. The design utilized a remote maintenance and operation concept.

191 (HEDL-SA--2235) PROCESS CONSIDERATIONS FOR HOT PRESSING CERAMIC NUCLEAR WASTE FORMS. Wilson, C.N.; Brite, D.W. (Hanford Engineering Development Lab., Richland, WA (USA); Battelle Pacific Northwest Labs., Richland, WA (USA)). 25 Mar 1981. Contract AC14-76FF02170. 25p. (CONF-810528--7). NTIS, PC A02/MF A01. Order Number DE81023183.

From 83. symposium on nucleation and crystallization in glasses; Washington, DC, USA (3 May 1981).

Spray calcined simulated ceramic nuclear waste powders were hot pressed in graphite, nickel-lined graphite and ZrO_2-lined Al_2O_3 dies. Densification, initial off-gas, waste element retention and pellet-die interactions were evaluated. Indicated process considerations and limitations are discussed. 15 figures.

192 (PNL--2265-4) RESEARCH AND DEVELOPMENT ACTIVITIES, WASTE FIXATION PROGRAM. QUARTERLY PROGRESS REPORT, OCTOBER--DECEMBER 1977. McElroy, J.L. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Mar 1979. Contract EY-76-C-06-1830. 33p. Dep. NTIS, PC A03/MF A01.

Foaming was experienced in the joule-heated ceramic melter during conversion of simulated Savannah River Plant waste to glass. The foaming tendency was decreased by adding the glass-making additives as chemical compounds rather than as pre-formed glass. Operation of the 36-in.-dia spray calciner with a 300 l/h feed rate was demonstrated. Over 300 h of operating experience has now been accumulated with this unit. The variation of leach rate with pH was measured on four typical waste glasses over the pH range of 1 to 13. All four glasses had uniformly low leach rates over the range of pH 4 to 9. A series of waste glass samples were prepared in bead form and were spiked individually with ^{99}Tc, ^{233}U, ^{237}Np, ^{239}Am, and ^{244}Cm, for studying the effect of various naturally occurring waters on leach rate. Demonstration laboratory encapsulations of four multibarrier waste forms on a 1-liter scale were completed. The waste forms were: (1) uncoated supercalcine in a vacuum-cast Al--12Si matrix, (2) PyC/Al_2O_3-coated supercalcine in a gravity-sintered copper matrix, (3) glass-coated supercalcine in a vacuum-cast Al--12Si matrix, and (4) simulated waste glass marbles in a vacuum-cast Pb--10Sn matrix. 11 figures, 2 tables.

193 (PNL--2481) DEVELOPMENT OF GLASS FORMULATIONS CONTAINING HIGH-LEVEL NUCLEAR WASTES. Ross, W.A. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Feb 1978. Contract EY-76-C-06-1830. 34p. Dep. NTIS, PC A03/MF A01.

The effects of Na_2O, K_2O, B_2O_3, TiO_2, CaO, and ZnO contents were determined on the leach rate and homogeneity of a potential high-level waste glass. It was found that the two characteristics are in competition, with variations of CaO having the greatest effect and TiO_2 having the least effect. Boron oxide content is important in control of Na_2MoO_4 formation and separation. Sodium molybdate formation can also be controlled by the use of reducing agents. The waste glass discussed in this report can tolerate from 0 to 50 percent waste content with minor effects on leachability and viscosity.

194 (PNL--2486) CRITICAL ASSESSMENT OF METHODS FOR TREATING AIRBORNE EFFLUENTS FROM HIGH-LEVEL WASTE SOLIDIFICATION PROCESSES. Christian, J.D.; Pance, D.T. (Scientific Advances, Inc., Columbus, OH (USA)). Jun 1977. Contract EY-76-C-06-1830. 162p. (SAI--77-571-LJ). Dep. NTIS, PC A08/MF A01.

Off-gas treatment systems are reviewed for high-temperature processes which are being developed for the solidification of high-level liquid wastes from nuclear fuel reprocessing plants. A brief description of each of the processes is given and detailed analyses are made of the expected magnitudes of airborne effluent release rates from each system. The estimated release rates of the various processes are compared with present and anticipated regulatory limits. A number of recommendations are made for additional development studies to better understand and control certain airborne effluents from the solidification processes.

195 (PNL--2622) ASSESSMENT OF WATER/GLASS INTERACTIONS IN WASTE GLASS MELTER OPERATION. Postma, A.K.; Chapman, C.C.; Buelt, J.L. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Apr 1980. Contract AC06-76RL01830. 88p. NTIS, PC A05/MF A01.

A study was made to assess the possibility of a vapor explosion in a liquid-fed glass melter and during off-standard conditions for other vitrification processes. The glass melter considered is one designed for the vitrification of high-level nuclear wastes and is comprised of a ceramic-lined cavity with electrodes for joule heating and processing equipment required to add feed and withdraw glass. Vapor explosions needed to be considered because experience in other industrial processes has shown that violent interactions can occur if a hot liquid is mixed with a cooler, vaporizable liquid. Available experimental evidence and theoretical analyses indicate that destructive glass/water interactions are low probability events, if they are possible at all. Under standard conditions, aspects of liquid-fed melter operation which work against explosive interactions include: (1) the aqueous feed is near its boiling point; (2) the feed contains high concentrations of suspended particles; (3) molten glass has high viscosity (greater than 20 poise); and (4) the glass solidifies before film boiling can collapse. While it was concluded that vapor explosions are not expected in a liquid-fed melter, available information does not allow them to be ruled out altogether. Several precautionary measures which are easily incorporated into melter operation procedures were identified and additional experiments were recommended.

196 (PNL--2654) REVIEW OF RADIOACTIVE WASTE IMMOBILIZATION IN CONCRETE. Lokken, R.O. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jun 1978. Contract EY-76-C-06-1830. 112p. Dep. NTIS, PC A06/MF A01.

A discussion is given of properties of concrete waste forms, as obtained through research on immobilization of radioactive wastes in concrete. Types of radioactive waste discussed include low-level and intermediate-level radioactive waste, simulated defense high-level waste sludges and calcines, simulated neutralized AGNS acid fuel reprocessing waste and simulated power reactor fuel cycle HLW calcines. The waste form properties include water/cement ratio, set times, curing exotherms, compressive strength, impact strength, Sr/Cs/ transuranics leachabilities, thermal conductivity, thermal stability, and radiation stability. A discussion is also given of the conditions and restrictions that govern the feasibility of immobilizing HLW in concrete. Results of theoretical calculations are discussed and illustrated, as are the effects of waste loading on material requirements. Properties of glass and concrete waste forms and of hot-pressed cement used for the solidification are compared. A conceptual process for HLW immobilization in concrete is discussed with emphasis on processing problems associated with heat and radiation effects on water. Use of hydraulic cements for the solidification of low heat generating wastes will produce a product with acceptable properties; the high heat generating rates and radioactivity of HLW makes feasibility assessment difficult. Hot-pressed cement may make HLW immobilization feasible. 27 tables, 21 figures, 70 references. (DLC)

197 (PNL--2664) LEACHING OF FULLY RADIOACTIVE HIGH-LEVEL WASTE GLASS. Bradley, D.J. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Sep 1978. Contract EY-76-C-06-1830. 101p. Dep. NTIS, PC A06/MF A01.

As part of continuing Department of Energy (DOE)-sponsored studies in waste management, the Pacific Northwest Laboratory (PNL) has been conducting the High-Level Waste Immobilization Program. The purpose of this program is to develop and demonstrate technology for incorporating nuclear wastes into final waste forms. The preparation and leach testing of fully radioactive, zinc borosilicate glass, which was prepared from power reactor waste, are described. Leach testing using the International Atomic Energy Association (IAEA) procedure was performed in deionized water for a period of 1.75 years. Leach rates were determined for activation products, fission products, and actinides. These rates ranged from 4 x 10^{-5} g of glass/cm^2-day, based on cesium, to 4 x 10^{-9} g of glass/cm^2-day, based on cerium. Following is the ranking of the release rates of the elements, from highest to lowest: Cs > Sr > Co > Sb > Mn > Pu > Eu > Rh > Cm > Ce. A similar leach test, using the same glass composition but with nonradioactive elements, has recently been completed. The leach rates of Cs and Sr for the nonradioactive glass were found to be in close agreement with those in this study. Slopes calculated from curves of cumulative fractions leached show that radioisotope release begins with a diffusion-type mechanism and changes gradually to a silicate lattice alteration mechanism. Changes in sampling frequency altered the apparent release mechanism when leachant changes were longer than one month. The leach rates were quite constant for samples taken from the top to the bottom of the glass melt, indicating a homogeneous product. Safety assessment studies and modeling programs use leach rates to predict the amount of radioactive material released should the waste be contacted by aqueous solutions. Further tests, focusing on geologic storage conditions and using fully radioactive wastes, are planned.

198 (PNL--2668-1) MULTIBARRIER WASTE FORMS. PART I. DEVELOPMENT. Rusin, J.M.; Lokken, R.O.; Lukacs, J.M.; Sump, K.R.; Browning, M.F.; McCarthy, G.J. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Sep 1978. Contract EY-76-C-06-1830. 199p. Dep. NTIS, PC A06/MF A01.

The multibarrier concept produces a composite waste form with enhanced inertness through improvements in thermal stability, mechanical strength, and leachability by the use of coatings and metal matrices. This report describes research and development activities resulting in the demonstration of the multibarrier concept for nonradioactive simulated waste compositions. The multibarrier concept is to utilize up to three barriers to isolate radionuclides from the environment: a solid waste inner core, an impervious coating, and a metal matrix. Two inner core materials, sintered supercalcine 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. Py/Al_2O_3 coatings deposited by chemical vapor deposition (CVD) and glass coatings have been applied to supercalcine cores to improve inertness. The purpose of the metal matrix is to improve impact resistance, protect the inner core rom any adverse environments, provide radiation shielding, and increase thermal conductivity, yielding lower internal temperatures. The development of gravity sintering and vacuum casting techniques for matrix encapsulation are discussed. Four multibarrier products were demonstrated: (1) Glass marbles encapsulated in vacuum-cast Pb-10Sn; (2) uncoated, sintered supercalcine pellets encapsulated in vacuum-cast Al-12Si; (3) glass-coated, sintered supercalcine pellets encapsulated in vacuum-cast Al-12Si; and (4) PyC/Al_2O_3-coated supercalcine encapsulated in gravity-sintered Cu. 23 figs., 20 tables.

199 (PNL--2668-3) MULTIBARRIER WASTE FORMS. PART III: PROCESS CONSIDERATIONS. Lokken, R.O. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Oct 1979. Contract EY-76-C-06-1830. 32p. Dep. NTIS, PC A03/MF A01.

The multibarrier concept for the solidification and storage of radioactive waste utilizes up to three barriers to isolate radionuclides from the environment: a solidified waste inner core, an impervious coating, and a metal matrix. The coating and metal matrix give the composite waste form enhanced inertness with improvements in thermal stability, mechanical strength, and leach resistance. Preliminary process flow rates and material costs were evaluated for four multibarrier waste forms with the process complexity increasing thusly: glass marbles, uncoated supercalcine, glass-coated supercalcine, and PyC/Al_2O_3-coated supercalcine. This report discusses the process variables and their effect on optimization of product quality, processing simplicity, and material cost. 11 figures, 2 tables. (DLC)

200 (PNL--2690) CONCEPTUAL DESIGN OF A NUCLEAR WASTE VITRIFICATION FACILITY. Larson, D.E.; Blair, H.T.; Bonner, W.F.; Garrett, A.A.; Hanson, M.S.; Romero, L.S.; Siemens, D.H. (Battelle Pacific Northwest Labs., Richland, WA (USA)). May 1978. Contract EY-76-C-06-1830. 69p. Dep. NTIS, PC A04/MF A01.

This document describes a conceptual high-level waste immobilization facility. The facility would have the capability to calcine and then vitrify high-level liquid waste (HLLW). The vitrification would be accomplished in a canister which is seal-welded, checked for integrity, and decontaminated for movement to storage. Included in the facility would be the capability to repair faulty canister-lid seal welds, overpack failed canisters, and treat the process off-gas and cell ventilation air prior to release to the fuel reprocessing plant (FRP) atmospheric protection system (APS). The nuclear waste vitrification facility (NWVF) would be an integral part of the FRP structure. The operations of the facilities would be centered in the waste vitrification cell (WVC) which performs most of the facility functions. The cell is a reinforced concrete hot cell, lined with stainless steel. Most operation and maintenance activities would be performed remotely using a crane equipped with an impact wrench or yoke. The major facility equipment includes a feed tank, spray calciner, two melters, weld-inspection stations, canister storage rack, and a canister decontamination cubicle. Installation and removal of equipment in the cell would be done through shielding doors. The air lock system of the canister decontamination cubicle would permit placement and removal of the canister. Activities in the cell may be observed through four shielding glass windows and/or up to three periscopes. The operating, service, and pipe galleries which house operating personnel and equipment necessary for cell operations are located adjacent to the cell.

201 (PNL--2735) LIQUID-FED CERAMIC MELTER: A GENERAL DESCRIPTION REPORT. Buelt, J.L.; Chapman, C.C. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Oct 1978. Contract EY-76-C-06-1830. 25p. Dep. NTIS, PC A02/MF A01.

The Pacific Northwest Laboratory is conducting several research and development programs for the solidification of high-level wastes. The liquid-fed ceramic melter (LFCM) is a major component in the solidification process. This melter can solidify liquid high-level waste, as well as melt calcined waste with glass additives and then solidify the mixture. This report describes the LFCM system and shows the main features of the refractories, electrodes and power systems, melter box and lid, draining system, feeding system, and off-gas system.

202 (PNL--2786) PROCESSING OF WASTE SOLUTIONS FROM ELECTROCHEMICAL DECONTAMINATION. Charlot, L.A.; Allen, R.P.; Arrowsmith, H.W.; Hooper, J.L. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Sep 1979. Contract EY-76-C-06-1830. 52p. Dep. NTIS, PC A04/MF A01.

The use of electropolishing as a decontamination technique will be effective only if we can minimize the amount of secondary waste requiring disposal and economically recycle part of the decontamination electrolyte. Consequently, a solution purification method is needed to remove the dissolved contamination and metal in the electrolyte. This report describes the selection of a purification method for a phosphoric acid electrolyte from the following possible acid reclamation processes: ion exchange, solvent extraction, precipitation, distillation, electrolysis, and membrane seperation.

203 (PNL--2809) PHYSICAL MODELING OF JOULE HEATED CERAMIC GLASS MELTERS FOR HIGH LEVEL WASTE IMMOBILIZATION. Quigley, M.S.; Kreid, D.K. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Mar 1979. Contract EY-76-C-06-1830. 159p. Dep. NTIS, PC A08/MF A01.

This study developed physical modeling techniques and apparatus suitable for experimental analysis of joule heated ceramic glass melters designed for immobilizing high level waste. The physical modeling experiments can give qualitative insight into the design and operation of prototype furnaces and, if properly verified with prototype data, the physical models could be used for quantitative analysis of specific furnaces. Based on evaluation of the results of this study, it is recommended that the following actions and investigations be undertaken: It was not shown that the isothermal boundary conditions imposed by this study established prototypic heat losses through the boundaries of the model. Prototype wall temperatures and heat fluxes should be measured to provide better verification of the accuracy of the physical model. The VECTRA computer code is a two-dimensional analytical model. Physical model runs which are isothermal in the Y direction should be made to provide two-dimensional data for more direct comparison to the VECTRA predictions. The ability of the physical model to accurately predict prototype operating conditions should be proven before the model can become a reliable design tool. This will require significantly more prototype operating and glass property data than were available at the time of this study. A complete set of measurements covering power input, heat balances, wall temperatures, glass temperatures, and glass properties should be attempted for at least one prototype run. The information could be used to verify both physical and analytical models. Particle settling and/or sludge buildup should be studied directly by observing the accumulation of the appropriate size and density particles during feeding in the physical model. New designs should be formulated and modeled to minimize the potential problems with melter operation identifed by this study.

204 (PNL--2904) VITRIFICATION OF HANFORD WASTES IN A JOULE-HEATED CERAMIC MELTER AND EVALUATION OF RESULTANT CANISTERIZED PRODUCT. Chapman, C.C.; Buelt, J.L.; Slate, S.C.; Katayama, Y.B.; Bunnell, L.R. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Aug 1979. Contract EY-76-C-06-1830. 74p. Dep. NTIS, PC A04/MF A01.

Experience gained in the week-long vitrification test and characterization of the glass produced in the run support the following conclusions: The Hanford waste simulated in this test can be readily vitrified in a joule-heated ceramic melter. Physical properties of the molten glass were entirely compatible with melter operation. The average feed rate of 106 kg/h is high enough to make the ceramic melter a feasible piece of equipment for vitrifying Hanford wastes. The glass produced in this trial had good chemical durability, $6(10)^{-5}$ g/cm^2-d. When one of the canisters was purposely dropped onto a steel pad, the damage was limited to deformation of the steel can in the impact area, cracking of a weld, and fracturing of glass in the immediate vicinity of the impact area. No glass was released from the canister as a result of the drop test. The results of this vitrification test support the technical feasibility of vitrifying Hanford wastes by means of a joule-heated ceramic melter. Surface area for large glass castings is equivalent to the mass median particle diameters between 4.27 cm (1.75 in.) and 8.91 cm (3.51 in.) even when allowed to cool rapidly by standing in ambient air. Large canisters (up to 0.91 m in dia) can be cast without large voids while standing in air if the fill rate is over 100 kg/h. 34 figures, 10 tables.

205 (PNL--2925) IN-CAN MELTING PROCESS AND EQUIPMENT DEVELOPMENT FROM 1974 TO 1978. Blair, H.T. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Aug 1979. Contract EY-76-C-06-1830. 160p. Dep. NTIS, PC A08/MF A01.

Both the defense HLLW stores in tanks presently and the HLLW from proposed reprocessing of commercial LWR fuel can be vitrified as borosilicate glass in containers made of 300-series stainless steel by the ICM (in-can melting) process. Melting rates of 50 kg/h in 12-in.-dia cans and 117 kg/h in 28-in.-dia cans can be achieved in the ICM by using the rising-level charging method and internal heat-transfer plate assemblies in the cans. The ICM process can be monitored and remotely controlled without the aid of instrumentation attached to the waste can. The ICM process is compatible with both heated-wall spray calciners and fluidized-bed calciners. The ICM process causes residual tensile stresses as high as the yield strength in vitrified product containers made of 300-series stainless steel. Spall due to oxidation of the exterior of the can during an ICM process can be prevented by using an inert cover gas, by putting a protective coating on the can surface, or by using an oxidation-resistant alloy. Processing problems are minimized and product quality is improved when the complete can is located inside the furnace chamber by setting it on the hearth. A maximum of 24 kW and an average of 15 kW is required per 15-in.-high furnace zone to melt waste borosilicate glass at a rate of 117 kg/h in a 28-in.-dia ICM.

206 (PNL--3027) DESIGN FEATURES OF THE LABORATORY-SCALE RADIOCHEMICAL IMMOBILIZATION SYSTEM. Hanson, M.S.; Knox, C.A.; Berger, D.N. (Battelle Pacific Northwest Labs., Richland, WA (USA)). May 1980. Contract AC06-76RL01830. 46p. NTIS, PC A03/MF A01.

Under the High-Level Waste Immobilization Program, the Pacific Northwest Laboratory (PNL) is studying various ways to solidify high-level nuclear wastes. A variety of waste forms and processes are being investigated, with the most highly developed process being spray calcination coupled with in-can melting. This report describes a remote laboratory-scale system that was designed for the purpose of investigating the effects of different operating conditions and waste compositions on the product and on the effluents generated. It is termed laboratory-scale because of its nominal 1 L/h feed rate as compared to well over 300 L/h for full-scale equipment at PNL. The equipment currently consists of a feed system, a spray calciner, an in-can melter, and an effluent control system. It is operated in a shielded radiochemical hot cell using radioactive high-level liquid waste (HLLW) to answer questions on the deposition of radiochemicals during actual waste processing. The effluent control system can be modified in order to test different effluent systems, one of which has been proposed by the Savannah River Laboratories (SRL) for use in the Savannah River Plant vitrification system. The laboratory-scale system can also be used to test alternative immobilization processes, since spray calcination is a common processing step in many alternative waste form flowsheets. Thus, only the addition of a specific forming step such as pelletizing or sintering is necessary.

207 (PNL--3035) REVIEW OF HIGH-LEVEL WASTE FORM PROPERTIES. Rusin, J.M. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Dec 1980. Contract AC06-76RL01830. 133p. NTIS, PC A07/MF A01.

This report is a review of waste form options for the immobilization of high-level-liquid wastes from the nuclear fuel cycle. This review covers the status of international research and development on waste forms as of May 1979. Although the emphasis in this report is on waste form properties, process parameters are discussed where they may affect final waste form properties. A summary table is provided listing properties of various nuclear waste form options. It is concluded that proposed waste forms have properties falling within a relatively narrow range. In regard to crystalline versus glass waste forms, the conclusion is that either class of crystalline materials can be shown to have some advantage when a single property is considered; however, at this date no single waste form offers optimum properties over the entire range of characteristics investigated. A long-term effort has been applied to the development of glass and calcine waste forms. Several additional waste forms have enough promise to warrant continued research and development to bring their state of development up to that of glass and calcine. Synthetic minerals, the multibarrier approach with coated particles in a metal matrix, and high pressure-high temperature ceramics offer potential advantages and need further study. Although this report discusses waste form properties, the total waste management system should be considered in the final selection of a waste form option. Canister design, canister materials, overpacks, engineered barriers, and repository characteristics, as well as the waste form, affect the overall performance of a waste management system. These parameters were not considered in this comparison.

208 (PNL--3038) TECHNICAL SUMMARY: NUCLEAR WASTE VITRIFICATION PROJECT. Wheelwright, E.J.; Bjorklund, W.J.; Browne, L.M.; Bryan, G.H.; Holton, L.K.; Irish, E.R.; Siemens, D.H. (Battelle Pacific Northwest Labs., Richland, WA (USA)). May 1979. Contract EY-76-C-06-1830. 82p. Dep. NTIS, PC A05/MF A01.

Six PWR fuel assemblies, containing 2.3 metric tons uranium from Point Beach, have been processed by a conventional Purex-type process. U and other chemicals were added to the dilute HLLW, and the waste was then vitrified to produce two canisters of glass. The on-stream efficiency of the waste preparation facility exceeded 90% for the first 3 weeks; the overall average was 62%. The only processing difficulty in the vitrification facility was a partial failure in the spray calciner nozzle. The Pu byproduct of waste preparation was purified by ion exchange and calcined to oxide; one can of oxide ruptured due to self-heating. 27 figures, 16 tables. (DLC)

209 (PNL--3050-1) RESEARCH AND DEVELOPMENT ACTIVITIES: HIGH-LEVEL WASTE IMMOBILIZATION PROGRAM. QUARTERLY PROGRESS REPORT, JANUARY-MARCH 1979. McElroy, J.L.; Mendel, J.E.; Bonner, W.F.; Henry, M.H. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Nov 1979. Contract EY-76-C-06-1830. 50p. Dep. NTIS, PC A03/MF A01.

Liquid waste, made from zirconium-clad UO_2 power reactor fuel with an average burnup of 25,000 MWd/MT, was converted to glass by the in-can melting process. An intrinsic-gamma melt-level detection system was tested during the NWVP demonstrations; results showed that if a sufficient number of collimators are used the system will track the melt surface with a precision of 1 in. during the filling of cans with waste glass. The two canisters filled in the NWVP are both 8 in. in diameter and contain borosilicate glass of very similar compositions. One canister contains 116 kg of glass that generated 0.38 kW of self-heat when produced; the other contains 145 kg of glass, and generates 1.01 kW. Spray calcination of simulated Savannah River Plant liquid waste at a rate of 400 L/h was demonstrated in the 36-in.-dia. calciner. Five waste forms are being compared: concrete-containing waste calcine, sintered waste glass, glass-ceramic, Synroc B (a crystalline assemblage of titanates), and borosilicate waste glass (composition 76-68). Results of initial tests indicate that the reaction rate of carbon with water, previously found to be very low, may be increased in a radiation field.

210 (PNL--3060) DEVELOPMENT AND CHARACTERIZATION OF SOLIDIFIED FORMS FOR HIGH-LEVEL WASTES: 1978. ANNUAL REPORT. Ross, W.A.; Mendel, J.E. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Dec 1979. Contract EY-76-C-06-1830. 108p. Dep. NTIS, PC A06/MF A01.

Development and characterization of solidified high-level waste forms are directed at determining both process properties and long-term behaviors of various solidified high-level waste forms in aqueous, thermal, and radiation environments. Waste glass properties measured as a function of composition were melt viscosity, melt electrical conductivity, devitrification, and chemical durability. The alkali metals were found to have the greatest effect upon glass properties. Titanium caused a slight decrease in viscosity and a significant increase in chemical durability in acidic solutions (pH-4). Aluminum, nickel and iron were all found to increase the formation of nickel-ferrite spinel crystals in the glass. Four multibarrier advanced waste forms were produced on a one-liter scale with simulated waste and characterized. Glass marbles encapsulated in a vacuum-cast lead alloy provided improved inertness with a minimal increase in technological complexity. Supercalcine spheres exhibited excellent inertness when coated with pyrolytic carbon and alumina and put in a metal matrix, but the processing requirements are quite complex. Tests on simulated and actual high-level waste glasses continue to suggest that thermal devitrification has a relatively small effect upon mechanical and chemical durabilities. Tests on the effects radiation has upon waste forms also continue to show changes to be relatively insignificant. Effects caused by decay of actinides can be estimated to saturate at near 10^{19} alpha-events/cm^3 in homogeneous solids. Actually, in solidified waste forms the effects are usually observed around certain crystals as radiation causes amorphization and swelling of th crystals.

211 (PNL--3090) THERMAL CONDUCTIVITY OF MULTIBARRIER WASTE FORM COMPONENTS. Lokken, R.O. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jan 1981. Contract AC06-76RL01830. 52p. NTIS, PC A04/MF A01.

The multiple barrier concept of radioactive waste immobilization under investigation at Pacific Northwest Laboratory (PNL) uses composite waste forms which exhibit enhanced inertness through improvements in thermal stability, mechanical strength, and leachability by the use of coatings and metal matrices. Since excessive heat may be generated by radioactive decay of the waste, the thermal conductivity of the various barriers, and more importantly of the composite, becomes an important parameter in design criteria. This report presents results of thermal conductivity measurements on 21 various glass, ceramic, metal and composite materials used in multibarrier waste forms development.

212 (PNL--3094) HIGH-LEVEL WASTE IMMOBILIZATION PROGRAM: AN OVERVIEW. Bonner, W.R. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Sep 1979. Contract EY-76-C-06-1830. 46p. Dep. NTIS, PC A03/MF A01.

The High-Level Waste Immobilization Program is providing technology to allow safe, affordable immobilization and disposal of nuclear waste. Waste forms and processes are being developed on a schedule consistant with national needs for immobilization of high-level wastes stored at Savannah River, Hanford, Idaho National Engineering Laboratory, and West Valley, New York. This technology is directly applicable to high-level wastes from potential reprocessing of spent nuclear fuel. The program is removing one more obstacle previously seen as a potential restriction on the use and further development of nuclear power, and is thus meeting a critical technological need within the national objective of energy independence.

213 (PNL--3109) SPRAY CALCINATION/IN-CAN MELTING: EFFLUENT CHARACTERIZATION AND TREATMENT. Hanson, M.S. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Mar 1980. Contract EY-76-C-06-1830. 105p. Dep. NTIS, PC A06/MF A01.

According to data obtained during calcination of nonradioactive, simulated waste, ruthenium and cesium losses from the spray calciner are small, on the order of 3.5×10^{-2}% and 3.4×10^{-2}%, respectively. Calciner-melter and filter decontamination factors for ruthenium and cesium averaged 3.6×10^{4} and 3.9×10^{4}, respectively. Particulate decontamination factors of 10^{3} to 10^{4} have been obtained using sintered stainless steel filters. A significant portion of the ruthenium and cesium lost to the process off-gas system was due to particle penetration of the filters. The particles penetrating the filters have a mass distribution centering about a size large enough to control with available technology. Processing wastes containing fluoride will probably volatilize a portion of the fluoride to the off-gas system, thus increasing the probability of corrosion problems. 34 figures, 30 tables.

214 (PNL--3137) NUMERICAL MODELING OF LIQUID FEEDING IN THE LIQUID-FED CERAMIC MELTER. Hjelm, R.L.; Donovan, T.E. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Oct 1979. Contract EY-76-C-06-1830. 28p. Dep. NTIS, PC A03/MF A01.

A modeling scheme developed by the Pacific Northwest Laboratory numerically simulates the behavior of the Liquid-Fed Ceramic Melter (LFCM) during liquid feeding. The computer code VECTRA (Vorticity Energy Code for TRansport Analysis) was used to simulate the LFCM in the idling and liquid feeding modes. Results for each simulation include molten glass temperature profiles and isotherm contour plots, stream function contour plots, heat generation rate contour plots, refractory isotherms, and heat balances. The results indicated that the model showed no major deviations from real LFCM behavior and that high throughput should be attainable. They also indicated that reboil was a possibility as a steady liquid feeding state was approached, very steep temperature gradients exist in the Monofrax K-3, and that phase separation could occur in the bottom corners during liquid feeding and over the entire floor while idling.

215 (PNL--3145) VITRIFICATION OF RADIOACTIVE HIGH-LEVEL WASTE BY SPRAY CALCINATION AND IN-CAN MELTING. Hanson, M.S.; Bjorklund, W.J. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jul 1980. Contract AC06-76RL01830. 87p. NTIS, PC A05/MF A01.

After several nonradioactive test runs, radioactive waste from the processing of 1.5 t of spent, light-water-reactor fuel was successfully concentrated, dried and converted to a vitreous product. A total of 97 L of waste glass (in two stainless steel canisters) was produced. The spray calcination process coupled to the in-can melting process, as developed at Pacific Northwest Laboratory, was used to vitrify the waste. An effluent system consisting of a variety of condensation of scrubbing steps more than adequately decontaminated the process off gas before it was released to the atmosphere.

216 (PNL--3170) CONNECTING SECTION AND ASSOCIATED SYSTEMS CONCEPT FOR THE SPRAY CALCINER/IN-CAN MELTER PROCESS. Petkus, L.L.; Gorton, P.S.; Blair, H.T. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jun 1981. Contract AC06-76RL01830. 47p. NTIS, PC A03/MF A01.

For a number of years, researchers at the Pacific Northwest Laboratory have been developing processes and equipment for converting high-level liquid wastes to solid forms. One of these processes is the Spray Calciner/In-Can Melter system. To immobiliza high-level liquid wastes, this system must be operated remotely, and the calcine must be reliably conveyed from the calciner to the melting furnace. A concept for such a remote conveyence system was developed at the Pacific Northwest Laboratory, and equipment was tested under full-scale, nonradioactive conditions. This concept and the design of demonstration equipment are described, and the results of equipment operation during experimental runs of 7 d are presented. The design includes a connecting section and its associated systems - a canister support and alignment concept and a weight-monitoring system for the melting furnace. Overall, the runs demonstrated that the concept design is an acceptable method of connecting the two pieces of process equipment together. Although the connecting section has not been optimized in all areas of concern, it provides a first-generation design of a production-oriented system.

217 (PNL--3181) CHARACTERIZATION OF GASEOUS AND PARTICULATE EFFLUENTS FROM THE NUCLEAR VITRIFICATION PROJECT. Goles, R.W.; Hamilton, D.C.; Brauer, F.P.; Rieck, H.G. Jr.; Robertson, D.M.; Gordon, R.L.; Kaye, J.H. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Oct 1979. Contract EY-76-C-06-1830. 35p. Dep. NTIS, PC A03/MF A01.

Samples were taken during the second high-level liquid waste vitrification campaign associated with the NWVP. Sample analysis established the following total average emission levels in the undiluted vitrification off-gas stream: ^{3}H - 1.2 μCi/m^3, ^{14}C - 0.2 μCi/m^3, ^{129}I - 1 nCi/m^3, NO/sub x/ - 0.5%, ^{99}Tc - 1.6 pCi/m^3, and γ-emitters - 10^{5}(γ/s)/m^3. The aerosol size distribution is composed almost entirely of particles exhibiting smaller diameters than the minimum value for which absolute filters are rated, with an empirical geometric mean diameter of 0.13 μm and a particle mass concentration of 84.7 pg/cm^3. The particulate matter was composed of ^{106}Ru, ^{125}Sb, /sup 125m/Te, ^{134}Cs, ^{137}Cs, ^{144}Ce, ^{154} ^{155}Eu and ^{241}Am. The particulate emission levels in the undiluted process off-gas stream were: ^{106}Ru - 44 nCi/m^3, ^{125}Sb - 0.52 nCi/m^3, /sup 125m/Te - 1.7 nCi/m^3, ^{134}Cs - 7.8 nCi/m^3, ^{137}Cs - 52 nCi/m^3, ^{144}Ce - 0.8 μCi/m^3, ^{154}Eu - 0.3 nCi/m^3, ^{155}Eu - 0.18 nCi/m^3, and ^{241}Am - 0.19 nCi/m^3. All these environmental gaseous and airborne emissions level liquid waste were well within PNL waste management guidelines.

218 (PNL--3186) TECHNICAL REQUIREMENTS FOR THE CONTROL OF ^{129}I IN A NUCLEAR FUELS REPROCESSING PLANT. Burger, L.L.; Burns, R.E. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Nov 1979. Contract EY-76-C-06-1830. 34p. Dep. NTIS, PC A03/MF A01.

A proposed regulation would limit the release of ^{129}I to the environment to 0.005 Ci/GWe-yr of nuclear power produced. This corresponds to an overall ^{129}I retention factor of about 250 for the LWR fuel cycle. Technolgies available and under development for removing iodine from off-gas and weste water streams in an LWR nuclear fuel reprocessing plant and for converting iodine to stable forms and retaining it for interim and long-term storage, transportation and isolation are discussed. Possible developments are reviewed. Technologies available for retaining iodine from off-gas and excess water streams appear adequate, when properly combined, to meet the proposed regulation in an LWR fuel reprocessing plant, though additional work on solid sorbents is needed. Demonstrations under plant conditions are needed to define and resolve problems not met in laboratory experiments and to define practical operating conditions. Criteria for storage and disposal forms for retained iodine are needed to guide effective development of such forms. Because no disposal form for ^{129}I is likely to survive in any environment for even one half-life (1.6 x 10^7 yr), work on a disposal form should seek forms that disintegrate as slowly as possible to maximize dilution before the isotope reaches the biosphere.

219 (PNL--3188) EFFECTS OF COMPOSITION ON PROPERTIES IN AN 11-COMPONENT NUCLEAR WASTE GLASS SYSTEM. Chick, L.A.; Piepel, G.F.; Mellinger, G.B.; May, R.P.; Gray, W.J.; Buckwalter, C.Q. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Sep 1981. Contract AC06-76RL01830. 120p. NTIS, PC A06/MF A01. Order Number DE81030781.

Ninety simplified nuclear waste glass compositions within an 11-component oxide composition matrix were tested for crystallinity, viscosity, volatility, and chemical durability. Empirical models of property response as a function of glass composition were developed using statistical experimental design and modeling techniques. A new statistical technique was developed to calculate the effects of oxide components on each property. Independent melts were used to check the prediction accuracy of the models.

220 (PNL--3244) PRELIMINARY EVALUATION OF ALTERNATIVE WASTE FORM SOLIDIFICATION PROCESSES. VOLUME I. IDENTIFICATION OF THE PROCESSES. Treat, R.L.; Nesbitt, J.F.; Blair, H.T.; Carter, J.G.; Gorton, P.S.; Partain, W.L.; Timmerman, C.L. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Apr 1980. Contract AC06-76RL01830. 535p. NTIS, PC A23/MF A01.

This document contains preconceptual design data on 11 processes for the solidification and isolation of nuclear high-level liquid wastes (HLLW). The processes are: in-can glass melting (ICGM) process, joule-heated glass melting (JHGM) process, glass-ceramic (GC) process, marbles-in-lead (MIL) matrix process, supercalcine pellets-in-metal (SCPIM) matrix process, pyrolytic-carbon coated pellets-in-metal (PCCPIM) matrix process, supercalcine hot-isostatic-pressing (SCHIP) process, SYNROC hot-isostatic-pressing (SYNROC HIP) process, titanate process, concrete process, and cermet process. For the purposes of this study, it was assumed that each of the solidification processes is capable of handling similar amounts of HLLW generated in a production-sized fuel reprocessing plant. It was also assumed that each of the processes would be enclosed in a shielded canyon or cells within a waste facility located at the fuel reprocessing plant. Finally, it was assumed that all of the processes would be subject to the same set of regulations, codes and standards. Each of the solidification processes converts waste into forms that may be acceptable for geological disposal. Each process begins with the receipt of HLLW from the fuel reprocessing plant. In this study, it was assumed that the original composition of the HLLW would be the same for each process. The process ends when the different waste forms are enclosed in canisters or containers that are acceptable for interim storage. Overviews of each of the 11 processes and the bases used for their identification are presented in the first part of this report. Each process, including its equipment and its requirements, is covered in more detail in Appendices A through K. Pertinent information on the current state of the art and the research and development required for the implementation of each process are also noted in the appendices.

221 (PNL--3336) VIBRATORY FINISHING AS A DECONTAMINATION PROCESS. McCoy, M.W.; Arrowsmith, H.W.; Allen, R.P. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Oct 1980. Contract AC06-76RL01830. 52p. NTIS, PC A04/MF A01.

The major objective of this research is to develop vibratory finishing into a large-scale decontamination technique that can economicaly remove transuranic and other surface contamination from large volumes of waste produced by the operation and decommissioning of retired nuclear facilities. The successful development and widespread application of this decontamination technique would substantially reduce the volume of waste requiring expensive geologic disposal. Other benefits include exposure reduction for decontamination personnel and reduced risk of environmental contamination. Laboratory-scale studies showed that vibratory finishing can rapidly reduce the contamination level of transuranic-contaminated stainless steel and Plexiglas to well below the 10-nCi/g limit. The capability of vibratory finishing as a decontamination process was demonstrated on a large scale. The first decontamination demonstration was conducted at the Hanford N-Reactor, where a vibratory finisher was installed to reduce personnel exposure during the summer outage. Items decontaminated included fuel spacers, process-tube end caps, process-tube inserts, pump parts, ball-channel inspection tools and miscellaneous hand tools. A second demonstration is currently being conducted in the decontamination facility at the Hanford 231-Z Building. During this demonstration, transuranic-contaminated material from decommissioned plutonium facilities is being decontaminated to <10 nCi/g to minimize the volume of material that will require geologic disposal. Items that are being decontaminated include entire glove boxes, process-hood structural material and panels, process tanks, process-tank shields, pumps, valves and hand tools used during the decommissioning work.

222 (PNL--3343) HIGH-TEMPERATURE VITRIFICATION OF HANFORD RESIDUAL-LIQUID WASTE IN A CONTINUOUS MELTER. Barnes, S.M. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Apr 1980. Contract AC06-76RL01830. 39p. NTIS, PC A03/MF A01.

Over 270 kg of high-temperature borosilicate glass have been produced in a series of three short-term tests in the High-Temperature Ceramic Melter vitrification system at PNL. The glass produced was formulated to vitrify simulated Hanford residual-liquid waste. The tests were designed to (1) demonstrate the feasibility of utilizing high-temperature, continuous-vitrification technology for the immobilization of the residual-liquid waste, (2) test the airlift draining technique utilized by the high-temperature melter, (3) compare glass produced in this process to residual-liquid glass produced under laboratory conditions, (4) investigate cesium volatility from the melter during waste processing, and (5) determine the maximum residual-liquid glass production rate in the high-temperature melter. The three tests with the residual-liquid composition confirmed the viability of the continuous-melting vitrification technique for the immobilization of this waste. The airlift draining technique was demonstrated in these tests and the glass produced from the melter was shown to be less porous than the laboratory-produced glass. The final glass produced from the second test was compared to a glass of the same composition produced under laboratory conditions. The comparative tests found the glasses to be indistinguishable, as the small differences in the test results fell within the precision range of the characterization testing equipment. The cesium volatility was examined in the final test. This examination showed that 0.44 wt % of the cesium (assumed to be cesium oxide) was volatilized, which translates to a volatilization rate of 115 mg/cm^2-h.

223 (PNL--3372) IMMOBILIZATION OF HIGH-LEVEL DEFENSE WASTE IN A SLURRY-FED ELECTRIC GLASS MELTER. Brouns, R.A.; Mellinger, G.B.; Nelson, T.A.; Oma, K.H. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Nov 1980. Contract AC06-76RL01830. 109p. NTIS, PC A06/MF A01.

Scoping studies have been performed at the Pacific Northwest Laboratory related to the direct liquid-feeding of a generic high-level defense waste to a joule-heated ceramic melter. Tests beginning on the laboratory scale and progressing to full-scale operation are reported. Laboratory work identified the need for a reducing agent in the feed to help control the foaming tendencies of the waste glass. These tests also indicated that suspension agents were helpful in reducing the tendency of solids to settle out of the liquid feed. Testing was then moved to a larger pilot-scale melter (designed for approx. 2.5 kg/h) where verification of the flowsheet examined in the lab was accomplished. It was found that the reducing agent controlled foaming and did not result in the precipitation of metals. Pumping problems were encountered when slurries with higher than normal solids content were fed. A demonstration (designed for approx. 50 kg/h) in a full-scale melter was then made with the tested flowsheet; however, the amount of reducing agent had to be increased. In addition, it was found that feed control needed further development; however, steady-state operation was achieved giving encouraging results on process capacities. During steady-state operation, ruthenium losses to the offgas system averaged less than 0.16%, while cesium losses were somewhat higher, ranging from 0.91 to 24% and averaging 13%. Particulate decontamination factors from feed to offgas in the melter ranged from 5 x 10^2 to greater than 10^3 without any filtration or treatment. Approximately 1050 kg of glass was produced from 2900 L of waste at rates up to 40 kg/h.

224 (PNL--3375) ENGINEERING-SCALE VITRIFICATION OF COMMERCIAL HIGH-LEVEL WASTE. Bonner, W.F.; Bjorklund, W.J.; Hanson, M.S.; Knowlton, D.E. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Apr 1980. Contract AC06-76RL01830. 40p. NTIS, PC A03/MF A01.

To date, technology for immobilizing commercial high-level waste (HLW) has been extensively developed, and two major demonstration projects have been completed, the Waste Solidification Engineering Prototypes (WSEP) Program and the Nuclear Waste Vitrification Project (NWVP). The feasibility of radioactive waste solidification was demonstrated in the WSEP program between 1966 and 1970 (McElroy et al. 1972) using simulated power-reactor waste composed of nonradioactive chemicals and HLW from spent, Hanford reactor fuel. Thirty-three engineering-scale canisters of solidified HLW were produced during the operations. In early 79, the NWVP demonstrated the vitrification of HLW from the processing of actual commercial nuclear fuel. This program consisted of two parts, (1) waste preparation and (2) vitrification by spray calcination and in-can melting. This report presents results from the NWVP.

225 (PNL--3387) DESIGN AND PERFORMANCE OF A 100-KG/H, DIRECT CALCINE-FED ELECTRIC-MELTER SYSTEM FOR NUCLEAR-WASTE VITRIFICATION. Dierks, R.D. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Nov 1980. Contract AC06-76RL01830. 70p. NTIS, PC A04/MF A01.

This report describes the physical characteristics of a ceramic-lined, joule-heated glass melter that is directly connected to the discharge of a spray calciner and is currently being used to study the vitrification of simulated nuclear-waste slurries. Melter performance characteristics and subsequent design improvements are described. The melter contains 0.24 m^3 of glass with a glass surface area of 0.76 m^2, and is heated by the flow of an alternating current (ranging from 600 to 1200 amps) between two Inconel-690 slab-type electrodes immersed in the glass at either end of the melter tank. The melter was maintained at operating temperature (900 to 1260°C) for 15 months, and produced 62,000 kg of glass. The maximum sustained operating period was 122 h, during which glass was produced at the rate of 70 kg/h.

226 (PNL--3405) IN-CAN MELTING DEMONSTRATION OF WASTES FROM THE IDAHO CHEMICAL PROCESSING PLANT. Bjorklund, W.J.; Chick, L.A.; Hollis, H.H.; Mellinger, G.B.; Nelson, T.A.; Petkus, L.L. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jul 1980. Contract AC06-76RL01830. 58p. NTIS, PC A04/MF A01.

The immobilization of Idaho Chemical Processing Plant (ICPP) zirconia calcine using Idaho glass composition (ICPP-127) was evaluated at Pacific Northwest Laboratory (PNL) in two engineering-scale in-can melter tests. The glass was initially characterized in the laboratory to verify processing parameters. Glass was then produced in a pilot-scale melter and then in a full-scale melter to evaluate the processing and the resultant product. Potential corrosion problems were identified with the glass and some processing problems were encountered, but neither is insurmountable. The product is a durable leach-resistant glass. The glass appears to be nonhomogeneous, but chemically it is quite uniform.

227 (PNL--3406) INVESTIGATION OF CORROSION EXPERIENCED IN A SPRAY CALCINER/CERAMIC MELTER VITRIFICATION SYSTEM. Dierks, R.D.; Mellinger, G.B.; Miller, F.A.; Nelson, T.A.; Bjorklund, W.J. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Aug 1980. Contract AC06-76RL01830. 173p. NTIS, PC A08/MF A01.

After periodic testing of a large-scale spray calciner/ceramic melter vitrification system over a 2-yr period, sufficient corrosion was noted on various parts of the vitrification system to warrant its disassembly and inspection. A majority of the 316 SS sintered metal filters on the spray calciner were damaged by chemical corrosion and/or high temperature oxidation. Inconel-601 portions of the melter lid were attacked by chlorides and sulfates which volatilized from the molten glass. The refractory blocks, making up the walls of the melter, were attacked by the waste glass. This attack was occurring when operating temperatures were >1200°C. The melter floor was protected by a sludge layer and showed no corrosion. Corrosion to the Inconel-690 electrodes was minimal, and no corrosion was noted in the offgas treatment system downstream of the sintered metal filters. It is believed that most of the melter corrosion occurred during one specific operating period when the melter was operated at high temperatures in an attempt to overcome glass foaming behavior. These high temperatures resulted in a significant release of volatile elements from the molten glass, and also created a situation where the glass was very fluid and convective, which increased the corrosion rate of the refractories. Specific corrosion to the calciner components cannot be proven to have occurred during a specific time period, but the mechanisms of attack were all accelerated under the high-temperature conditions that were experienced with the melter. A review of the materials of construction has been made, and it is concluded that with controlled operating conditions and better protection of some materials of construction corrosion of these systems will not cause problems. Other melter systems operating under similar strenuous conditions have shown a service life of 3 yr.

228 (PNL--3465) ANNUAL REPORT ON THE DEVELOPMENT AND CHARACTERIZATION OF SOLIDIFIED FORMS FOR NUCLEAR WASTES, 1979. Chick, L.A.; McVay, G.L.; Mellinger, G.B.; Roberts, F.P. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Dec 1980. Contract AC06-76RL01830. 90p. NTIS, PC A05/MF A01.

Development and characterization of solidified nuclear waste forms is a major continuing effort at Pacific Northwest Laboratory. Contributions from seven programs directed at understanding chemical composition, process conditions, and long-term behaviors of various nuclear waste forms are included in this report. The major findings of the report are included in extended figure captions that can be read as brief technical summaries of the research, with additional information included in a traditional narrative format. Waste form development proceeded on crystalline and glass materials for high-level and transuranic (TRU) wastes. Leaching studies emphasized new areas of research aimed at more basic understanding of waste form/aqueous solution interactions. Phase behavior and thermal effects research included studies on crystal phases in defense and TRU waste glasses and on liquid-liquid phase separation in borosilicate waste glasses. Radiation damage effects in crystals and glasses from alpha decay and from transmutation are reported.

229 (PNL--3477) PRELIMINARY EVALUATION OF ALTERNATIVE WASTE FORM SOLIDIFICATION PROCESSES. VOLUME II. EVALUATION OF THE PROCESSES. (Johnson (E.R.) Associates, Inc., Reston, VA (USA)). Aug 1980. Contract AC06-76RL01830. 222p. NTIS, PC A10/MF A01.

This Volume II presents engineering feasibility evaluations of the eleven processes for solidification of nuclear high-level liquid wastes (HHLW) described in Volume I of this report. Each evaluation was based in a systematic assessment of the process in respect to six principal evaluation criteria: complexity of process; state of development; safety; process requirements; development work required; and facility requirements. The principal criteria were further subdivided into a total of 22 subcriteria, each of which was assigned a weight. Each process was then assigned a figure of merit, on a scale of 1 to 10, for each of the subcriteria. A total rating was obtained for each process by summing the products of the subcriteria ratings and the subcriteria weights. The evaluations were based on the process descriptions presented in Volume I of this report, supplemented by information obtained from the literature, including publications by the originators of the various processes. Waste form properties were, in general, not evaluated. This document describes the approach which was taken, the developent and application of the rating criteria and subcriteria, and the evaluation results. A series of appendices set forth summary descriptions of the processes and the ratings, together with the complete numerical ratings assigned; two appendices present further technical details on the rating process.

230 (PNL--3479) DESCRIPTION OF PROCESSES FOR THE IMMOBILIZATION OF SELECTED TRANSURANIC WASTES. Timmerman, C.L. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Dec 1980. Contract AC06-76RL01830. 42p. NTIS, PC A03/MF A01.

Processed sludge and incinerator-ash wastes contaminated with transuranic (TRU) elements may require immobilization to prevent the release of these elements to the environment. As part of the TRU Waste Immobilization Program sponsored by the Department of Energy (DOE), the Pacific Northwest Laboratory is developing applicable waste-form and processing technology that may meet this need. This report defines and describes processes that are capable of immobilizing a selected TRU waste-stream consisting of a blend of three parts process sludge and one part incinerator ash. These selected waste streams are based on the compositions and generation rates of the waste processing and incineration facility at the Rocky Flats Plant. The specific waste forms that could be produced by the described processes include: in-can melted borosilicate-glass monolith; joule-heated melter borosilicate-glass monolith or marble; joule-heated melter aluminosilicate-glass monolith or marble; joule-heated melter basaltic-glass monolith or marble; joule-heated melter glass-ceramic monolith; cast-cement monolith; pressed-cement pellet; and cold-pressed sintered-ceramic pellet.

231 (PNL--3493) SAFETY ASSESSMENT OF THE LIQUID-FED CERAMIC MELTER PROCESS. Buelt, J.L.; Partain, W.L. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Aug 1980. Contract AC06-76RL01830. 113p. NTIS, PC A06/MF A01.

As part of its development program for the solidification of high-level nuclear waste, Pacific Northwest Laboratory assessed the safety issues for a complete liquid-fed ceramic melter (LFCM) process. The LFCM process, an adaption of commercial glass-making technology, is being developed to convert high-level liquid waste from the nuclear fuel cycle into glass. This safety assessment uncovered no unresolved or significant safety problems with the LFCM process. Although in this assessment the LFCM process was not directly compared with other solidification processes, the safety hazards of the LFCM process are comparable to those of other processes. The high processing temperatures of the glass in the LFCM pose no additional significant safety concerns, and the dispersible inventory of dried waste (calcine) is small. This safety assessment was based on the nuclear power waste flowsheet, since power waste is more radioactive than defense waste at the time of solidification, and all accident conditions for the power waste would have greater radiological consequences then those for defense waste. An exhaustive list of possible off-standard conditions and equipment failures was compiled. These accidents were then classified according to severity of consequence and type of accident. Radionuclide releases to the stack were calculated for each group of accidents using conservative assumptions regarding the retention and decontamination features of the process and facility. Two recommendations that should be considered by process designers are given in the safety assessment.

232 (PNL--3495) SPRAY CALCINER/IN-CAN MELTER HIGH-LEVEL WASTE SOLIDIFICATION TECHNICAL MANUAL. Larson, D.E. (ed.). (Battelle Pacific Northwest Labs., Richland, WA (USA)). Sep 1980. Contract AC06-76RL01830. 761p. NTIS, PC A99/MF A01.

This technical manual summarizes process and equipment technology developed at Pacific Northwest Laboratory over the last 20 years for vitrification of high-level liquid waste by the Spray Calciner/In-Can Melter process. Pacific Northwest Laboratory experience includes process development and demonstration in laboratory-, pilot-, and full-scale equipment using nonradioactive synthetic wastes. Also, laboratory- and pilot-scale process demonstrations have been conducted using actual high-level radioactive wastes. In the course of process development, more than 26 tonnes of borosilicate glass have been produced in 75 canisters. Four of these canisters contained radioactive waste glass. The associated process and glass chemistry is discussed. Technology areas described include calciner feed treatment and techniques, calcination, vitrification, off-gas treatment, glass containment (the canister), and waste glass chemistry. Areas of optimization and site-specific development that would be needed to adapt this base technology for specific plant application are indicated. A conceptual Spray Calciner/In-Can Melter system design and analyses are provided in the manual to assist prospective users in evaluating the process for plant application, to provide equipment design information, and to supply information for safety analyses and environmental reports. The base (generic) technology for the Spray Calciner/In-Can Melter process has been developed to a point at which it is ready for plant application.

233 (PNL--3505) NATURALLY OCCURRING CRYSTALLINE PHASES: ANALOGUES FOR RADIOACTIVE WASTE FORMS. Haaker, R.F.; Ewing, R.C. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jan 1981. Contract AC06-76RL01830. 284p. NTIS, PC A13/MF A01.

Naturally occurring mineral analogues to crystalline phases that are constituents of crystalline radioactive waste forms provide a basis for comparison by which the long-term stability of these phases may be estimated. The crystal structures and the crystal chemistry of the following natural analogues are presented: baddeleyite, hematite, nepheline; pollucite, scheelite;sodalite, spinel, apatite, monazite, uraninite, hollandite-priderite, perovskite, and zirconolite. For each phase in geochemistry, occurrence, alteration and radiation effects are described. A selected bibliography for each phase is included.

234 (PNL--3512) SURVEY OF MATRIX MATERIALS FOR SOLIDIFIED RADIOACTIVE HIGH-LEVEL WASTE. Gurwell, W.E. (Pacific Northwest Lab., Richland, WA (USA)). Sep 1981. Contract AC06-76RL01830. 58p. NTIS, PC A04/MF A01. Order Number DE82000708.

Pacific Northwest Laboratory (PNL) has been investigating advanced waste forms, including matrix waste forms, that may provide a very high degree of stability under the most severe repository conditions. The purpose of this study was to recommend practical matrix materials for future development that most enhance the stability of the matrix waste forms. The functions of the matrix were reviewed. Desirable matrix material properties were discussed and listed relative to the matrix functions. Potential matrix materials were discussed and recommendations were made for future matrix development. The matrix mechanically contains waste cores, reduces waste form temperatures, and is capable of providing a high-quality barrier to leach waters. High-quality barrier matrices that separate and individually encapsulate the waste cores are fabricated by powder fabrication methods, such as sintering, hot pressing, and hot isostatic pressing. Viable barrier materials are impermeable, extremely corrosion resistent, and mechanically strong. Three material classes potentially satisfy the requirements for a barrier matrix and are recommended for development: titanium, glass, and graphite. Polymers appear to be marginally adequate, and a more thorough engineering assessment of their potential should be made.

235 (PNL--3516) COMPARATIVE WASTE FORMS STUDY. Wald, J.W.; Lokken, R.O.; Shade, J.W.; Rusin, J.M. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Dec 1980. Contract AC06-76RL01830. 59p. NTIS, PC A04/MF A01.

A number of alternative process and waste form options exist for the immobilization of nuclear wastes. Although data exists on the characterization of these alternative waste forms, a straightforward comparison of product properties is difficult, due to the lack of standardized testing procedures. The characterization study described in this report involved the application of the same volatility, mechanical strength and leach tests to ten alternative waste forms, to assess product durability. Bulk property, phase analysis and microstructural examination of the simulated products, whose waste loading varied from 5% to 100% was also conducted. The specific waste forms investigated were as follows: Cold Pressed and Sintered PW-9 Calcine; Hot Pressed PW-9 Calcine; Hot Isostatic Pressed PW-9 Calcine; Cold Pressed and Sintered SPC-5B Supercalcine; Hot Isostatic pressed SPC-5B Supercalcine; Sintered PW-9 and 50% Glass Frit; Glass 76-68; Celsian Glass Ceramic; Type II Portland Cement and 10% PW-9 Calcine; and Type II Portland Cement and 10% SPC-5B Supercalcine. Bulk property data were used to calculate and compare the relative quantities of waste form volume produced at a spent fuel processing rate of 5 metric ton uranium/day. This quantity ranged from 3173 L/day (5280 Kg/day) for 10% SPC-5B supercalcine in cement to 83 L/day (294 Kg/day) for 100% calcine. Mechanical strength, volatility, and leach resistance tests provide data related to waste form durability. Glass, glass-ceramic and supercalcine ranked high in waste form durability where as the 100% PW-9 calcine ranked low. All other materials ranked between these two groupings.

236 (PNL--3550) EVALUATION OF DEFENSE-WASTE GLASS PRODUCED BY FULL-SCALE VITRIFICATION EQUIPMENT. Lukacs, J.M.; Petkus, L.L.; Mellinger, G.B. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Sep 1981. Contract AC06-76RL01830. 102p. NTIS, PC A06/MF A01. Order Number DE81030408.

Three full-scale vitrification processes at the Pacific Northwest Laboratory produced over 67,000 kg of simulated nuclear-waste glass from March 1979 to August 1980. Samples were analyzed to monitor process operation and evaluate the resulting glass product. These processes are: Spray Calciner/In-Can Melter (SC/ICM); Spray Calciner/Calcine-Fed Ceramic Melter (SC/CFCM); and Liquid-Fed Ceramic Melter (LFCM). Waste components in the process feed varied less than +- 10%. The SC/ICM and SC/CFCM which use separate waste and frit feed systems showed larger glass compositional variation than the LFCM, which processed only premixed feed during this period. The SC/ICM and SC/CFCM product contained significant amounts of acmite crystals, while the LFCM product was largely amorphous. In addition, the lower portion of all SC/ICM-filled canisters contained a zone rich in waste components. A product chemical durability as determined by pH4 and soxhlet leach tests varied considerably. Aside from increased durability under pH4 conditions with decreasing waste content, glass composition, microstructure and melting process did not correlate with glass durability. For all samples analyzed, the weight loss under pH4 conditions ranged from 17.7 to 85.2 wt %. Soxhlet conditions produced weight losses from 1.78 to 3.56 wt %.

237 (PNL--3552) INVESTIGATION OF FOAMING DURING NUCLEAR DEFENSE-WASTE SOLIDIFICATION BY ELECTRIC MELTING. Blair, H.T.; Lukacs, J.M. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Dec 1980. Contract AC06-76RL01830. 66p. NTIS, PC A04/MF A01.

To determine the cause of foaming, the physical and chemical composition of the glass formers that are added to the waste to produce a borosilicate melt were investigated. It was determined that the glass-forming frit was not the source of the foam-causing gases. Incomplete calcination of the waste, which results in residual hydrates, carbonates and nitrates, and the relatively high carbon and sulfate contents of the waste glass composition were also eliminated as possible sources of the foam. It was finally shown that the oxides of the multivalent ions of manganese and iron that are in the defense waste in high concentrations are the source of the foaming. Nickel oxide is also present in the waste and is suspected of contributing to the foaming. In investigating methods to reduce the foam, the focus was on the chemistry of the materials being processed rather than on the mechanical aspects of the processing equipment to avoid increasing the mechanical complexity of the melter operation. Reducing the waste loading in the host glass from 28 to 14 wt. % produced the most significant reduction in the foam. Of course this did not increase the rate at which waste can be processed. Adding carbonaceous additives or barium metaphosphate to the waste/frit mixture (batch) reduced the foaming somewhat. However, if too much reducing agent was added to the batch, iron-nickel alloys separated from the melt. Likewise, melting the batch in an inert or a reducing atmosphere reduced the foaming but produced a heterogeneous product. Finally, initial attempts to control foaming by adding reducing agents to the liquid waste and then spray-calcining it using an inert atomizing gas were not successful. The possibilities for liquid-waste treatment need to be investigated further.

238 (PNL--3554) STARTUP OF THE REMOTE LABORATORY-SCALE WASTE-TREATMENT FACILITY. Knox, C.A.; Siemens, D.H.; Berger, D.N. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jan 1981. Contract AC06-76RL01830. 27p. NTIS, PC A03/MF A01.

The Remote Laboratory-Scale Waste-Treatment Facility was designed as a system to solidify small volumes of radioactive liquid wastes. The objectives in operating this facility are to evaluate solidification processes, determine the effluents generated, test methods for decontaminating the effluents, and provide radioactive solidified waste products for evaluation. The facility consists of a feed-preparation module, a waste-solidification module and an effluent-treatment module. The system was designed for remote installation and operation. Several special features for remotely handling radioactive materials were incorporated into the design. The equipment was initially assembled outside of a radiochemical cell to size and fabricate the connecting jumpers between the modules and to complete some preliminary design-verification tests. The equipment was then disassembled and installed in the radiochemical cell. When installation was completed the entire system was checked out with water and then with a nonradioactive simulated waste solution. The purpose of these operations was to start up the facility, find and solve operational problems, verify operating procedures and train personnel. The major problems experienced during these nonradioactive runs were plugging of the spray calciner nozzle and feed tank pumping failures. When these problems were solved, radioactive operations were started. This report describes the installation of this facility, its special remote design feature and the startup operations.

239 (PNL--3608-1) WEST VALLEY DEMONSTRATION PROJECT: ALTERNATIVE PROCESSES FOR SOLIDIFYING THE HIGH-LEVEL WASTES. Holton, L.K.; Larson, D.E.; Partain, W.L.; Treat, R.L. (Pacific Northwest Lab., Richland, WA (USA)). Oct 1981. Contract AC06-76RL01830. 303p. NTIS, PC A14/MF A01. Order Number DE82000936.

In 1980, the US Department of Energy (DOE) established the West Valley Solidification Project as the result of legislation passed by the US Congress. The purpose of this project was to carry out a high level nuclear waste management demonstration project at the Western New York Nuclear Service Center in West Valley, New York. The DOE authorized the Pacific Northwest Laboratory (PNL), which is operated by Battelle Memorial Institute, to assess alternative processes for treatment and solidification of the WNYNSC high-level wastes. The Process Alternatives Study is the suject of this report. Two pretreatment approaches and several waste form processes were selected for evaluation in this study. The two waste treatment approaches were the salt/sludge separation process and the combined waste process. Both terminal and interim waste form processes were studied.

240 (PNL--3617) SOLID STATE STORAGE OF RADIOACTIVE KRYPTON IN A SILICA MATRIX.

Tingey, G.L.; Lytle, J.M.; Gray, W.J.; Wheeler, K.R. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Dec 1980. Contract AC06-76RL01830. 25p. NTIS, PC A02/MF A01.

The feasibility of loading a low density SiO_2 glass with krypton for storage of radioactive ^{85}Kr has been demonstrated by studies using non-radioactive krypton. A 96% SiO_2 glass with 28% porosity was heated at an elevated pressure of Kr gas to a temperature of 850 to 900°C and held at that temperature to sinter the glass-krypton composite to a density of about 2 g/cm³. A krypton content of 30 cm³ of Kr(STP)/cm³ of glass has been demonstrated when loading pressures of 140 MPa are used. Krypton release rates from the glass are lower than reported for any other waste form considered currently. At 420°C a diffusion parameter, D/r_0^2, of 8.66×10^{-13} min^{-1} was determined which leads to a total release of 0.7% of the krypton in 10 years. Release rates increase moderately with increasing temperature up to 600°C and increase rapidly above 600°C. The lower loading pressures (about 40 MPa) may appear to yield a more favorable product from the point of view of krypton release than the high pressures. Advantages and disadvantages of the technique are given in the conclusions section.

241 (PNL--3702) DESIGN AND PERFORMANCE OF ATOMIZING NOZZLES FOR SPRAY CALCINATION OF HIGH-LEVEL WASTES. Miller, F.A.; Stout, L.A. (Battelle Pacific Northwest Labs., Richland, WA (USA)). May 1981. Contract AC06-76RL01830. 52p. NTIS, PC A04/MF A01.

A key aspect of high-level liquid-waste spray calcination is waste-feed atomization by using air atomizing nozzles. Atomization substantially increases the heat transfer area of the waste solution, which enhances rapid drying. Experience from the spray-calciner operations has demonstrated that nozzle flow conditions that produce 70-μ median-volume-diameter or smaller spray droplets are required for small-scale spray calciners (drying capacity less than 80 L/h). For large-scale calciners (drying capacity greater than 300 L/h), nozzle flow conditions that produce 100-μ median-volume-diameter or smaller spray droplets are required. Mass flow ratios of 0.2 to 0.4, depending on nozzle size, are required for proper operation of internal-mix atomizing nozzles. Both internal-mix and external-mix nozzles have been tested at PNL. Due to the lower airflow requirements and fewer large droplets produced, the internal-mix nozzle has been chosen for primary development in the spray calciner program at PNL. Several nozzle air-cap materials for internal-mix nozzles have been tested for wear resistance. Results show that nozzle air caps of stainless steel and Cer-vit (a machineable glass ceramic) are suceptible to rapid wear by abrasive slurries, whereas air caps of alumina and reaction-bonded silicon nitride show only slow wear. Longer-term testing is necessary to determine more accurately the actual frequency of nozzle replacement. Atomizing nozzle air caps of alumina are subject to fracture from thermal shock, whereas air caps of silicon nitride and Cer-vit are not. Fractured nozzles are held in place by the air-cap retaining ring and continue to atomize satisfactorily. Therefore, fractures caused by thermal shocking do not necessarily result in nozzle failure.

242 (PNL--3741) BEHAVIOR OF RADIOACTIVE IODINE AND TECHNETIUM IN THE SPRAY CALCINATION OF HIGH-LEVEL WASTE. Knox, C.A.; Farnsworth, R.K. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Aug 1981. Contract AC06-76RL01830. 22p. NTIS, PC A02/MF A01. Order Number DE81028405.

The Remote Laboratory-Scale Waste Treatment Facility (RLSWTF) was designed and built as a part of the High-Level Waste Immobilization Program (now the High-Level Waste Process Development Program) at the Pacific Northwest Laboratory. In this facility, which is installed in a radiochemical cell, small volumes of radioactive liquid wastes can be solidified, the process off gas can be analyzed, and the methods for decontaminating this off gas can be tested. Initial operations were completed with nonradioactive, simulated waste solutions (Knox, Siemens and Berger 1981). The first radioactive operations in this facility were performed with a simulated, commercial waste composition containing tracer levels of ^{99}Tc and ^{131}I. This report describes the facility and test operations and presents the results of the behavior of ^{131}I and ^{99}Tc during solidification of radioactive liquid wastes. During the spray calcination of commercial high-level liquid waste spiked with ^{99}Tc and ^{131}I, there was a 0.3 wt% loss of particulates, a 0.15 wt% loss of ^{99}Tc and a 31 wt% loss of ^{131}I past the sintered-metal filters. These filters and a venturi scrubber were very efficient in removing particulates and ^{99}Tc from the off-gas stream. Liquid scrubbers were not efficient in removing ^{131}I, as 25% of the total lost went to the building off-gas system. Therefore, solid adsorbents will be needed to remove iodine. For all future RLSWTF operations where iodine is present, a silver zeolite adsorber will be used.

243 (PNL--3742) DESIGN AND PERFORMANCE OF A FULL-SCALE SPRAY CALCINER FOR NONRADIOACTIVE HIGH-LEVEL-WASTE-VITRIFICATION STUDIES. Miller, F.A. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jun 1981. Contract AC06-76RL01830. 243p. NTIS, PC A11/MF A01. Order Number DE81024477.

In the spray calcination process, liquid waste is spray-dried in a heated-wall spray dryer (termed a spray calciner), and then it may be combined in solid form with a glass-forming frit. This mixture is then melted in a continuous ceramic melter or in an in-can melter. Several sizes of spray calciners have been tested at PNL- laboratory scale, pilot scale and full scale. Summarized here is the experience gained during the operation of PNL's full-scale spray calciner, which has solidified approx. 38,000 L of simulated acid wastes and approx. 352,000 L of simulated neutralized wastes in 1830 h of processing time. Operating principles, operating experience, design aspects, and system descriptions of a full-scale spray calciner are discussed. Individual test run summaries are given in Appendix A. Appendices B and C are studies made by Bechtel Inc., under contract by PNL. These studies concern, respectively, feed systems for the spray calciner process and a spray calciner vibration analysis. Appendix D is a detailed structural analysis made at PNL of the spray calciner. These appendices are included in the report to provide a complete description of the spray calciner and to include all major studies made concerning PNL's full-scale spray calciner.

244 (PNL--3750) NUCLEAR-WASTE ENCAPSULATION BY METAL-MATRIX CASTING. Nelson, R.G.; Nesbitt, J.F.; Slate, S.C. (Battelle Pacific Northwest Labs., Richland, WA (USA)). May 1981. Contract AC06-76RL01830. 60p. NTIS, PC A04/MF A01.

Several encapsulation casting processes are described that were developed or used at the

Pacific Northwest Laboratory to embed simulated high-level wastes of two different forms (glass marbles and ceramic pellets) in metal matrices. Preliminary evaluations of these casting processes and the products are presented. Demonstrations have shown that 5- to 10-mm-dia glass marbles can be encapsulated on an engineering scale with lead or lead alloys by gravity or vacuum processes. Marbles approx. 12 mm in dia were successfully encapsulated in a lead alloy on a production scale. Also, 4- to 9-mm-dia ceramic pellets in containers of various sizes were completely penetrated and the individual pellets encased with aluminum-12 wt % silicon alloy by vacuum processes. Indications are that of the casting processes tested, aluminum 12 wt % silicon alloy vacuum-cast around ceramic pellets had the highest degree of infiltration or coverage of pellet surfaces.

245 (PNL--3776) CHARACTERIZATION OF TRANSURANIUM: CONTAMINATED SOLID WASTES RESIDUES. Bryan, G.H. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Apr 1981. Contract AC06-76RL01830. 30p. NTIS, PC A03/MF A01.

The selection of technologies for the immobilization of transuranic wastes will depend in part upon the chemical and physical nature of the wastes. Characterization of some of the more important defense transuranic wastes is necessary to develop and evaluate alternative immobilization systems that might be applied to selected defense transuranic wastes. Process sludge and incinerator ash were selected for detailed study because of their importance in the transuranic waste inventory and because of proposed waste acceptance criteria that would exclude sludge and ash as-generated. By experimentally determining the variations in the chemical and physical properties of these wastes, it will be possible to develop and characterize immobilization products (and processes) and to examine their sensitivity to variations in the waste. Results indicate that there can be rather large differences in the composition of the two wastes. In the case of incinerator residues, large variations in the chemical composition may be expected if the feed is not homogenized to some extent. With sludge, analyses indicated variations in some metallic constituents at a given site and significant variations between sites.

246 (PNL--3802) STATE-OF-THE-ART REVIEW OF MATERIALS PROPERTIES OF NUCLEAR WASTE FORMS. Mendel, J.E.; Nelson, R.D.; Turcotte, R.P.; Gray, W.J.; Merz, M.D.; Roberts, F.P.; Weber, W.J.; Westsik, J.H. Jr.; Clark, D.E. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Apr 1981. Contract AC06-76RL01830. 210p. NTIS, PC A10/MF A01.

The Materials Characterization Center (MCC) was established at the Pacific Northwest Laboratory to assemble a standardized nuclear waste materials data base for use in research, systems and facility design, safety analyses, and waste management decisions. This centralized data base will be provided through the means of a Nuclear Waste Materials Handbook. The first issue of the Handbook will be published in the fall of 1981 in looseleaf format so that it can be updated as additional information becomes available. To ensure utmost reliability, all materials data appearing in the Handbook will be obtained by standard procedures defined in the Handbook and approved by an independent Materials Review Board (MRB) comprised of materials experts from Department of Energy laboratories and from universities and industry. In the interim before publication of the Handbook there is need for a report summarizing the existing materials data on nuclear waste forms. This review summarizes materials property data for the nuclear waste forms that are being developed for immobilization of high-level radioactive waste. It is intended to be a good representation of the knowledge concerning the properties of HLW forms as of March 1981. The table of contents lists the following topics: introduction which covers waste-form categories, and important waste-form materials properties; physical properties; mechanical properties; chemical durability; vaporization; radiation effects; and thermal phase stability.

247 (PNL--3818) DEFENSE WASTE VITRIFICATION STUDIES DURING FY 1980. Bjorklund, W.J. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Aug 1981. Contract AC06-76RL01830. 91p. NTIS, PC A05/MF A01. Order Number DE81029616.

During FY-1980, Pacific Northwest Laboratory (PNL) tested three vitrification processes on simulated high-level radioactive waste typical of that stored or being produced at US defense facilities. Processes tested included a spray calciner/in-can melter, spray calciner/ceramic melter and direct liquid feeding of a ceramic melter. Tests were made on pilot-scale as well as fullscale equipment. Over 16,000 kg of glass product were produced from 68,000 L of simulated waste. Several compositions were tested, and the glass products were evaluated. Emphasis was placed on determining the processing rates and the ability of the waste to be processed. Off-gas data were collected on several runs. Major conclusions drawn from this test program are divided into processing results, glass-product results, and general information.

248 (PNL--3832) MATERIALS CHARACTERIZATION CENTER WORKSHOP ON COMPOSITIONAL AND MICROSTRUCTURAL ANALYSIS OF NUCLEAR WASTE MATERIALS. SUMMARY REPORT. Daniel, J.L.; Strachan, D.M.; Shade, J.W.; Thomas, M.T. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jun 1981. Contract AC06-76RL01830. 87p. NTIS, PC A05/MF A01. Order Number DE81027029.

The purpose of the Workshop on Compositional and Microstructural Analysis of Nuclear Waste Materials, conducted November 11 and 12, 1980, was to critically examine and evaluate the various methods currently used to study non-radioactive, simulated, nuclear waste-form performance. Workshop participants recognized that most of the Materials Characterization Center (MCC) test data for inclusion in the Nuclear Waste Materials Handbook will result from application of appropriate analytical procedures to waste-package materials or to the products of performance tests. Therefore, the analytical methods must be reliable and of known accuracy and precision, and results must be directly comparable with those from other laboratories and from other nuclear waste materials. The 41 participants representing 18 laboratories in the United States and Canada were organized into three working groups: Analysis of Liquids and Solutions, Quantitative Analysis of Solids, and Phase and Microstructure Analysis. Each group identified the analytical methods favored by their respective laboratories, discussed areas needing attention, listed standards and reference materials currently used, and recommended means of verifying interlaboratory comparability of data. The major conclusions from this workshop are presented.

249 (PNL--3838) REFERENCE COMMERCIAL HIGH-LEVEL WASTE GLASS AND CANISTER DEFINITION. Slate, S.C.; Ross, W.A.; Partain, W.L. (Pacific Northwest Lab., Richland, WA (USA)). Sep 1981. Contract AC06-76RL01830. 103p. NTIS, PC E05/MF A01. Order Number DE82001503.

Includes 1 sheet of 48x reduction microfiche.

This report presents technical data and performance characteristics of a high-level waste glass and canister intended for use in the design of a complete waste encapsulation package suitable for disposal in a geologic repository. The borosilicate glass contained in the stainless steel canister represents the probable type of high-level waste product that will be produced in a commercial nuclear-fuel reprocessing plant. Development history is summarized for high-level liquid waste compositions, waste glass composition and characteristics, and canister design. The decay histories of the fission products and actinides (plus daughters) calculated by the ORIGEN-II code are presented.

250 (PNL--3902) STABILITY OF MULTILAYER EARTHEN BARRIERS USED TO ISOLATE MILL TAILINGS: GEOLOGIC AND GEOTECHNOLOGICAL CONSIDERATIONS. Zellmer, J.T. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Aug 1981. Contract AC06-76RL01830. 49p. NTIS, PC A03/MF A01. Order Number DE81028403.

This report briefly discusses how seismic activity, erosion, climatic change, slope stability, differential settlement, and cover design could affect the long-term integrity of multilayer earthen cover systems. In addition, the report suggests ways to design and construct covers so that adverse impacts can be avoided or minimized. The stability of multilayer earthen barriers used to isolate uranium mill tailings depends on the morphology of the disposal site, the engineering of the barrier, the condition of the tailings, and the possible impacts of earthquakes, erosion, and climatic changes. When designing a cover for or siting a tailings pile, one must take into account both geologic and geotechnological variables. To alleviate the adverse effects of possible seismic activity, tailings piles should never be located on or near active or capable faults. Existing piles near faults should be moved to safer sites or engineered to withstand possible displacement and shaking. Liquefaction generally can be prevented if the tailings and their underlying material are compacted to a relative density of 60% or greater, or if they are kept dry. If the tailings are saturated, dewatering schemes may have to be used. Erosion may be caused by streams, glaciers, or wind, depending on the geomorphic and atmospheric conditions at the site. Fluvial erosion can be prevented by using dikes (and avoided by initially siting the pile a safe distance away from stream courses). In some cases, fluvial waters or rainfall may have to be rerouted over the pile via armored ditches. Eolian erosion can be minimized by vegetating the disposal site or covering it with gravel. Because of the wide geomorphic and geotechnological variations extant at most sites, a single cover design cannot be cost-effectively and efficiently used at all sites.

251 (PNL--3943) SECTIONING OF CONTAMINATED COMPONENTS FOR DECONTAMINATION BY VIBRATORY FINISHING AND ELECTROPOLISHING. Fetrow, L.K.; Allen, R.P. (Pacific Northwest Lab., Richland, WA (USA)). Sep 1981. Contract AC06-76RL01830. 43p. NTIS, PC A03/MF A01. Order Number DE82001869.

This report summarizes work conducted to develop, adapt, and evaluate a variety of techniques for sectioning glove boxes, chemical processing equipment, pipes, ducts, and other contaminated components in preparation for decontamination by vibratory finishing and electropolishing. These sectioning studies were conducted with a special 10-ft x 20-ft x 10-ft stainless-steel, walk-in glove box equipped for either hands-on operation via gloves and personnel entry, or remote operation using master slave manipulators and a bridge crane. Several sectioning techniques have been evaluated with respect to effectiveness, versatility, secondary waste generation, and capability for remote operation. The methods include wet and dry plasma arc torch cutting, mechanical sawing and nibbling, abrasive cutting, and hydraulic sheering and punching. The results of these comparison studies show that the plasma arc torch is a very rapid and effective metal cutting tool for size reduction applications. However, its use to prepare material for decontamination should be minimized because of problems with smoke generation, torch manipulation, waste generation, and entrainment of contamination. Mechanical saws eliminate all but the waste generation problem, but are very slow and labor intensive. Mechanical nibblers are fast and produce a waste form that can be decontaminated, but are limited with respect to the geometry and thickness of material that can be sectioned. High-speed abrasive saws provide high cutting rates, but produce nontreatable waste from the cut as well as from blade wear. Hydraulic shearing rapidly produces sectioned material in the small sizes required for decontamination by vibratory finishing. The kerf material also can be decontaminated. However, the glove box first must be sectioned into relatively narrow strips by one of the other techniques.

252 (PNL--3948) FRACTURING OF SIMULATED HIGH-LEVEL WASTE GLASS IN CANISTERS. Peters, R.D.; Slate, S.C. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Sep 1981. Contract AC06-76RL01830. 58p. NTIS, PC A04/MF A01. Order Number DE81030621.

Waste-glass castings generated from engineering-scale developmental processes at the Pacific Northwest Laboratory are generally found to have significant levels of cracks. The causes and extent of fracturing in full-scale canisters of waste glass as a result of cooling and accidental impact are discussed. Although the effects of cracking on waste-form performance in a repository are not well understood, cracks in waste forms can potentially increase leaching surface area. If cracks are minimized or absent in the waste-glass canisters, the potential for radionuclide release from the canister package can be reduced. Additional work on the effects of cracks on leaching of glass is needed. In addition to investigating the extent of fracturing of glass in waste-glass canisters, methods to reduce cracking by controlling cooling conditions were explored. Overall, the study shows that the extent of glass cracking in full-scale, passively-cooled, continuous melting-produced canisters is strongly dependent on the cooling rate. This observation agrees with results of previously reported Pacific Northwest Laboratory experiments on bench-scale annealed canisters. Thus, the cause of cracking is principally bulk thermal stresses. Fracture damage resulting from shearing at the glass/metal interface also contributes to cracking, more so in stainless steel canisters than in carbon steel canisters.

This effect can be reduced or eliminated with a graphite coating applied to the inside of the canister. Thermal fracturing can be controlled by using a fixed amount of insulation for filling and cooling of canisters. In order to maintain production rates, a small amount of additional facility space is needed to accomodate slow-cooling canisters. Alternatively, faster cooling can be achieved using the multi-staged approach. Additional development is needed before this approach can be used on full-scale (60-cm) canisters.

253 (PNL--3957) DEVELOPMENT OF IN-CAN MELTING PROCESS AND EQUIPMENT, 1979 AND 1980. Petkus, L.L.; Larson, D.E.; Bjorklund, W.J.; Holton, L.K. (Pacific Northwest Lab., Richland, WA (USA)). Sep 1981. Contract AC06-76RL01830. 46p. NTIS, PC A03/MF A01. Order Number DE82001050.

Nonradioactive process testing continued with the in-can melter as part of an investigation into the applicability of this vitrification process to various calcined high-level and incinerator ash radioactive wastes. The investigation in this report concentrated on how waste composition and canister fins affect in-can melter capacity and how waste composition affects glass quality. Process performance proved to be generally satisfactory. Pilot-scale in-can melter runs were performed with synthetic, nonradioactive, high-level wastes to produce eight canisters of glass. The synthetic wastes processed included high-level wastes from Savannah River, West Valley, and ICPP, as well as transuranic ash waste. Full-scale in-can melter runs using nonradioactive materials were also conducted, producing ten canisters of glass. Of the ten canisters, nine contained Savannah River Plant glass and one canister contained glass from synthetic zirconia calcine waste from the ICPP. 11.4 tons of glass was produced in test runs. In the full-scale in-can melter furnace, the baffles separating the six heating zones were removed because of baffle warping. A remotely operated section connecting the spray calciner to the canister was tested. Some problems were encountered with calcine plugging.

254 (PNL--3959) MATERIALS AND DESIGN EXPERIENCE IN A SLURRY-FED ELECTRIC GLASS MELTER. Barnes, S.M.; Larson, D.E. (Pacific Northwest Lab., Richland, WA (USA)). Aug 1981. Contract AC06-76RL01830. 194p. NTIS, PC A09/MF A01. Order Number DE82000957.

The design of a slurry-fed electric gas melter and an examination of the performance and condition of the construction materials were completed. The joule-heated, ceramic-lined melter was constructed to test the applicability of materials and processes for high-level waste vitrification. The developmental Liquid-Fed Ceramic Melter (LFCM) was operated for three years with simulated high-level waste and was subjected to conditions more severe than those expected for a nuclear waste vitrification plant.

255 (PNL--3964) BEHAVIOR OF MERCURY AND IODINE DURING VITRIFICATION OF SIMULATED ALKALINE PUREX WASTE. Holton, L.K. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Sep 1981. Contract AC06-76RL01830. 68p. NTIS, PC A04/MF A01. Order Number DE82004325.

Current plans indicate that the high-level wastes stored at the Savannah River Plant will be solidified by vitrification. The behavior of mercury and iodine during the vitrification process is of concern because: mercury is present in the waste in high concentrations (0.1 to 2.8 wt%); mercury will react with iodine and the other halogens present in the waste during vitrification and; the mercury compounds formed will be volatilized from the vitrification process placing a high particulate load in the vitrification system off-gas. Twelve experiments were completed to study the behavior of mercury during vitrification of simulated SRP Purex waste. The mercury was completely volatized from the vitrification system in all experiments. The mercury reacted with iodine, chlorine and oxygen to form a fine particulate solid. Quantitative recovery of mercury compounds formed in the vitrification system off-gas was not possible due to high (37 to 90%) deposition of solids in the off-gas piping. The behavior of mercury and iodine was most strongly influenced by the vitrification system atmosphere. During experiments performed in which the oxygen content of the vitrification system atmosphere was low (< 1 vol%); iodine retention in the glass product was 27 to 55%, the mercury composition of the solids recovered from the off-gas scrub solutions was 75 to 85 wt%, and a small quantity of metallic mercury was recovered from the off-gas scrub solution. During experiments performed in which the oxygen content of the vitrification system atmosphere was high (20 vol%), iodide retention in the glass product was 3 to 15%, the mercury composition of the solids recovered from the off-gas scrub solutions was 60 to 80 wt%, and very little metallic mercury was recovered from the off-gas scrub solution.

256 (PNL--4007) MATERIALS CHARACTERIZATION CENTER MEETING ON IMPACT TESTING OF WASTE FORMS. SUMMARY REPORT. Merz, M.D.; Atteridge, D.; Dudder, G. (Pacific Northwest Lab., Richland, WA (USA)). Oct 1981. Contract AC06-76RL01830. 56p. NTIS, PC A04/MF A01. Order Number DE82003912.

A meeting was held on March 25-26, 1981 to discuss impact test methods for waste form materials to be used in nuclear waste repositories. The purpose of the meeting was to obtain guidance for the Materials Characterization Center (MCC) in preparing the MCC-10 Impact Test Method to be approved by the Materials Review Board. The meeting focused on two essential aspects of the test method, namely the mechanical process, or impact, used to effect rapid fracture of a waste form and the analysis technique(s) used to characterize particulates generated by the impact.

257 (PNL--4045) SELECTION OF A FORM FOR FIXATION OF IODINE-129. Burger, L.L.; Scheele, R.D.; Wiemers, K.D. (Pacific Northwest Lab., Richland, WA (USA)). Dec 1981. Contract AC06-76RL01830. 66p. NTIS, PC A04/MF A01. Order Number DE82007872.

This report summarizes work on the selection of an ^{129}I disposal form. Iodine compounds have been screened on the basis of solubilities, thermal stabilities, cost and availability, toxicity of the cation, and the thermodynamic resistance to oxidation and hydrolysis, and leaching of that compound in portland type III cement. Also considered were iodine capture technology, disposal criteria or guidelines, and the disposal site/strategy. The recommended iodine fixation forms, based on their leach resistance and chemical stability and contingent on the disposal strategy/site and capture technique, are silver iodide in cement and barium, calcium, or strontium, and mercuric iodates in cement. Iodine sodalite appears promising and merits further study. If compatible with disposal requirements, the recommended forms for Mercurex and Iodox are insoluble iodates in cement and for silver sorbents the sorbents in cement or AgI in cement. Conversion between the different oxidation states of iodine is feasible but complicates the iodine treatment. For the different disposal strategies, isotopic dilution or ocean disposal has the least stringent disposal form requirements. Any of the recommended forms should be suitable with proper site selection. Isolation in a geologic repository for thousands of years requires the disposal form to be thermally and chemically stable and resistant to leaching at elevated temperatures. Probably the best form studied for isolation is silver iodide in cement. For extraterrestrial disposal, the disposal form may have to withstand reentry impact and surface disposal in the event of an aborted mission; this assumes the capsule is not recovered. Thus the primary containment barrier is critical. The suggested iodine form for space disposal is a heavy metal iodide.

258 (PNL--4052) FEASIBILITY OF VITRIFYING EPICOR II ORGANIC RESINS. Buelt, J.L. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Nov 1981. Contract AC06-76RL01830. 24p. NTIS, PC A02/MF A01. Order Number DE82003954.

Two laboratory-scale runs have recently been completed to test the feasibility of a single-step incineration/vitrification process for Three Mile Island EPICOR II resins. The process utilizes vitrification equipment, specifically a 15-cm-dia in-can melter, and a specially designed feed technique. Two process tests, each conducted with 1.2 kg of EPICOR II resins loaded with nonradioactive cesium and strontium, showed excellent operational characteristics. Less than 0.8 wt% of the resins were entrained with the gaseous effluents in the second test. Cesium and strontium losses were controlled to 0.71 wt% and less. In addition, all the carbonaceous resins were converted completely to CO_2 with no detectable CO. Future activities are being directed to longer-term tests in laboratory-scale equipment to determine attainable volume reduction, process rates, and material conformance to processing conditions.

259 (PNL--4098) DEVELOPMENT AND TESTING OF MATRICES FOR THE ENCAPSULATION OF GLASS AND CERAMIC NUCLEAR WASTE FORMS. Weld, J.W.; Brite, D.W.; Gurwell, W.E.; Buckwalter, C.Q.; Bunnell, L.R.; Gray, W.J.; Blair, H.T.; Rusin, J.M. (Pacific Northwest Lab., Richland, WA (USA)). Feb 1982. Contract AC06-76RL01830. 116p. NTIS, PC A06/MF A01. Order Number DE82008774.

This report details the results of research on the matrix encapsulation of high level wastes at PNL over the past few years. The demonstrations and tests described were designed to illustrate how the waste materials are effected when encapsulated in an inert matrix. Candidate materials evaluated for potential use as matrices for encapslation of pelletized ceramics or glass marbles were categorized into four groups: metals, glasses, ceramics, and graphite. Two processing techniques, casting and hot pressing, were investigated as the most promising methods of formation or densification of the matrices. The major results reported deal with the development aspects. However, chemical durability tests (leach tests) of the matrix materials themselves and matrix-waste form composites are also reported. Matrix waste forms can provide a low porosity, waste-free barrier resulting in increased leach protection, higher impact strength and improved thermal conductivity compared to unencapsulated glass or ceramic waste materials. Glass marbles encapsulated in a lead matrix offer the most significant improvement in waste form stability of all combinations evaluated. This form represents a readily demonstrable process that provides high thermal conductivity, mechanical shock resistance, radiation shielding and increased chemical durability through both a chemical passivation mechanism and as a physical barrier. Other durable matrix waste forms evaluated, applicable primarily to ceramic pellets, involved hot-pressed titanium or TiO_2 materials. In the processing of these forms, near 100% dense matrices were obtained. The matrix materials had excellent compatibility with the waste materials and superior potential chemical durability. Cracking of the hot-pressed ceramic matrix forms, in general, prevented the realization of their optimum properties.

260 (PNL-SA--6719) RECENT ADVANCES IN SPRAY CALCINATION OF NUCLEAR WASTES. Mikols, W.J.; Romero, L.S.; Bonner, W.F. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jun 1978. Contract EY-76-C-06-1830. 35p. (CONF-780622--75). Dep. NTIS, PC A03/MF A01. From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).

Developments over the last two decades have led to the recent successful demonstration of large-scale spray calcination equipment. The calcination process is ready for full-scale radioactive plant application. The spray calciner is a relatively simple machine. Its components are few and uncomplicated; thus maintenance is minimal, and the equipment lends itself well to remote use. Perhaps the most attractive feature of the spray calciner is its ability to accept an extremely wide range of liquid waste feed compositions. A flow control valve has previously been used to regulate the feed rate to the calciner. Recent tests demonstrate that this item may be eliminated from the feed line, thereby simplifying the overall process flowsheet. Regulation of the atomizing gas-to-liquid feed line pressures can accurately control the liquid flow rate to the calciner. Various types of spray nozzles further allow flexibility in system design. An internal mix nozzle allows the system to operate using a pressurized feed system. Nozzle abrasion has been essentially eliminated by ceramic nozzle inserts. Designs of nonpressurized feed systems have been demonstrated. To insure negligible calcine holdup on the spray chamber wall, pneumatic vibrators have been successfully used for several years without problems. Studies have now been completed based upon vibrator tests to ascertain the long-term effect of vibrator operation on the calcination system. Results indicate that the spray calciner design would not be adversely affected by vibrator operation over a typical 20-year plant life cycle.

261 (PNL-SA--6768) FULL-SCALE IN-CAN MELTER DEMONSTRATION FOR VITRIFICATION OF NUCLEAR WASTE. Blair, H.T. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jan 1979. Contract EY-76-C-06-1830. 19p. (CONF-780622--82). Dep. NTIS, PC A02/MF A01. From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).

A full-scale, nonradioactive in-can melter was made operational in April of 1977 at Pacific Northwest Laboratory (PNL). The furnace's six independently controlled hot zones are capable of providing 30 kW each at 1200°C and are able to accommodate canisters up to 28 in. in dia and 7.5 ft tall. Several new system concepts were demonstrated with this equipment. These included supporting the canister from the bottom, placing the entire can within the furnace, and charging the melter through a water-cooled spout. These new concepts allowed one to eliminate accumulations both of batch over the top of the heat transfer plates and of unvitrified waste in the top of the can by using a test batch of simulated acid waste composition combined with borosilicate glass former: one was able to attain a melting rate of 117 kg/h in a 28-in.-dia canister. A 10-day continuous run was also made in conjunction with a heated wall spray calciner to demonstrate equipment reliability and operability. In addition, the operation of the in-can melting process was demonstrated using only remote monitoing equipment outside of the canister.

262 (PNL-SA--6955) UNITED STATES DEVELOPMENTS IN THE SOLIDIFICATION OF HIGH-LEVEL RADIOACTIVE WASTE. Bonner, W.F.; McElroy, J.L. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jun 1978. Contract EY-76-C-06-1830. 18p. (CONF-781004--6). Dep. NTIS, PC A02/MF A01.

From Nuclear meeting; Basel, Switzerland (3 Oct 1978).

In the spring of 1977, three high-28i.apecity processes for solidifying simulated high-level liquid waste became operational. The three processes are: the joule-heated, ceramic-lined continuous melter, the in-can melter, and the spray calciner. Recent operational experience with these and other processes are described in this report. Since the continuous ceramic melter began operation, both liquid and calcined simulated high-level waste have been fed to the melter. Melting rates of over 100 kg per hour have been demonstrated while calcine feeding an liquid feeding rates of 100 liters per hour have also been demonstrated. A wide range of waste composition has been converted to glass in the ceramic melter including compositions simulating defense wastes stored at Hanford and Savannah River in the United States. The in-can melting furnace is capable of producing canisters of glass up to 28 in. diameter by 8-1/2 ft tall, The maximum melting rate of the in-can melting process has been found to be roughly a linear function of the canister diameter. Melting rates in a 24 in. canister can be over 100 kg per hour. The full-scale heated wall spray calciner has been operated at liquid feed rates exceeding 300 liters per hour. All waste compositions tested are calcineable, even wastes containing over 95 percent sodium cations. Borosilicate glasses have been developed for most of the anticipated high-level radioactive waste compositions. Long-term thermal and radiation stability tests are underway. Glass devitrified during high-temperature storage has a modest factor of 2 to 10 increase in leach rate. Radiation doses equivalent to beyond 250,000 years have not shown any evidence of physical damage to the glass.

263 (PNL-SA--7292) DEVELOPMENT OF MULTIBARRIER NUCLEAR WASTE FORMS. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Mar 1979. Contract EY-76-C-06-1830. 14p. Dep. NTIS, PC A02/MF A01.

The multibarrier concept aims to separate the radionuclide-containing inner core material and the environment by the use of coatings and matrices. Two options were developed for the inner core of the multibarrier concept: supercalcine pellets and glass marbles. Supercalcine is a crystalline assemblage of mutually compatible, refractory, and leach-resistant solid solution phases incorporating high-level liquid waste ions. Supercalcine powder is produced by spray calcining the liquid waste stream to which Al_2O_3, CaO, SiO_2, and SrO have been added. Supercalcine pellets are produced by disc pelletizing. The amorphous supercalcine crystallizes into solid solution phases after subsequent heat treatment. Based on the multibarrier processes described, several conclusions can be made: gravity sintering and vacuum casting are both applicable methods for metal matrix encapsulation. The multibarrier concept of glass marbles encapsulated in a vacuum-cast lead alloy provides enhanced inertness at a minimum increase in technological complexity. If it were desirable to develop a crystalline multibarrier waste form, uncoated sintered supercalcine pellets would offer enhanced inertness at a much lower level of technological complexity than glaze- or CVD-coated supercalcine. The 16-inch diameter pelletizer unit has enough capacity to handle the output of a large PNL spray calciner (52.5 kg of calcine/hr) and it can form spray-calcined material into pellets with diameters of 2 mm to 20 mm having strength enough to withstand handling without significant breakage.Chemical vapor deposition coating of supercalcine should be pursued only if a very high level of inertness is required.

264 (PNL-SA--7358) PHYSICAL MODELING OF ELECTRIC GLASS-MELTING FURNACES FOR HIGH-LEVEL-WASTE IMMOBILIZATION. Quigley, M.S.; Kreid, D.K. (Pacific Northwest Lab., Richland, WA (USA)). Jan 1979. Contract AC06-76RL01830. 15p. (CONF-790808--24). NTIS, PC A02/MF A01. Order Number DE82010744.

From 18. ASME national heat transfer conference; San Diego, CA, USA (Aug 1979).

Physical models have been developed to investigate thermophysical processs in electric glass melting furnaces and to assist with the development of alternative designs. The models were constructed of transparent plexiglass with water cooled surfaces to simulate the boundary conditions in the protctype. The experiments were designed to maintain equality of the Rayleigh, Peclet, and Nusselt numbers and a dimensionless source term. Models with two different length/width/depth ratios have been tested using segmented bar and plate type electrodes. The model fluids were solutions of 7.5% LiCl in glycerine. Fluid velocities were determined from multiple exposure and streak photographs of particles added to the fluid as tracers. The results show that the bulk of the fluid was essentially isothermal with flow patterns that were dominated by two large but indistinct counter-rotating cells. Within each cell the flow was chaotic with no stable structure. Although the results to date have been primarily qualitative, they have been useful in understanding the glass melting processes in the furnaces.

265 (PNL-SA--7414) PROCESSES FOR PRODUCTION OF ALTERNATIVE WASTE FORMS. Ross, W.A.; Rusin, J.M.; McElroy, J.L. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 1979. Contract EY-76-C-06-1830. 14p. (CONF-790204--11). Dep. NTIS, PC A02/MF A01.

From Symposium on waste management and fuel cycles 1979; Tucson, AZ, USA (26 Feb 1979).

During the past 20 years, numerous waste forms and processes have been proposed for solidification of high-level radioactive wastes (HLW). The number has increased significantly during the past 3 to 4 years. At least five factors must be considered in selecting the waste form and process method: 1) processing flexibility, 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 simple system is proposed for making comparisons. This system suggests that one goal for processes would be to reduce the number of process steps, thereby providing less complex processing systems.

266 (PNL-SA--7499) COMPARISON OF BITUMEN AND CEMENT IMMOBILIZATION OF INTERMEDIATE- AND LOW-LEVEL RADIOACTIVE WASTE. Voss, J.W. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 1979. Contract EY-76-C-06-1830. 26p. (CONF-790204--12). Dep. NTIS, PC A03/MF A01.

From Symposium on waste management and fuel cycles 1979; Tucson, AZ, USA (28 Feb 1979).

This paper discusses a systems comparison of two available immobilization processes for intermediate- and low-level radioactive wastes -- bitumen and cement. This study examines a conceptual coprocessed UO_2 - PuO_2 fuel cycle. Radioactive wastes are generated at each stage of this fuel cycle. This study focuses on these transuranic (TRU) wastes generated at a conceptual Fuel Coprocessing Facility. In this report, these wastes are quantified, the immobilization systems conceptualized to process these wastes are presented, and a comparison of the systems is made.

267 (PNL-SA--7535) IMPACT OF DECONTAMINATION ON LWR RADIOACTIVE WASTE TREATMENT SYSTEMS. Hoenes, G.R.; Perrigo, L.D.; Divine, J.R.; Faust, L.G. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 1979. Contract EY-76-C-06-1830. 34p. (CONF-790923--10). Dep. NTIS, PC A03/MF A01.

From Decontamination and decommissioning of nuclear facilities conference; Sun Valley, ID, USA (16 Sep 1979).

Only at N-Reactor is there a means to accommodate radwaste produced during decontamination. The Dresden system is expected to be ready to accommodate such solutions by the summer of 1979. Solidification of the processed decontamination waste may be a significant problem. There is doubt that the materials in current radwaste treatment systems can handle chemicals from a concentrated process. The total storage volume, for concentrated decontamination, is not sufficient in existing radwaste treatment systems. Greater attention should be placed on designing reactors and radwaste treatment systems for decontamination. A means of handling waste material resulting from leaks in the primary system during the decontamination must be developed. On-site storage of solidified decontamination wastes may be a viable option, but license amendments will be necessary.

268 (PNL-SA--7730) ALTERNATIVE WASTE FORMS FOR IMMOBILIZATION OF TRANSURANIC WASTES. Rusin, J.M.; Palmer, C.R. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Sep 1979. Contract AC06-76RL01830. 38p. (CONF-791045--9). NTIS, PC A03/MF A01.

From American Ceramic Society conference; New Orleans, LA, USA (14 Oct 1979).

Several alternative waste forms are being considered for the immobilization of consolidated transuranic (TRU) wastes. General characteristics and applicability of these waste forms to TRU wastes are reviewed. Specific attention is given to recent experimental results on sintered ceramic TRU waste forms including bulk properties, impact resistance, leachability, and volatility.

269 (PNL-SA--7322) GLASS: AN AVAILABLE MATERIAL FOR THE IMMOBILIZATION OF NUCLEAR WASTE. Mendel, J.E. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Sep 1979. Contract AC06-76RL01830. 21p. (CONF-791103--113). NTIS, PC A02/MF A01.

From American Nuclear Society meeting; San Francisco, CA, USA (12 Nov 1979).

This paper describes the properties of waste glass as it can be produced today, and shows how the properties of waste glass can fit into a multibarrier nuclear waste management system. The composition of the wastes that are to be immobilized and the design of processing equipment are also discussed.

270 (PNL-SA--8014) DECOMMISSIONING OF COMMERCIAL SHALLOW-LAND BURIAL SITES. Murphy, E.S.; Holter, G.M. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 1979. Contract AC06-76RL01830. 15p. (CONF-790923--13). NTIS, PC A02/MF A01.

From Decontamination and decommissioning of nuclear facilities conference; Sun Valley, ID, USA (16 Sep 1979).

Estimated costs and safety considerations for decommissioning LLW burial grounds have been evaluated. Calculations are based on a generic burial ground assumed to be located at a western and an eastern site. Decommissioning modes include: (1) site stabilization followed by long-term care of the site; and (2) waste relocation. Site stabilization is estimated to cost from $0.4 million to $7.5 million, depending on the site and the stabilization option chosen. Long-term care is estimated to cost about $100,000 annually, with somewhat higher costs during early years because of increased site maintenance and environmental monitoring requirements. Long-term care is required until the site is released for unrestricted public use. Occupational and public safety impacts of site stabilization and long-term care are estimated to be small. Relocation of all the waste from a reference burial ground is estimated to cost more than $1.4 billion and to require more than 20 years for completion. Over 90% of the cost is associated with packaging, transportation, and offsite disposal of the exhumed waste. Waste relocation results in significant radiation exposure to decommissioning workers.

271 (PNL-SA--8027) DESIGN AND OPERATION OF A 100 KG/H ELECTRIC MELTER FOR NUCLEAR-WASTE VITRIFICATION. Dierks, R.D. (Pacific Northwest Lab., Richland, WA (USA)). 1980. Contract AC06-76RL01830. 16p. (CONF-800914--4). NTIS, PC A02/MF A01. Order Number DE82005782.

From IEEE meeting; Cincinnati, OH, USA (29 Sep 1980).

The proposed use of ceramic-lined, electric-heated glass melters for the vitrification of nuclear wastes requires careful adaptation of commercial glass furnace practices. The vitrification of simulated nuclear waste calcines was studied in a ceramic-lined melter with a glass surface area of 0.76 m^2. The melter contained 0.25 m^3 of glass, which was heated by the flow of an ac current (ranging from 600 to 1200 amps) between two Inconel-690, slab-type elecrodes immersed directly in the glass at either end of the melter tank. The melter was maintained at operating temperatures for 13.5 mo, and produced 62,000 kg of glass. The maximum sustained operating period was 122 h, during which glass was produced at the rate of 70 kg/h. The basic design features of the melter, and some of the operating experiences, are discussed.

272 (PNL-SA--8078) VITRIFICATION OF TRU INCINERATOR ASH. Mellinger, G.B.; Palmer, C.R.; Petkus, L.L. (Pacific Northwest Lab., Richland, WA (USA)). Sep 1979. Contract AC06-76RL01830. 32p. (CONF-791045--10). NTIS, PC A03/MF A01. Order Number DE82005776.
From American Ceramic Society conference; New Orleans, LA, USA (14 Oct 1979).
Compositions and properties of glasses developed for vitrification of TRU incinerator ashes are presented. The effects on glasses of variations in the ash are explored. Preliminary experience with engineering-scale melting tests is reported.

273 (PNL-SA--8085) HIGH LEVEL WASTE PROPERTIES. Turcotte, R.P. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 1979. Contract AC06-76RL01830. 36p. (CONF-7910184--1). NTIS, PC A03/MF A01.
From IAEA evaluation of solidified high level radioactive waste products workshop; Berlin, F.R. Germany (15 Oct 1979).
Devitrification and leaching analyses of four waste glasses were made to compare non-radioactive compositions to compositions made using fully radioactive waste calcine. Microstructural analyses of the phase behavior of glasses were performed by means of optical microscopy, x-ray diffraction, x-ray fluorescence, scanning electron microscopy, and electron microprobe analysis. The author's summary of the major findings are: Melt insolubles and crystallization products were found to the same extent in both radioactive and non-radioactive glasses of similar composition. High radiation field appeared to have no effect on the crystallization behavior. The results of long-term IAEA static leach tests indicated no significant difference between the average leach rates of the fully radioactive and non-radioactive glass formulations. Glass composition was more important in determining leach rates than was the extent of devitrification. In both short time tests at 75°C or longer leach tests at 25°C elemental analyses suggested that congruent dissolution did not occur.

274 (PNL-SA--8125) ASPHALT-EMULSION SEALING OF URANIUM-MILL TAILINGS. Hartley, J.N.; Koehmstedt, P.L.; Esterl, D.J. (Pacific Northwest Lab., Richland, WA (USA)). 1979. Contract AC06-76RL01830. 14p. (CONF-791140--2). NTIS, PC A02/MF A01. Order Number DE82010746.
From 2. symposium on uranium mill tailings management; Fort Collins, CO, USA (19 Nov 1979).
Portions of document are illegible.
The use of asphalt emulsion to contain radon and radium in uranium tailings is being investigated at the Pacific Northwest Laboratory. Results of these studies indicate that a radon flux reduction of greater than 99% can be obtained using either a poured-on/sprayed-on seal (3.0 to 7.0 mm thick) or an admix seal (2.5 to 12.7 cm thick) containing about 18 wt % residual asphalt. A field test was carried out at the Grand Junction tailings pile in order to demonstrate the sealing process. A reduction in radon flux ranging from 4.5 to greater than 99% (76% average) was achieved using a 12.7-cm (5-in.) admix seal with a sprayed-on top coat. A hydrostatic stabilizer used to apply the admix was followed by compaction, which formed the radon seal. Overburden was applied to provide a protective soil layer over the seal. Included in part of the overburden was a herbicide to prevent root penetration.

275 (PNL-SA--8169) DEVELOPMENT OF A REMOTE LABORATORY-SCALE WASTE TREATMENT FACILITY. Knox, C.A.; Hanson, M.S. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jun 1980. Contract AC06-76RL01830. 21p. (CONF-800607--90). NTIS, PC A02/MF A01.
From American Nuclear Society annual meeting; Las Vegas, NV, USA (8 Jun 1980).
A waste treatment facility, designed on the basis of a feedrate of 1 l/hr of concentrated waste to a spray calciner, has been installed in a radiochemical hot cell at Pacific Northwest Laboratory. The facility includes three modules: feed preparation (storage tanks, evaporator, condenser), waste solidification (a spray calciner and in-can melter), and effluent control (venturi scrubber, cyclone separator, fission product adsorbers, nitrogen oxides destructor, iodine adsorber, HEPA filter, and packed scrubber). The system is flexible. The spray calciner and in-can melter can be easily removed and replaced by alternative solidification systems, and the effluent control system can be operated in many different sequences. Other components can be easily added to the effluent system for tests. Two effluent control flowsheets, designed to simulate those in defense waste and commercial waste processing plants, will be evaluated during the first radioactive runs. Most operational data from the system are remotely recorded continuously on strip-chart and multipoint recorders. Data on equipment operating parameters and upset conditions will be used to help maximize data on effluents, effluent decontamination factors and product quality. Five laboratory, pilot- and full-scale radioactive and nonradioactive waste solidification systems have already been operated at PNL. Experience with these systems demonstrated a need for additional radioactive work. Thus, the Remote Laboratory-Scale Waste Treatment Facility was developed. Operations completed with the other systems have indicated that scaling factors related to equipment size will not be a major consideration in the interpretation and usage of results from this equipment. These results can be used to provide guidance in developing full-scale radioactive waste treatment equipment.

276 (PNL-SA--8236) DESIGN AND INSTALLATION OF A LABORATORY-SCALE SYSTEM FOR RADIOACTIVE WASTE TREATMENT. Berger, D.N.; Knox, C.A.; Siemens, D.H. (Battelle Pacific Northwest Labs., Richland, WA (USA)). May 1980. Contract AC06-76RL01830. 14p. (CONF-800607--92). NTIS, PC A02/MF A01.
From American Nuclear Society annual meeting; Las Vegas, NV, USA (8 Jun 1980).
Described are the mechanical design features and remote installation of a laboratory-scale radiochemical immobilization system which is to provide a means at Pacific Northwest Laboratory of studying effluents generated during solidification of high-level liquid radioactive waste. Detailed are the hot cell, instrumentation, two 4-in. and 12-in. service racks, the immobilization system modules - waste feed, spray calciner unit, and effluent - and a gamma emission monitor system for viewing calcine powder buildup in the spray calciner/in-can melter.

277 (PNL-SA--8267) ESTIMATES OF AMOUNTS OF SOIL REMOVAL FOR CLEAN-UP OF TRANSURANICS AT NAEG OFFSITE SAFETY-SHOT SITES. Kinnison,

R.R.; Gilbert, R.C. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jan 1980. Contract AC06-76RL01830. 8p. (CONF-800157--1). NTIS, PC A02/MF A01.

From Nevada applied ecology group (NAEG) information meeting; Las Vegas, NV, USA (29 Jan 1980).

Rough estimates are given for the amount of soil removal necessary to decontaminate five representative safety-shot areas. In order to decontaminate to levels of less than 160 pCi ^{239}Pu per gram of surface soil, it is estimated that over one-half million tons of soil would have to be removed from the five areas. This is a preliminary estimate based on summary data and concentration contour maps readily available in NAEG publications. More accurate estimates could be obtained by applying Kriging techniques to available soil data if the need arises. The inclusion of ^{241}Am and ^{238}Pu activities do not significantly increase the soil tonnage estimates obtained for $^{239,240}Pu$ because of their relatively small contributions to total transuranic activity. The magnitude of the errors inherent in our use of summary data to obtain rough estimates also suggests that a revision of the tonnage estimates for $^{239,240}Pu$ to include ^{241}Am and ^{238}Pu is not warranted.

278 (PNL-SA--8280) COMPARATIVE STUDY OF ALTERNATIVE HIGH-LEVEL WASTE-SOLIDIFICATION PROCESSES. Treat, R.L. (Pacific Northwest Lab., Richland, WA (USA)). Jan 1980. Contract AC06-76RL01830. 13p. (CONF-800313--12). NTIS, PC A02/MF A01. Order Number DE82010671.

From Waste management conference; Tucson, AZ, USA (10 Mar 1980).

The followig nine processes were evaluated: in-can glass melting process; joule-heated glass melting process; glass-ceramic process; marbles-in-lead matrix process; coated supercalcine pellets-in-lead matrix process; SYNROC process; titanate process; concrete process; and cermet process. In general, and probably as expected, it can be concluded that the processes with the fewest processing components require the fewest motorized parts and the fewest steps to produce a canister of waste. The requirements for the in-can glass melting, joule-heated glass melting and glass-ceramic processes are lower in all categories when compared to any of the remaining six processes. The average difference in requirements between the in-can glass melting, joule-heated glass melting and glass-ceramic processes and the remaining six processes is about a factor of 1.5 to 2. As previously stated, in-cell and out-of-cell service equipment were not included. Equipment requirements are also likely to change as these processes reach a more advanced state of development. Other equipment will likely be required as process and waste form criteria are established. Although the equipment and process steps identified in this study are of somewhat limited value, they could serve as a basis for more advanced analyses. For example, the reliability of each unit of equipment and each process step identified in this study could be estimated. Repair or replacement times for failed equipment could also be estimated. With reliability and repair or replacement data, it may also be possible to determine the additional equipment requirements, if any, for operating the process at its design rate. With a more complete summary of equipment requirements, other important processing parameters, such as product variability, safety and ease of decontamination, could be estimated.

279 (PNL-SA--8289) HIGH-LEVEL WASTE VITRIFICATION OFF-GAS CLEANUP TECHNOLOGY. Hanson, M.S. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jan 1980. Contract AC06-76RL01830. 11p. (CONF-800313--10). NTIS, PC A02/MF A01.

From Waste management conference; Tucson, AZ, USA (10 Mar 1980).

This brief overview is intended to be a basis for discussion of needs and problems existing in the off-gas clean-up technology. A variety of types of waste form and processes are being developed in the United States and abroad. A description of many of the processes can be found in the Technical Alternative Documents (TAD). Concurrently, off-gas processing systems are being developed with most of the processes. An extensive review of methodology as well as decontamination factors can be found in the literature. Since it is generally agreed that the most advanced solidification process is vitrification, discussion here centers about the off-gas problems related to vitrification. With a number of waste soldification facilities around the world in operation, it can be shown that present technology can satisfy the present requirement for off-gas control. However, a number of areas within the technology base show potential for improvement. Fundamental as well as verification studies are needed to obtain the improvements.

280 (PNL-SA--8358) RECOVERY AND USE OF FISSION PRODUCT NOBLE METALS. Jensen, G.A.; Rohmann, C.A.; Perrigo, L.D. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jun 1980. Contract AC06-76RL01830. 23p. (CONF-800607--91). NTIS, PC A02/MF A01.

From American Nuclear Society annual meeting; Las Vegas, NV, USA (9 Jun 1980).

Noble metals in fission products are of strategic value. Market prices for noble metals are rising more rapidly than recovery costs. A promising concept has been developed for recovery of noble metals from fission product waste. Although the assessment was made only for the three noble metal fission products (Rh, Pd, Ru), there are other fission products and actinides which have potential value. (DLC)

281 (PNL-SA--8368) SURFACE DECONTAMINATION OF SOLID WASTE. McCoy, M.W.; Allen, R.P.; Arrowsmith, H.W. (Pacific Northwest Lab., Richland, WA (USA)). Apr 1980. Contract AC06-76RL01830. 22p. (CONF-800802--28). NTIS, PC A02/MF A01. Order Number DE82007084.

From 89. annual meeting of the American Institute of Chemical Engineers; Portland, OR, USA (17 Aug 1980).

This paper summarizes work in progress at Pacific Northwest Laboratory to develop vibratory finishing into a large-scale decontamination system that can minimize the volume of surface-contaminated metallic and nonmetallic waste requiring geologic disposal. Vibratory finishing is a mass finishing process used in the metal finishing industry to debur, clean and improve surface finishes. The process combines a mechanical scrubbing action of a solid medium with the cleaning action of a liquid compound. The process takes place in a vibrating tub. Tests have demonstrated the ability to rapidly reduce contamination levels of transuranic-contaminated waste to substantially less than 10 nCi/g, the current limit for transuranic waste. The process is effective on a wide range of materials including stainless steel, Plexiglas, Neoprene, and Hypalon, the principal materials in Hanford glove boxes.

282 (PNL-SA--8414) URANIUM MILL TAILINGS STABILIZATION. Hartley, J.N.; Koenmstedt, P.L.; Esterl, D.J.; Freeman, H.D. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Feb 1980. Contract AC06-76RL01830. 16p. (CONF-800313--9). NTIS, PC A02/MF A01.

From Waste management conference; Tucson, AZ, USA (10 Mar 1980).

Uranium mill tailings pose a potential radiation health hazard to the public. Therefore, stabilization or disposal of these tailings in a safe and environmentally sound way is needed to minimize radon exhalation and other environmental hazards. One of the most promising concepts for stabilizing U tailings is the use of asphalt emulsion to contain radon and other hazardous materials within uranium tailings. This approach is being investigated at the Pacific Northwest Laboratory. Results of these studies indicate that a radon flux reduction of greater than 99% can be obtained using either a poured-on/sprayed-on seal (3.0 to 7.0 mm thick) or an admixture seal (2.5 to 12.7 cm thick) containing about 18 wt % residual asphalt. A field test was carried out in June 1979 at the Grand Junction tailings pile in order to demonstrate the sealing process. A reduction in radon flux ranging from 4.5 to greater than 99% (76% average) was achieved using a 15.2-cm (6-in.) admix seal with a sprayed-on top coat. A hydrostatic stabilizer was used to apply the admix. Following compaction, a spray coat seal was applied over the admix as the final step in construction of a radon seal. Overburden was applied to provide a protective soil layer over the seal. Included in part of the overburden was a herbicide to prevent root penetration.

283 (PNL-SA--8521) TRIBUTYL PHOSPHATE REMOVAL FROM REPROCESSING OFF-GAS STREAMS USING A SELECTED SORBENT. Parker, G.B. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 1980. Contract AC06-76RL01830. 15p. (CONF-801038--18). NTIS, PC A02/MF A01.

From 16. DOE nuclear air cleaning conference; San Diego, CA, USA (20 Oct 1980).

Laboratory experiments used small laboratory-scale columns packed with selected sorbant materials to remove tributyl phosphate (TBP) and iodine at conditions approaching those in actual reprocessing off-gas streams. The sorbent materials for TBP removal were placed upstream of iodine sorbent materials to protect the iodine sorbent from the deleterious effects of TBP. Methyl iodide in an airstream containing 30% TBP in normal paraffin hydrocarbons (NPH) and water vapor was metered to two packed columns of sorbents simultaneously (in parallel). One column contained a segment of 8-in. x 14-in. mesh alumina sorbent for TBP removal, the other did not. The measure of the effectiveness of TBP sorbent materials for TBP removal was determined by comparing the iodine retention of the iodine sorbent materials in the two parallel columns. Results from an 18 wt % Ag substituted mordenite iodine sorbent indicated that the iodine retention capacity of the sorbent was reduced 60% by the TBP and that the column containing iodine sorbent material protected by the alumina TBP sorbent retained 30 times more iodine than the column without TBP sorbent. TBP concentration was up to 500 mg/m^3. Similar experiments using a 7 wt % Ag impregnated silica gel indicated that the TBP vapor had little effect on the iodine retention of the silica gel material. The stoichiometric maximum amount of iodine was retained by the silica gel material. Further experiments were conducted assessing the effects of NO_2 on iodine retention of this 7 wt % Ag sorbent. After the two columns were loaded with iodine in the presence of TBP (in NPH), one column was subjected to 2 vol % NO_2 in air. From visual comparison of the two columns, it appeared that the NO_2 regenerated the silica gel iodine sorbent and that iodine was washed off the silica gel iodine sorbent leaving the sorbent in the original state.

284 (PNL-SA--8732) RADIATION EFFECTS IN CRYSTALLINE HIGH-LEVEL NUCLEAR WASTE SOLIDS. Weber, W.J.; Wald, J.W.; Gray, W.J. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Oct 1980. Contract AC06-76RL01830. 9p. (CONF-801124--33). NTIS, PC A02/MF A01.

From 3. annual meeting of the Materials Research Society; Boston, MA, USA (17 Nov 1980).

Glass, cement, and crystalline-ceramic waste forms are being considered as potential solid forms for incorporation of nuclear wastes. The solidified waste will be subjected to high doses of many different radiations which may measurably alter physical properties and/or affect the durability of the solid waste form. As a result, the long-term stability of these waste forms is a subject of continuing research. The present paper defines the general radiation damage problem and summarizes experimental results from studies of the effects of alpha decay, alpha bombardment, and transmutations on crystalline waste forms, related single-phase compounds and some glass waste forms. The results lead to the following conclusions: both alpha-recoil damage in actinide host phases and alpha damage to all other phases must be considered since significant structural changes may occur from either source; the ingrowth of damage follows exponential behavior for both alpha-recoil damage and alpha damage, leading to saturation effects in most materials; and preliminary results show no significant effect of transmutations on waste form stability.

285 (PNL-SA--8755) MANAGEMENT OF INTERMEDIATE-LEVEL RADIOACTIVE WASTES IN THE UNITED STATES. Aaberg, R.L.; Lakey, L.T.; Greenboro, J. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jul 1980. Contract AC06-76RL01830. 21p. (CONF-800802--25). NTIS, PC A02/MF A01.

From 89. annual meeting of the American Institute of Chemical Engineers; Portland, OR, USA (17 Aug 1980).

While used extensively, the term intermediate-level waste is not a clearly defined waste category. Assuming the ILW includes all radioactive wastes requiring shielding but not ordinarily included in a high-level waste canister, its major sources include power plant operations, spent fuel storage, and spent fuel reprocessing. While the volume is approx. 10^2 greater than that of high-level waste, ILW contains only approx. 1% of the radioactivity. Power plant waste, constituting approx. 87% of the waste volume, is generally nontransuranic waste. The other approximately 13% from fuel reprocessing is generally transuranic. Intermediate-level wastes fall into the general categories of highly radioactive hardware, failed equipment, HEPA filters, wet wastes, and noncombustible solids. Within each category, however, the waste characteristics can vary widely, necessitating different treatments. The wet wastes, primarily power plant resins and sludges, contribute the largest volume; fuel hulls and core hardware represent the greatest activity. Numerous treatments for intermediate-level wastes are available and have been used successfully. Packaging and transportation systems are also available. Intermediate-level wastes from power plants are disposed of by shallow-land burial. However, the alpha-bearing wastes are being stored pending eventual disposal to a geologic repository or by other means, e.g., intermediate-depth burial, sea disposal. Problem areas associated with intermediate-level wastes include: disposal criteria need to be established; fixation of organic ion exchange resins from power plant operation needs improvement; and reprocessing of LWR fuels will produce ILW considerably different from power plant ILW and requiring different treatment.

286 (PNL-SA--8762) DIRECT LIQUID-FEEDING OF DEFENSE WASTES TO A CONTINUOUS MELTER. Brouns, R.A. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 1980. Contract AC06-76RL01830. 24p. (CONF-801107--67). NTIS, PC A02/MF A01.

From ANS international conference; Washington, DC, USA (17 Nov 1980).

Full-scale studies involving the slurry-feeding of a simulated high-level defense waste to an electric glass-melter have been successfully completed. Steady-state operation was achieved, giving encouraging results on process capacities. Reducing agents added to the feed helped to control glass-foaming during melting. A reduction in the formation of insoluble precipitates in the melter was noted. Volatility of the radionuclide substitutes in the feed was found to be low, except for Cs. Further chemical volatility studies are required due to the unexplainably high volatility measured for the Cs. Particulate entrainment in the offgas during melter operation was also determined to be low.

287 (PNL-SA--8931) EFFECTS OF CRACKS ON GLASS-LEACHING. Perez, J.M. Jr.; Westsik, J.H. Jr. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 1980. Contract AC06-76RL01830. 19p. (CONF-801206--10). NTIS, PC A02/MF A01.

From ORNL conference on the leachability of radioactive solids; Gatlinburg, TN, USA (9 Dec 1980).

The effects of cracks on glass leaching have been studied to determine to what degree the increase in the glass surface area caused by cracks might contribute to the overall leachability of the glass. Borosilicate glass in the shape of cylindrical disks (pellets) has been used in a static leach test. The pellets were leached in deionized water at 90°C. Parameters included crack widths of nominally zero and 0.038-cm, and four ratios of crack surface area to total surface area; zero, 0.28, 0.55, and 0.70. Leaching from the cracks with 0.038-cm crack widths are found to be two to five times less than leaching from uncracked samples. No apparent leaching from crack surfaces with nominally zero crack-widths was observed.

288 (PNL-SA--9377) CRYSTAL STRUCTURE AND STOICHIOMETRY OF THE CA/SUB 2+X/ND/SUB 8-X/ $(SIO_4)_6O$/SUB 2-1/2X/ SYSTEM. Fahey, J.A.; Weber, W.J. (Pacific Northwest Lab., Richland, WA (USA); Bronx Community Coll., New York (USA)). Jun 1981. Contract AC06-76RL01830. 5p. (CONF-810675--4). NTIS, PC A02/MF A01. Order Number DE81030079.

From 15. rare earth research conference; Rolla, MO, USA (15 Jun 1981).

A systematic study of Ca/sub 2+x/Nd/sub 8-x/ $(SiO_4)_6O$/sub 2-1/2x/, one of the mineral-like phases formed by the rare earths in the supercalcine-ceramic nuclear waste forms, has been undertaken. Until now, the structure and stoichiometry of this apatite phase has only been inferred from the hexagonal symmetry revealed by its powder diffraction data. The goal of this study was to obtain a complete set of atomic coordinates and the temperature factor and occupation factor for each atom in this structure by applying the Rietveld profile analysis technique to powder x-ray diffraction data. Samples of several bulk compositions in the range 0 less than or equal to x less than or equal to 4 were prepared in order to evaluate the solid solution limits of the apatite phase. The structure of $Ca_2Nd_8(SiO_4)_6O_2$ has been shown to be isostructural with $Ca_{10}(PO_4)_6F_2$. The metal ions occupying the 4f site are coordinated by six silicate oxygens at an average distance of 2.56 A. The Nd(+3) ions occupying the 6h site are coordinated by six silicate oxygens at an average distance of 2.49 A and by one anionic oxygen at 2.27 A. The anionic oxygen in 2a is coordinated by three Nd(+3) ions. The structure of $Ca_3Nd_7(SiO_4)_6O_1$ is similar except that five Nd(+3) and one Ca(+2) ions occupy the 6h site and 1/4 of the anionic oxygens are missing in a random fashion.

289 (PNL-SA--9440) REVIEW OF MULTIBARRIER WASTE FROM DEVELOPMENT. Rusin, J.M. (Pacific Northwest Lab., Richland, WA (USA)). Apr 1981. Contract AC06-76RL01830. 6p. (CONF-810650--3). NTIS, PC A02/MF A01. Order Number DE82005982.

From International seminar on chemistry and process engineering for high-level waste solidification; Julich, F.R. Germany (1 Jun 1981).

The application of coatings and matrices as part of a multibarrier concept for the immobilization of nuclear wastes has been demonstrated by several investigators. The multibarrier concept provides an enhanced waste form by improvements in mechanical strength, leach resistence, and thermal stability. Process complexity is increased in the multibarrier process due to additional processing steps, high-coating temperatures, molten metal transfer, and flammable coating gas mixtures.

290 (PNL-SA--9591) CRYSTALLIZATION BEHAVIOR OF NUCLEAR WASTE FORMS. Rusin, J.M.; Lokken, R.O.; May, R.P.; Wald, J.W. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Sep 1981. Contract AC06-76RL01830. 21p. (CONF-811013--12). NTIS, PC A02/MF A01. Order Number DE82003863.

From Electrochemical Society conference; Denver, CO, USA (11 Oct 1981).

Several waste form options have been or are being developed for the immobilization of high-level wastes. The final selection of a waste form must take into consideration both waste form product as well as process factors. Crystallization behavior has an important role in nuclear waste form technology. For glass or vitreous waste forms, crystallization is generally controlled to a minimum by appropriate glass formulation and heat treatment schedules. With glass ceramic waste forms, crystallization is essential to convert glass products to highly crystalline waste forms with a minimum residual glass content. In the case of ceramic waste forms, additives and controlled sintering schedules are used to contain the radionuclides in specific tailored crystalline phases.

291 (PNL-SA--9642) THREE MILE ISLAND ZEOLITE VITIRIFICATION DEMONSTRATION PROGRAM.

Siemens, D.H.; Knowlton, D.E.; Shupe, M.W. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jun 1981. Contract AC06-76RL01830. 12p. (CONF-810814--5). NTIS, PC A02/MF A01. Order Number DE81028010.

From 91. national meeting of the American Institute of Chemical Engineers; Detroit, MI, USA (16 Aug 1981).

The cleanup of the high-activity-level water at Three Mile Island (TMI) provides an opportunity to further develop waste management technology. Approximately 790,000 gallons of high-activity-level water at TMI's Unit-2 Nuclear Power Station will be decontaminated at the site using the submerged demineralizer system (SDS). In the SDS process, the cesium and strontium in the water are sorbed onto zeolite that is contained within metal liners. The Department of Energy has asked the Pacific Northwest Laboratory (PNL) to take a portion of the zeolite from the SDS process and demonstrate, on a production scale, that this zeolite can be vitrified using the in-can melting process. This paper is a brief overview of the TMI zeolite vitrification program. The first section discusses the formulation of a glass suitable for immobilizing SDS zeolite. The following section describes a feed system that was developed to feed zeolite to the in-can melter. It also describes the in-can melting process and the government owned facilities in which the demonstrations will take place. Finally, the schedule for completing the program activities is outlined.

292 (PNL-SA--9649) STANDARD TESTS FOR THERMAL STABILITY OF NUCLEAR WASTE FORMS. Weber, W.J.; Gray, W.J.; May, R.P. (Pacific Northwest Lab., Richland, WA (USA)). Oct 1981. Contract AC06-76RL01830. 8p. (CONF-811122--41). NTIS, PC A02/MF A01. Order Number DE82005232.

From Annual meeting of the Materials Research Society; Boston, MA, USA (16 Nov 1981).

Three test methods have been proposed for evaluating the thermal stability of nuclear waste forms. The time-temperature dependence of thermally activated phase transformations are measured by the proposed MCC-7 Recommended Practice for Testing Thermal Phase Stability in Radioactive Waste Forms. The proposed MCC-8 High Temperature Vaporization Test Method measures the vaporization of fission products and other condensable elements. The potential pressurization of a canister at elevated temperatures is determined by the proposed MCC-9 Thermal Gas Generation Test Method. These three test methods are described in detail and the results of applying the MCC-7 test method to PNL 78-157 waste glass are presented. The PNL 78-157 shows a maximum ingrowth of CeO_2 (5 wt %) at 950°C. The time-temperature-transformation behavior of PNL 76-157 is described with respect to the ingrowth of CeO_2.

293 (PNL-SA--9654) COMPARATIVE ASSESSMENT OF TRU WASTE FORMS AND IMMOBILIZATION PROCESSES. Ross, W.A.; Harvey, C.O.; Lokken, R.O.; May, R.P.; Roberts, F.P.; Timmerman, C.L.; Treat, R.L.; Westsik, J.H. Jr. (Pacific Northwest Lab., Richland, WA (USA)). Oct 1981. Contract AC06-76RL01830. 9p. (CONF-811122--42). NTIS, PC A02/MF A01. Order Number DE82005231.

From Annual meeting of the Materials Research Society; Boston, MA, USA (16 Nov 1981).

Six alternative TRU waste forms and seven waste immobilization processes are comparatively assessed on the basis of both product properties and process costs and risks. The waste forms are characterized for their leachability, mechanical strength, and thermal and radiation stability. The processes are evaluated in terms of costs (for processing, transportation, and repository disposal) and in terms of occupational exposure, industrial hazard, and quality assurance. Cast cement is recommended for immobilization of defense TRU wastes. A glass system, either borosilicate or aluminosilicate, is recommended for immobilization of commercial TRU wastes.

294 (PNL-SA--9739) LIQUID-WASTE IMMOBILIZATION BY AGGLOMERATION. Timmerman, C.L.; Carter, J.G. (Pacific Northwest Lab., Richland, WA (USA)). Aug 1981. Contract AC06-76RL01830. 15p. (CONF-810897--2). NTIS, PC A02/MF A01. Order Number DE82005917.

From 17. biennial conference of the Institute of Briquetting and Agglomeration; Reno, NV, USA (31 Aug 1981).

Liquid nuclear and chemical wastes may require their immobilization into a stable waste form via a processing scheme. An agglomeration process to immobilize these wastes with a solid, absorbent binder is presented. An experiment using a simulated (nonradioactive) liquid nuclear-waste stream and sodium titanate as the solid binder was performed to test the feasibility of this process. Results from this experiment are encouraging and indicate a need to test this liquid waste agglomeration process on other liquid waste streams.

295 (PNL-SA--9855) REVIEW OF RADIATION EFFECTS IN SOLID-NUCLEAR-WASTE FORMS. Weber, W.J. (Pacific Northwest Lab., Richland, WA (USA)). Sep 1981. Contract AC06-76RL01830. 54p. (CONF-8110113--2). NTIS, PC A04/MF A01. Order Number DE82003842.

From 3. Argonne workshop on basic problems in nuclear waste management; Argonne, IL, USA (1 Oct 1981).

Radiation effects on the stability of high-level nuclear waste (HLW) forms are an important consideration in the development of technology to immobilize high-level radioactive waste because such effects may significantly affect the containment of the radioactive waste. Since the required containment times are long (10^3 to 10^6 years), an understanding of the long-term cumulative effects of radiation damage on the waste forms is essential. Radiation damage of nuclear waste forms can result in changes in volume, leach rate, stored energy, structure/microstructure, and mechanical properties. Any one or combination of these changes might significantly affect the long-term stability of the nuclear waste forms. This report defines the general radiation damage problem in nuclear waste forms, describes the simulation techniques currently available for accelerated testing of nuclear waste forms, and reviews the available data on radiation effects in both glass and ceramic (primarily crystalline) waste forms. 76 references.

296 (PNL-SA--9959) ESTIMATING ANALYTICAL AND PROCEDURAL COMPONENTS OF VARIANCE FROM A ROUND ROBIN ON A STATIC LEACH TEST. Johnston, J.W.; Mendel, J.E.; Daniel, J.L. (Pacific Northwest Lab., Richland, WA (USA)). 1981. Contract AC06-76RL01830. 9p. (CONF-811125--3). NTIS, PC A02/MF A01. Order Number DE82010740.

From DOE statistical symposium; Long Island, NY, USA (4 Nov 1981).

The Materials Characterization Center (MCC) at the US Department of Energy's (DOE) Pacific

Northwest Laboratory (PNL) conducted an interlaboratory test of the MCC-1 Static Leach Test, a standard method developed by MCC for measuring the leachability of nuclear waste forms. Results were gathered from 20 laboratories on 25 element by waste form by leachant combinations. These results were used to characterize the precision of the method and to allocate the total variability of the method into four components: (1) within-laboratory analytical; (2) within-laboratory procedural; (3) between-laboratory analytical; and (4) between-laboratory procedural. Procedural refers to preparing the waste form specimens and leachants used and conducting the leach test to produce leachates for analysis. Analysis refers to sampling an aliquot of the leachate and chemically analyzing it. Within-laboratory variability both analytical and procedural accounts for only about 5% of the total. Between-laboratory variability was dominated by the analytical component in the lower element average range and by the procedural component in the higher range. An optimistic characterization of the overall single determination precision of the method is that it has a one standard deviation relative reproducibility of 31%. Depending on the element by waste form by leachant combination, the relative reproducibility ranged from 15% to 120%. The statistical results discussed here are preliminary.

297 (RHO-SA--145) REMOVAL OF RADIONUCLIDES FROM THE WATER-SOLUBLE FRACTION OF HANFORD NUCLEAR DEFENSE WASTES. Strachan, D.M.; Schulz, W.W. (Battelle Pacific Northwest Labs., Richland, WA (USA); Atomics International Div., Richland, WA (USA). Rockwell Hanford Operations). Feb 1980. Contract AC06-77RL01030. 33p. (CONF-800313--9). NTIS, PC A03/MF A01.

From Waste management conference; Tucson, AZ, USA (10 Mar 1980).

The current Hanford Waste Management Program has operated since 1968 to remove the bulk of the long-lived heat emitters ^{90}Sr and ^{137}Cs from stored high-level wastes. The liquid waste remaining after removal of ^{90}Sr and ^{137}Cs is returned to underground tanks for eventual evaporation to damp solid salt cake. Hanford salt cake is an admixture of large amounts of $NaNO_3$ with minor amounts of $NaAl(OH)_4$, Na_2CO_3, Na_2CrO_4, Na_3PO_4, Na_2SO_4, and NaOH. Small amounts of ^{90}Sr, ^{137}Cs, ^{99}Tc and other radionuclides are also associated with the salt cake. Results of successful bench-scale radionuclide removal tests with various actual salt cakes and certain liquid waste solutions representative of residual liquids were reported earlier. This paper describes conditions and results of recent hot cell tests of the complete Hanford Radionuclide Removal Process. These advanced tests, made with actual residual liquid containing large concentrations of ethylenediaminetetraacetic acid (EDTA) and other organic compounds, provided a rigorous and convincing proof of the process flowsheet.

298 (PNL-TR--415) PROCESS FOR THE DENITRIFICATION OF HIGH-LEVEL RADIOACTIVE LIQUID WASTES. Gattys, F.J. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 10 Jul 1981. Contract AC06-76RL01830. Translation of German(FRG) Patent 2,807,324. 17p. NTIS, PC A02/MF A01. Order Number DE81026875.

A process is claimed for denitrification of high-level concentrated radioactive liquid wastes by using powdered paraformaldehyde. Commercial paraformaldehyde with a low water content and mean particle size smaller than 60 microns is introduced as denitrifying agent at about 110°C into the liquid waste. A buffer of citric acid and phosphate is added during the denitrification reaction. The escaping gases contain nitrogen oxides which are recombined and concentrated to about 40% by weight nitric acid.

299 (PNL-TR--386) FABRICATION AND CHARACTERIZATION OF AN IMPROVED BOROSILICATE GLASS FOR SOLIDIFICATION OF HIGH-LEVEL FISSION-PRODUCT SOLUTIONS (HLW). PART 1. STUDIES OF 25 BOROSILICATE-GLASS PRODUCTS FOR SELECTION OF AN IMPROVED BOROSILICATE GLASS. Guber, W.; Hussain, M.; Kahl, L.; Ondracek, G.; Saidl, J.; Dippel, T. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.)). 4 Feb 1980. Contract EY-76-C-06-1830. Translation of KFK--2721, August 1979 52p. Dep. NTIS, PC A04/MF A01.

The proportion of the glass constituents in the glass were varied. Taking gadolinia-containing waste into account, this approach led to 25 different basis glasses. Using the basic glasses and 15 wt % of inactive HLW oxides in each case, 25 borosilicate-glass products were melted in the laboratory, under conditions approximating the industrial fabrication process. The density, thermal conductivity, thermal expansion, characteristic temperatures, viscosity, specific heat, vaporization losses from the glass melt, electrical conductance, mechanical strength, hydrolytic resistance, crystallization behavior and tendency towards formation of additional phases were determined. On the basis of the individual values obtained, the glass products GP 98/12 (without Gd_2O_3) and GP 98/26 (with Gd_2O_3) were selected as improved matrix glasses for further characterization. The hydrolytic resistance, viscosity at 1420°K, crystallization tendency and tendency towards formation of further phases proved to be the most important.

300 (PNL-TR--377) PRESENT STATUS OF TECHNIQUES USED FOR HIGH-LEVEL LIQUID-WASTE SOLIDIFICATION. Sombret, C. 8 Jan 1980. Contract EY-76-C-06-1830. Translated from Annual meeting of IAEA Technical Committee on Alpha-Bearing Waste, Cadarache, France, October 1979 55p. Dep. NTIS, PC A04/MF A01.

Many nations have pursued research on the solidification of high-level liquid wastes. The different waste forms under consideration are distinguished. Techniques used in making calcinates, glass-metal composites, modified calcinate-metal composites, and glasses (batch and continuous processes) are discussed, particularly the latter. Developments achieved in different countries are summarized. 17 figures. (DLC)

301 (PNL-TR--387) MATERIAL STUDIES ON SOLIDIFICATION OF MEDIUM-LEVEL WASTE SOLUTIONS WITH CEMENT: LEACHABILITY OF CESIUM AND STRONTIUM. Rudolph, G.; Koester, R. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.)). Jan 1980. Contract EY-76-C-06-1830. Translation of KFK--2942, August 1979 50p. Dep. NTIS, PC A03/MF A01.

Two leaching techniques, i.e., the IAEA method and an accelerated test, were used to search for methods of reducing the leachability of cesium and strontium from products formed during cementation of medium-level waste solutions from reprocessing plants. A certain natural calcium bentonite lowers the leachability of cesium by a factor of up to 100. This effect increases with the quantity added. Strontium leachability can be reduced to a smaller extent by addition of a barium silicate hydrate. This effect is found particularly at high temperature. Leaching into a salt brine which is in equilibrium with carnallite and other salts is significantly lower than into distilled water. Excess cesium increases the leachability of cesium. Other parameters such as cement type, curing time, fluidizers and salt content have little or even no influence. Tests on the settlement behavior of cement morters containing simulated waste solution revealed very unfavorable behavior of blast-furnace slag cement.

302 VITRIFICATION OF ACTUAL HIGH-LEVEL WASTE FROM LIGHT WATER REACTOR FUEL. Bjorklund, W.J.; Hanson, M.S. (Battell Pacific Northwest Lab., Richland, WA). pp 396-401 of Proceedings of the 27th conference on remote systems technology. Wallin, D.R. (ed.). La Grange Park, IL; American Nuclear Society, Inc. (1980).

From American Nuclear Society meeting; San Francisco, CA, USA (12 Nov 1979).

The high-level waste from the processing of 1.5 MT of spent light water reactor fuel has been successfully concentrated, dried, and converted to a vitreous product consisting of 94 l of glass contained in two stainless-steel canisters. The spray-calcination process coupled to the in-canister vitrification process as developed at Pacific Northwest Laboratory was used to solidify the waste. An effluent system consisting of a variety of condensation and scrubbing steps gave more than adequate cleanup to the process off-gas before it was released to the atmosphere.

303 DEVITRIFICATION OF NUCLEAR WASTE GLASSES. Turcotte, R.P.; Wald, J.W.; May, R.P. (Battelle Pacific Northwest Labs., Richland, WA). pp 141-146 of Scientific basis for nuclear waste management. Northrup, C.J.M. Jr. (ed.). New York, NY; Plenum Press (1980).

From Materials Research Society annual meeting; Boston, MA, USA (26 Nov 1979).

Devitrification studies of waste glasses are undertaken for a variety of reasons, but mainly to determine processing conditions through which significant crystallization can be avoided. The first detailed kinetic study by Turcotte and Wald (Devitrification Behavior in a Zinc Borosilicate Nuclear waste Glass, PNL-2247, Pacific Northwest Laboratory, March 1978) demonstrated that although behavior is complex, the kinetics follow expected trends, based on simple theoretical models of simple glasses.

304 HIGH-LEVEL WASTE VITRIFICATION PRODUCT CHARACTERIZATION. Bryan, G.H.; Bjorklund, W.J.; Kuhn, W.L. (Battelle Pacific Northwest Labs., Richland, WA). pp 147-154 of Scientific basis for nuclear waste management. Northrup, C.J.M. Jr. (ed.). New York, NY: Plenum Press (1980).

From Materials Research Society annual meeting; Boston, MA, USA (26 Nov 1979).

The Nuclear Waste Vitrification Project (NWVP) was conducted to provide a demonstration of the vitrification of high-level liquid waste (HLLW) from spent fuel discharged from an operating LWR. The scope of NWVP included the design, installation, and operation of a small-scale plant preparing actual HLLW for vitrification, and waste solidification by operation of a previously used 1 tU/day spray calciner in-can melter vitrification system. The HLLW was vitrified in two 20.3-cm-(8-in.-) diameter by 2.44-m-(8-ft-) long canisters. Non-destructive testing of the first canister of high level waste glass has indicated very little in the way of unexpected behavior. The glass appears to be uniform with no large anomalies noted. Non-destructive testing of the second canister is incomplete at this time; however, tests have not indicated any unexpected or unusual results. Further testing by destructive methods (e.g., core drilling for an interior view of the glass within the canister, core sampling for determining density, crystallinity, microscopic homogeneity, mechanical conditions, etc., and leach tests to provide fundamental information about leach kinetics and equilibria, etc.) will yield more specific information on the glass. Until such time as these tests are complete, a full comparison of expected and obtained product cannot be made.

305 STATISTICALLY DESIGNED STUDY OF A NUCLEAR WASTE GLASS SYSTEM. Chick, L.A.; Piepel, G.F.; Gray, W.J.; Mellinger, G.B.; Barnes, B.O. (Battelle Pacific Northwest Labs., Richland, WA). pp 175-181 of Scientific basis for nuclear waste management. Northrup, C.J.M. Jr. (ed.). New York, NY; Plenum Press (1980).

From Materials Research Society annual meeting; Boston, MA, USA (26 Nov 1979).

This paper discusses studies performed on a somewhat restricted glass matrix with 11 oxide components selected for the purpose of verifying the applicability of newly developed statistical design and analysis techniques. Properties measured and modeled for this study include those of the melt at high temperature (shear viscosity, electrical conductivity and volatility) and those of the final product in solid form (devitrification products; weight loss by chemical attack in distilled water, buffered pH 9 slution and buffered pH 4 solution; and chemical analysis of ions leached into distilled water). This paper discusses one processing property (volatility) and one immobilization property (weight loss in pH 4 solution). 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 of complex glasses. General agreement between the component effects predicted by modeling of independent sets of data lends credence to the results. Further work must include crystalline phases as part of the defined compositions so that the effects of components which form the crystals can be better understood.

306 TEMPERATURE DEPENDENCE FOR HYDROTHERMAL REACTIONS OF WASTE GLASSES AND CERAMICS. Westsik, J.H. Jr.; Shade, J.W.; McVay, G.L. (Battelle Pacific Northwest Labs., Richland, WA). pp 239-248 of Scientific basis for nuclear waste management. Northrup, C.J.M. Jr. (ed.). New York, NY; Plenum Press (1980).

From Materials Research Society annual meeting; Boston, MA, USA (26 Nov 1979).

Samples of simulated high-level waste glasses, supercalcine, SYNROC-B and the natural glass were leached three days in deionized water. The test temperatures were 25°, 50°, 100°, 150°, 250° and 350°C, except that the SYNROC was not tested at below 100°C. Each sample was placed in a Hastelloy autoclave with 100 ml of deionized water, giving solution-volume-to-sample-surface-area ratios of between 12 and 17 cm. The autoclaves were pressurized with nitrogen so that the operating pressure at temperature was 13.8 MPa (2000 psi) - except at 350°C, when the pressure was 16.9 MPa (2450 psi). After the tests, the solutions were chemically analyzed using the inductive-coupled plasma spectrometer and the atomic adsorption spectrometer. Each waste form sample was sectioned and examined optically and with the scanning electron microscope with an energy-dispersive x-ray attachment. This study shows that simulated high level waste glasses, supercalcine 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 we expect actual repository temperatures to lie, releases from the waste glasses are lowest. The actual temperature relationship observed depends on the elements studied. The chemistry of the element, changes in the solution properties and secondary reactions all influence the release of elements from the waste forms.

307 ALTERNATIVE WASTE FORMS: A COMPARATIVE STUDY. Rusin, J.M.; Wald, J.W.; Lokken, R.O. (Battelle Pacific Northwest Labs., Richland, WA). pp 255-264 of Scientific basis for nuclear waste management. Northrup, C.J.M. Jr. (ed.). New York, NY; Plenum Press (1980). From Materials Research Society annual meeting; Boston, MA, USA (26 Nov 1979).

A characterization study utilizing comparative tests has been conducted to assess product inertness of alternative waste form materials, having evaluated at this point four basic product types: sintered ceramics, glass ceramics, glass and concrete. The seven specific waste form materials studied represent simulated nuclear waste loading of 5% to 100%, processed between room temperature and 1200°C and subjected to characterization tests including phase analysis, microstructure, compression testing, volatility and leach testing. Significant conclusions based upon the results obtained to date are: sintered calcine waste form PW-9 does not retain Na, Mo and Cs when leached 90°C and, in fact, does not remain a solid; glass and supercalcine are alike under both hydrous and hydrothermal leach conditions with glass exhibiting a greater retention of sodium and molybdenum, supercalcine having a greater retention of cesium, and both forms approximately equal in strontium retention; volatility measurements indicate that an order of magnitude decrease in volatility occurs when a calcine waste form is incorporated in a crystalline or glassy host; glass 76-68 is superior to supercalcine SPC-5B in retention of volatiles below 1100°C because of the high release of Na from SPC-5B, however, as the temperature approaches or exceeds the glass melt temperature, volatile losses of the glass equal or exceed that of SPC-5B; glass 76-68 and supercalcine SPC-5B have high compressive strengths when compared to sintered PW-9 and cement products. This is apparently due to a stronger continuum bond resulting from a glassy matrix or crystalline ingrowth over a simple mechanical agglomeration of particles.

308 PROCESSING OF HIGH-TEMPERATURE SIMULATED WASTE GLASS IN A CONTINUOUS CERAMIC MELTER. Barnes, S.M.; Brouns, R.A.; Hanson, M.S. (Battelle Pacific Northwest Labs., Richland, WA). pp 859-866 of Scientific basis for nuclear waste management. Northrup, C.J.M. Jr. (ed.). New York, NY; Plenum Press (1980). From Materials Research Society annual meeting; Boston, MA, USA (26 Nov 1979).

Recent operations have demonstrated that high-melting-point glasses and glass-ceramics can be successfully processed in joule-heated, ceramic-lined melters with minor modifications to the existing technology. Over 500 kg of simulated waste glasses have been processed at temperatures up to 1410°C. The processability of the two high-temperature waste forms tested is similar to existing borosilicate waste glasses. High-temperature waste glass formulations produced in the bench-scale melter exhibit quality comparing favorably to standard waste glass formulations.

309 DESIGN AND INSTALLATION OF A LAB-SCALE SYSTEM FOR RADIOACTIVE WASTE TREATMENT. Berger, D.N.; Knox, C.A.; Siemens, D.H. (Battelle Northwest Labs., Richland, WA). Transactions of the American Nuclear Society ; 34: 840-842(1980). (CONF-800607--). From American Nuclear Society annual meeting; Las Vegas, NV, USA (8 Jun 1980).

RADIOACTIVE WASTE PROCESSING;BENCH-SCALE EXPERIMENTS;SOLIDIFICATION;HIGH-LEVEL RADIOACTIVE WASTES;PNEUMATICS;MEASURING INSTRUMENTS;HOT CELLS;EQUIPMENT;DESIGN; EVAPORATORS;TANKS;CONDENSERS;SCRUBBERS;PUMPS; REMOTE HANDLING

310 LEACHING TEST METHODS FOR WASTE FORMS. Mendel, J.E.; Turcotte, R.P.; Westsik, J.H. Jr. (Battelle Northwest Labs., Richland, WA). Transactions of the American Nuclear Society ; 34: 193-194(1980). (CONF-800607--). From American Nuclear Society annual meeting; Las Vegas, NV, USA (8 Jun 1980).

RADIOACTIVE WASTE PROCESSING;SOLIDIFICATION; HIGH-LEVEL RADIOACTIVE WASTES;SOLID WASTES; LEACHING;MEASURING METHODS

311 EVALUATING RADIATION-INDUCED CHANGES IN HIGH-LEVEL WASTE SOLIDS. Roberts, F.P. (Battelle Northwest Labs., Richland, WA). Transactions of the American Nuclear Society ; 34: 194-195(1980). (CONF-800607--). From American Nuclear Society annual meeting; Las Vegas, NV, USA (8 Jun 1980).

HIGH-LEVEL RADIOACTIVE WASTES;RADIOACTIVE WASTE PROCESSING;SOLIDIFICATION;CHEMICAL RADIATION EFFECTS;SOLID WASTES;GAMMA RADIATION; BETA SOURCES;ALPHA SOURCES;COMPARATIVE EVALUATIONS

312 REVIEW OF WASTE FORMS. Rusin, J.M. (Battelle Northwest Labs., Richland, WA). Transactions of the American Nuclear Society ; 34: 196-197(1980). (CONF-800607--). From American Nuclear Society annual meeting; Las Vegas, NV, USA (8 Jun 1980).

RADIOACTIVE WASTE PROCESSING;SOLIDIFICATION; VITRIFICATION;GLASS;CERAMICS;CALCINED WASTES; ENCAPSULATION;HIGH-LEVEL RADIOACTIVE WASTES; CERAMIC MELTERS;COMPARATIVE EVALUATIONS

313 CHARACTERISTICS OF SOME TRANSURANIUM-CONTAMINATED SOLID WASTES. Bryan, G.H.; Palmer, C.R. (Battelle Northwest Labs., Richland, WA). Transactions of the American Nuclear Society ; 34: 397-398(1980). (CONF-800607--). From American Nuclear Society annual meeting; Las Vegas, NV, USA (8 Jun 1980).

RADIOACTIVE WASTE PROCESSING;HIGH-LEVEL RADIOACTIVE WASTES;TRANSURANIUM ELEMENTS; SLUDGES;COMBUSTION;IRON HYDROXIDES; PRECIPITATION;FLOCCULATION;FILTRATION;ASHES; CHEMICAL COMPOSITION;PHYSICAL PROPERTIES;ROCKY FLATS PLANT;LASL;ALPHA-BEARING WASTES

314 DEVELOPMENT OF A REMOTE LAB-SCALE WASTE TREATMENT FACILITY. Knox, C.A.; Hanson, M.S. (Battelle Pacific Northwest Labs., Richland, WA). Transactions of the American Nuclear Society ; 34: 398-400(1980). (CONF-800607--).

From American Nuclear Society annual meeting; Las Vegas, NV, USA (8 Jun 1980).
RADIOACTIVE WASTE PROCESSING;HIGH-LEVEL RADIOACTIVE WASTES;RADIOACTIVE WASTE FACILITIES; SOLIDIFICATION;BENCH-SCALE EXPERIMENTS; EQUIPMENT;DESIGN;EVAPORATORS;TANKS;CONDENSERS; CYCLONE SEPARATORS;SCRUBBERS;FILTERS; PERFORMANCE TESTING;RADIOACTIVE EFFLUENTS

315 DECOMMISSIONING A URANIUM FUEL PRODUCTION PLANT. Elder, H.K.; Plahnik, D.E. (Battelle Pacific Northwest Labs., Richland, WA). Transactions of the American Nuclear Society ; 34: 430-432(1980). (CONF-800607--).
From American Nuclear Society annual meeting; Las Vegas, NV, USA (8 Jun 1980).
DECOMMISSIONING;FUEL FABRICATION PLANTS; PLANNING;DECONTAMINATION;TRANSPORT;CLEANING; INSPECTION;COST;SAFETY;RADIATION DOSES

316 SOLID STATE CONTAINMENT OF NOBLE GASES IN SPUTTER DEPOSITED METALS AND LOW DENSITY GLASSES. Tingey, G.L.; McClenahan, E.D.; Bayne, M.A.; Gray, W.J. (Battelle Pacific Northwest Labs., Richland, WA (USA)). pp 279-290 of Management of gaseous wastes from nuclear facilities. Proceedings of an international symposium jointly organized by the IAEA and the NEA of the OECD and held in Vienna, 18-22 February 1980. Vienna, Austria; IAEA (1980).
From International symposium on management of gaseous wastes from nuclear facilities; Vienna, Austria (18 Feb 1980).
Two techniques have been developed for the solid state storage of the radioactive inert gases. The first technique involves ion implantation of gaseous ions during the sputter deposition of a metal matrix. This technique yields a deposited metal containing the implanted gas atomically dispursed throughout the matrix. Relatively pure metals such as Fe, Al, Ni, and Ti have been sputtered with Kr to yield a crystalline product containing up to 5 at.%Kr. Metal alloys, which sputter to yield a glassy structure, have been shown to accept a Kr atom more readily and contain up to 10 at.%Kr. One of the more promising glassy alloys studied has an empirical formula Fesub(0.79)Ysub(0.12)Krsub(0.09). Long-term release measurements indicate that less than 2% of the Kr would be released in 10 years at 300°C. Other metal alloys show even lower Kr release rates. A second process under investigation at our laboratory for storing radioactive inert gases involves dissolution and/or trapping in the very fine pores of a low density glass matrix. Although the quantity dissolved in fully dense glasses is so small as not to be of interest, the same process yields up to 7 cm^3 of Kr(STP)/g of glass (approximately 2 at.%) in glasses with about 30% open porosity when loaded at about 35 MPa pressure.

317 BEHAVIOUR OF SELECTED CONTAMINANTS IN SPRAY CALCINER/IN-CAN MELTER WASTE VITRIFICATION OFF-GAS. Hanson, M.S.; Goles, R.W.; Hamilton, C.C. (Battelle Pacific Northwest Labs., Richland, WA (USA)). pp 371-390 of Management of gaseous wastes from nuclear facilities. Proceedings of an international symposium jointly organized by the IAEA and the NEA of the OECD and held in Vienna, 18-22 February 1980. Vienna, Austria; IAEA (1980).
From International symposium on management of gaseous wastes from nuclear facilities; Vienna, Austria (18 Feb 1980).
Product loss from spray calciner/in-can melter vitrification of high-level wastes was evaluated with respect to volatile, gaseous and particulate materials. Investigations of the off-gases in a non-radioactive system are discussed, including gaseous constituents, particulate size distributions and loadings. Monitoring of gases leaving the off-gas system during spray calcination/in-can melting of radioactive waste gave material concentrations and material forms in the gases. The most significant conclusion drawn from these studies was that particulate loss accounts for a significant portion of the fission products in the off-gas system.

318 FACILITATION OF DECOMMISSIONING LIGHT WATER REACTORS. Moore, E.B. Jr. (Battelle Pacific Northwest Labs., Richland, WA (USA)). pp 57-68 of Proceedings of the specialist meeting on decommissioning requirements in the design of nuclear facilities, Paris, 17-19 March 1980. Paris, France; OECD (1980).
From NEA specialist meeting on decommissioning requirements in the design of nuclear facilities; Paris, France (17 Mar 1980).
Information on design features, special equipment, and construction methods useful in the facilitation of decommissioning light water reactors is presented in this paper.

319 ENGINEERING-SCALE VITRIFICATION OF COMMERCIAL HIGH-LEVEL WASTE. Bonner, W.F.; Bjorklund, W.J.; Hanson, M.S.; Knowlton, D.E. (Battelle, Pac Northwest Lab, Richland, Wash). Proceedings of the Symposium on Waste Management; 2: 339-363(1980). (CONF-800313--).
From Waste management conference; Tucson, AZ, USA (10 Mar 1980).
The feasibility of radioactive waste solidification was demonstrated in the Waste Solidification Engineering Prototypes program between 1966 and 1970 using simulated power-reactor waste composed of nonradioactive chemicals and high-level waste (HLW) from spent, Hanford reactor fuel. Thirty-three engineering-scale canisters of solidified HLW were produced during the operations. In early 1979, the Nuclear Waste Vitrification Project (NWVP), demonstrated the vitrification of HLW from the processing of actual commercial nuclear fuel. This program consisted of two parts, waste preparation and vitrification by spray calcination and in-can melting. This paper discusses results from the NWVP.

320 ALTERNATIVES FOR VITRIFICATION OF EXISTING COMMERCIAL HIGH-LEVEL WASTE BY SPRAY CALCINATION/IN-CAN MELTING. Holton, L.K.; Larson, D.E.; Mellinger, G.B.; Nelson, T.A. (Battelle, Pac Northwest Lab, Richland, Wash). Proceedings of the Symposium on Waste Management; 2: 365-380(1980). (CONF-800313--).
From Waste management conference; Tucson, AZ, USA (10 Mar 1980).
Three vitrification flowsheet options have been investigated for immobilization of existing commercial high-level waste stored at the Western New York Service Center. Laboratory glass development studies and process feasibility studies using the Spray Calciner/In-Can Melter process were explored. The Spray Calciner/In-Can Melting (SC/ICM) Process converts high-level waste into a borosilicate glass for long-term storage of the radioactive material. This process is a candidate process for immobilization of this commercial waste; however to date the specific waste form and process for waste immobilization has not been chosen. Initial results obtained from the

laboratory investigations and the pilot plant testing using nonradioactive chemicals are reported in this paper.

321 COMPARATIVE STUDY OF ALTERNATIVE HIGH-LEVEL WASTE SOLIDIFICATION PROCESSES. Treat, R.L. (Battelle, Pac Northwest Lab, Richland, Wash). Proceedings of the Symposium on Waste Management; 2: 503-514(1980). (CONF-800313--).

From Waste management conference; Tucson, AZ, USA (10 Mar 1980).

The paper discusses basic requirements for nine high-level waste solidification processes. These processes are: in-can glass melting process, joule-heated glass melting process, glass-ceramic process, marbles-in-lead matric process, coated supercalcine pellets-in-lead matrix process, SYNROC process, titanate process, concrete process, and cermet process.

322 REMOVAL OF RADIONUCLIDES FROM THE WATER-SOLUBLE FRACTION OF HANFORD NUCLEAR DEFENSE WASTES. Strachan, D.M.; Schulz, W.W. (Battelle, Pac Northwest Lab, Richland, Wash). Proceedings of the Symposium on Waste Management; 2: 551-567(1980). (CONF-800313--).

From Waste management conference; Tucson, AZ, USA (10 Mar 1980).

The current Hanford Waste Management Program has operated since 1968 to remove the bulk of the long-lived heat emitters /sup 90/Sr and /sup 137/Cs from stored high-level wastes. The liquid waste remaining after removal of /sup 90/Sr and /sup 137/Cs is returned to underground tanks for eventual evaporation to damp solid salt cake. Approximately 95,000 m/sup 3/ of salt cake and 49,000 m/sup 3/ of ''sludge'' will eventually accumulate in approximately 50 underground single-shell tanks. One alternative for long-term management of high-level Hanford wastes involves retrieval, after a yet-to-be determined interim storage time, conversion to more immobile forms, and terminal storage in a suitable geologic repository. Another alternative for long-term management of salt cake and residual liquid involves removing most of the long-lived radionuclides and many of the shorter-lived ones from these wastes. This paper describes conditions and results of recent hot cell tests of the complete Hanford Radionuclide Removal Process. These advanced tests, made with actual residual liquid containing large concentrations of ethylenediaminetetracetic acid (EDTA) and other organic compounds, provided a rigorous and convincing proof of the process flowsheet. 16 refs.

323 SLURRY FEEDING OF NUCLEAR WASTES TO AN ELECTRIC GLASS MELTER. Buelt, J.L.; Chapman, C.C. (Battelle Northwest Labs., Richland, WA). Transactions of the American Nuclear Society ; 32: 393-394(Jun 1979). (CONF-790602--(Summ.)).

From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).

JOULE HEATING;CERAMICS;HEATERS;SLURRIES; MATERIALS HANDLING;EQUIPMENT;LIQUID WASTES; RADIOACTIVE WASTE PROCESSING;SOLIDIFICATION; VITRIFICATION

324 DEVELOPMENT OF MULTIBARRIER NUCLEAR WASTE FORMS. Rusin, J.M. (Battelle Pacific Northwest Labs., Richland, WA); Browning, M.F.; McCarthy, G.J. pp 169-180 of Scientific basis for nuclear waste management. McCarthy, G.J. (ed.). New York, NY; Plenum Press (1979).

From Annual meeting of Materials Research Society; Boston, MA, USA (28 Nov 1978).

The multibarrier concept aims to separate the radionuclide-containing inner core material and the environment by the use of coatings and metal matrices. The resultant composite waste form exhibits enhanced inertness due to improved thermal stability and mechanical strength, and the added barriers greatly improve leach resistance. Two options were developed for the inner core of the multibarrier concept: supercalcine pellets and glass marbles. The development of coatings for supercalcine was pursued to provide an additional protective layer between the radionuclides and the environment. The coatings include the application of a 40 µm pyrolytic carbon layer as a barrier to enhance leach resistance and a 60-µm Al_2O_3 layer as a barrier to increase oxidation resistance. Glass coating of supercalcine by frit and glaze was also investigated.

325 REVIEW OF HIGH-LEVEL WASTE-FORM PROPERTIES. Rusin, J.M.; McElroy, J.L. (Battelle Pacific Northwest Labs., Richland, WA). Transactions of the American Nuclear Society ; 33: 418-419(1979). (CONF-791103--).

From American Nuclear Society meeting; San Francisco, CA, USA (12 Nov 1979).

Sintered ceramics, concrete, glass-ceramics, hot-pressed ceramics, and metal matrices waste forms evaluated. RADIOACTIVE WASTE PROCESSING; SOLIDIFICATION;HIGH-LEVEL RADIOACTIVE WASTES; GLASS;CERAMICS;CALCINATION;EVALUATION;CONCRETES; SOLID WASTES

326 DIRECT DENITRATION OF PLUTONIUM NITRATE-NITRIC ACID SOLUTION IN A SCREW CALCINER. Bryan, G.H.; Wheelwright, E.J.; Browne, L.M. (Battelle Pacific Northwest Labs., Richland, WA). Transactions of the American Nuclear Society ; 33: 468(1979). (CONF-791103--).

From American Nuclear Society meeting; San Francisco, CA, USA (12 Nov 1979).

PLUTONIUM NITRATES;DENITRATION;NITRIC ACID; RADIOACTIVE WASTE PROCESSING;CALCINATION

327 GLASS: AN AVAILABLE MATERIAL FOR THE IMMOBILIZATION OF NUCLEAR WASTE. Mendel, J.E. (Battelle Pacific Northwest Labs., Richland, WA). Transactions of the American Nuclear Society ; 33: 278(1979). (CONF-791103--).

From American Nuclear Society meeting; San Francisco, CA, USA (12 Nov 1979).

GLASS;RADIOACTIVE WASTE PROCESSING; SOLIDIFICATION;STABILITY;RADIATION EFFECTS; LEACHING;MATERIALS;COMPATIBILITY

328 VITRIFICATION OF ACTUAL HIGH-LEVEL WASTE FROM LWR FUEL. Bjorklund, W.J.; Hanson, M.S. (Battelle Pacific Northwest Labs., Richland, WA). Transactions of the American Nuclear Society ; 33: 943-944(1979). (CONF-791103--).

From American Nuclear Society meeting; San Francisco, CA, USA (12 Nov 1979).

RADIOACTIVE WASTE PROCESSING;SOLIDIFICATION; VITRIFICATION;HIGH-LEVEL RADIOACTIVE WASTES; DEMONSTRATION PROGRAMS

329 BACKGROUND ON LEACH TESTING OF SOLIFIFIED RADIOACTIVE WASTES. Mendel, J.E. (Batelle Pacific Northwest Labs., Richland, WA). Transactions of the American Nuclear Society ; 33: 203(1979). (CONF-791103--).

From American Nuclear Society meeting; San Francisco, CA, USA (12 Nov 1979).

LOW-LEVEL RADIOACTIVE WASTES;LEACHING;SOLID WASTES;STANDARDS;RADIOACTIVE WASTE DISPOSAL; SAFETY

330 PROCESSES FOR PRODUCTION OF ALTERNATIVE WASTE FORMS. Ross, W.A.; Rusin, J.M.; McElroy, J.L. (Battelle, Pac Northwest Lab, Richland, Wash). Proceedings of the Symposium on Waste Management; 335-348(1979). (CONF-790204--).
From Symposium on waste management and fuel cycles 1979; Tucson, AZ, USA (28 Feb 1979).
Various processes and operations for radioactive waste forms are described, and a simple system is proposed for making comparisons. This system suggests that one goal for processes would be to reduce the number of process steps, thereby providing less complex processing systems. Among systems examined are these: crystalline waste forms, fusion-ceramic processes, cermets, and cements. 17 refs.

331 COMPARISON OF BITUMEN AND CEMENT IMMOBILIZATION OF INTERMEDIATE- AND LOW-LEVEL RADIOACTIVE WASTE. Voss, J.W. (Battelle, Pac Northwest Lab, Richland, Wash). Proceedings of the Symposium on Waste Management; 557-582(1979).
The Bitumen Immobilization Facility (BIF) is projected to produce 2530 drums/plant-year, while the Cement Immobilization Facility (CIF) will produce 5590 drums/plant-year. The capacity factor, the fraction of the time the facility operated during any year, of the CIF is 0.21, while that of the BIF is 0.36. The secondary radioactive wastes for the facilities are nearly equal except for the large volume of process distillate generated by the BIF. The utility requirements for the BIF are greater. The manpower requirements for the BIF and the CIF are equal. The economics of the two facilities are nearly identical. Safety differences slightly favor the CIF. 15 refs.

332 FULL-SCALE IN-CAN MELTING FOR VITRIFICATION OF NUCLEAR WASTES. Blair, H.T. (Battelle Northwest Lab., Richland, WA). Transactions of the American Nuclear Society ; 28: 355-356(Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622--.
RADIOACTIVE WASTE PROCESSING;SOLIDIFICATION; VITRIFICATION;HIGH-LEVEL RADIOACTIVE WASTES; LIQUID WASTES;EQUIPMENT

333 CONTINUOUS GLASS MELTER FOR IMMOBILIZATION OF RADIOACTIVE WASTE. Chapman, C.C.; Buelt, J.L. (Battelle Northwest Lab., Richland, WA). Transactions of the American Nuclear Society ; 28: 356-357(Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622--.
RADIOACTIVE WASTE PROCESSING;SOLIDIFICATION; GLASS;VITRIFICATION;EQUIPMENT;LIQUID WASTES; CHEMICAL REACTORS

334 CHARACTERISTICS OF HIGH-LEVEL RADIOACTIVE WASTE GLASSES. Ross, W.A. (Battelle Northwest Lab., Richland, WA). Transactions of the American Nuclear Society ; 28: 357-358(Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622--.
RADIOACTIVE WASTE PROCESSING;SOLIDIFICATION; GLASS;HIGH-LEVEL RADIOACTIVE WASTES;LEACHING; RADIATION EFFECTS;TEMPERATURE EFFECTS

335 RECENT ADVANCES IN SPRAY CALCINATION OF NUCLEAR WASTE. Mikols, W.J.; Romero, L.S.; Bonner, W.F. (Battelle Northwest Lab., Richland, WA). Transactions of the American Nuclear Society ; 28: 354-355(Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622--.
RADIOACTIVE WASTE PROCESSING;SOLIDIFICATION; CALCINATION;LIQUID WASTES

336 (BNWL-tr--315) TREATMENT OF RADIOACTIVE α RESIDUE AT MOL CEN-SCK (BELGIUM). Van de Voorde, N. (Centre d'Etude de l'Energie Nucleaire, Mol (Belgium)). May 1978. Contract EY-76-C-06-1830. Translated from pp 221-236 of (Problable) NEA IAEA Tech Seminar, Paris, 12/1977 21p. Dep. NTIS, PC A02/MF A01.
At the present time, these wastes--both flammable and nonflammable--are treated by compacting or cutting, then encased in hot asphalt to form a compact homogeneous mass which normally prevents any dispersion of toxic substances. This treatment is carried out in a special work area, designed so that radiotoxic aerosols produced in the treatment process remain confined to that area, which is possible because of a very efficient device which sweeps this area by means of a ventilation system keeping it at lower barometric pressure than the outside, under all circumstances. The staff works in diver suits. However, in view of the range of the problem and the need to establish the strictest standards of residue treatment, a new procedure has been adopted. It simultaneously solves the problems of volume and the final product, which has ideal properties for permanent storage. This procedure, called the complete system, consists of incinerating the residue at a very high temperature (1500°C), most of which is flammable, at the same time as selected nonflammable residue such as glass, metal, sludge, ion exchangers, etc. The incineration residue melts at that temperature, forming an insoluble granitic mass. This process is carried out in a special furnace equipped with a gas purification device and installed in a work area similar to that provided for treatment of α residue, in which the staff must also work in diving suits.

337 (BNWL-tr--289) TREATMENT OF GASEOUS EFFLUENTS IN FISSION-PRODUCT VITRIFICATION INSTALLATIONS. Bonniaud, R.; Jouan, A.; Laude, F.; Sombret, C. Jan 1978. Contract EY-76-C-06-1830. Translation of French report 37p. Dep. NTIS, PC A03/MF A01.
The purification methods used make it possible to achieve decontamination factors of 10^{10} for the gases, even in the worst cases. They also enable liquid wastes to be limited to a maximum degree, either by continuous direct recycling in the process or by indirect recycling by separate evaporation. Thus, in all instances it is technically feasible to adhere to established standards on wastes and to make vitrification processes into suitable techniques for the treatment of fission-product solutions.

338 DECONTAMINATION AND MELT DENSIFICATION OF FUEL HULL WASTES. Dillon, R.L. (Battelle Pacific Northwest Labs., Richland, Wash. (USA)). pp 299-312 of Traitement, conditionnement et stockage des dechets solides alpha et des coques de degainage. Paris, France: Organization for Economic Co-Operation and Development (1978).
From Technical seminar on the treatment, conditioning and storage of solid alpha-bearing

waste and cladding hulls; Paris, France (5 Dec 1977).

Ongoing work in the United States on treatment of cladding hull residues from nuclear fuel dissolution is described. The report proceeds to the description of the Pacific Northwest Laboratories melt densification and decontamination program. On a laboratory scale, a two-step process, initial treatment with 600°C HF-Ar mixtures followed by aqueous decontaminating, has produced effective decontamination. The Inductos lag process has been selected for densification of the predominately zircaloy waste stream. The process provides for direct melting of the low density fuel hull residue without intermediate preparation of a consumable electrode. Current efforts are addressed to the design and fabrication of decontamination and melting equipment to be installed in a hot cell and remote.

339 IMPACTS OF REACTOR-INDUCED DEFECTS ON SPENT FUEL STORAGE. Johnson, A.B. Jr. (Battelle Pacific Northwest Labs., Richland, WA (USA)). pp 235-253 of Compte rendu du seminaire de l'AEN sur le stockage des elements combustibles irradies. Paris, France; OECD (1978).

From Seminar on the storage of spent fuel elements; Madrid, Spain (Jun 1978).

Defects arise in the fuel cladding on a small fraction of fuel rods during irradiation in water-cooled power reactors. Defects from mechanical damage in fuel handling and shipping have been almost negligible. No commercial water reactor fuel has yet been observed to develop defects while stored in spent fuel pools. In some pools, defective fuel is placed in closed canisters as it is removed from the reactor. However, hundreds of defective fuel bundles are stored in numerous pools on the same basis as intact fuel. Radioactive species carried into the pool from the reactor coolant must be dealt with by the pool purification system. However, additional radiation releases from the defective fuel during storage appear to be minimal, with the possible exception of fuel discharged while the reactor is operating (CANDU fuel). Over approximately two decades, defective commercial fuel has been handled, stored, shipped and reprocessed.

340 (BNWL-tr--282) SET-UP AND START-UP OF THE BATCH-OPERATED VERA DENITRATION PLANT. Kartes, H.; Koschorke, H.; Kaufmann, F. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.)). Dec 1977. Contract EY-76-C-06-1830. Translation of KFK--2446 21p. Dep. NTIS, PC A02/MF A01.

At the start of 1976, a plant for denitration of high and medium activity fission-product solutions was set into inactive operation at the Karlsruhe Nuclear Research Center. The plant and its individual components are presented. The operation and start-up are described in detail. Furthermore, the measuring methods important for monitoring the process sequence are explained. The procedure for denitration of HAW with a molarity of 1.5 is described in detail. A failure analysis is run in this connection.

341 (BNWL-tr--274) GLASS AS THE FIRST BARRIER FOR LONG-TERM STORAGE OF HIGHLY ACTIVE RADIOACTIVE WASTES. Laude, F. 24 Oct 1977. Contract EY-76-C-06-1830. Translation of French report (CONF-770565--4). 34p. Dep. NTIS, PC A03/MF A01.

From Workshop on risk analysis and geologic modelling; Ispra, Italy (23 May 1977).

The glasses selected for this use are of siliceous composition, and ought to have adequate properties of nuclear, chemical and thermal stability. After a period of artificial storage at the surface to allow cooling, the glass blocks are normally buried in an appropriate geological site, where decay will be completed. To be able to evaluate the risks that activity will spread in the environment as soon as these blocks are buried, it is of paramount importance to know the characteristics of these glasses as well as the influence of time on their main properties. This report gives the first parts of the answer, by referring to studies carried out by the CEA at Marcoule on the properties of glasses. The main results are presented separately depending on whether the glasses contain actinides or not.

342 (BNWL-tr--263) MANAGEMENT OF HIGH-ACTIVITY NUCLEAR WASTE (FISSION PRODUCTS). Redon, A.; Mamelle, J.; Chambon, M.; Giraud, B. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 13 Sep 1977. Contract EY-76-C-06-1830. Translation of IAEA-CN-36/226 (CONF-770505--211(trans)). 20p. Dep. NTIS, PC A02/MF A01.

From International conference on nuclear power and its fuel cycles; Salzburg, Austria (2 May 1977).

The problem of high-activity radioactive waste has considerable psychological importance, and the solutions to be applied must have a high degree of safety for a very long period of time. After a review of definitions, the authors define some principles for conditioning fission product solutions by vitrification. The accent is on the characteristics of the product obtained, research and development efforts, and storage conditions (transitory). Finally, the passage to the industrial stage is described.

343 (BNWL-tr--309) SEPARATION OF CESIUM AND STRONTIUM WITH ZEOLITES. Kanno, T.; Hashimoto, H. Translated from Tohoku Daigaku Senko Seiren Kenkyusho Iho ; 32: No. 1, 14-21(1976). Contract EY-76-C-06-1830. 22p. Dep. NTIS, PC A02/MF A01.

Basic studies of separation of cesium and strontium were made with specimens of zeolite, including the synthetic zeolites A, X and Y, synthetic mordenite, natural modernite and clinoptilolite. Ammonium chloride, which is considered to be the most suitable eluent for alkaline chlorides, was used as the eluent. Cesium was readily eluted from zeolites A and X by ammonium chloride solution, but was eluted with difficulty from synthetic mordenite, natural mordenite and clinoptilolite. On the other hand, strontium was readily eluted from synthetic mordenite, natural mordenite and clinoptilolite by ammonium chloride solution, but was eluted with difficulty from zeolites A and X. The zeolite Y was the only one of these zeolites from which both cesium and strontium were readily eluted by ammonium chloride solution. Strontium could be separated from cesium with zeolites by formation of Sr-EDTA chelates at pH levels above 11. In this process, only cesium was exchanged in the zeolite column, while strontium flowed out from the column.

Waste Disposal and Storage

344 (ARH-CD--550) SEALED STORAGE CASK CONCEPT. CASK CLOSURE CONCEPT EVALUATION. INTERIM REPORT. Moore, E.L.; O'Brien, W.H.

(Battelle Pacific Northwest Labs., Richland, WA (USA)). [nd]. Contract EY-76-C-06-2130. 100p. Dep. NTIS $5.00.

A program was conducted to evaluate four candidate storage cask closure configurations. Four major areas were evaluated to provide a basis for selection of an acceptable closure(s). These were: (1) weldability, (2) suitability of available nondestructive examination methods for determining weld soundness, (3) closure configuration structural soundness, and (4) economics. A relative ranking of each closure configuration was made using the study results. The corner weld configuration was ranked as the most desirable, followed by the threaded plug and seal weld, the circumferential butt-weld and the multiplate in order of decreasing desirability. This report recommends that the structural analysis be continued on the first and second ranked closures to provide a quantitative evaluation capability of their structural integrity and that ultrasonic examination of the corner weld closure configuration be demonstrated. A remote demonstration of welding and nondestructive examination capability of the corner weld is also recommended.

345 (BNL--28845) PLUTONIUM LEACHABILITY FROM ALTERNATIVE TRANSURANIC INCINERATOR ASH WASTE FORMS. Neilson, R. Jr.; Colombo, P.; Bradley, D. (Brookhaven National Lab., Upton, NY (USA); Battelle Pacific Northwest Labs., Richland, WA (USA)). 1980. Contract AC02-76CH00016. 12p. (CONF-800611--11; IAEA-SM--246/45). NTIS, PC A02/MF A01.

From International symposium on the management of alpha-contaminated wastes; Vienna, Austria (2 Jun 1980).

Leaching experiments were conducted to determine the rate of plutonium release from Portland cement, urea-formaldehyde, and polyester-styrene waste forms incorporating incinerator ash waste. A modified IAEA leach test procedure was employing using demineralized water, simulated WIPP Brine B, simplified sodium dominated groundwater, simplified calcium dominated groundwater and simplified bicarbonate dominated groundwater leachants. The data obtained provided a good fit to a diffusion release model for semi-infinite media. This model allows the calculation of effective diffusivities for plutonium release and provides a means for the prediction of long-term plutonium releases from full-scale waste forms. The effective diffusivities determined for Portland cement and polyester-styrene waste forms varied from 1.6×10^{-22} to 3.9×10^{-20} cm^2/sec. Plutonium release was more rapid from urea-formaldehyde waste forms which exhibited effective diffusivities of 2.3×10^{-18} to 1.1×10^{-14} cm^2/sec. The lowest release rates were obtained for leaching in WIPP Brine B. Effective diffusivities in the range of 10^{-22} to 10^{-20} cm^2/sec result in predicted fraction plutonium releases of 1.9×10^{-6} to 1.9×10^{-5} in 10^5 years (neglecting decay) from 210 liter (55 gallon drum) waste forms. As a result of the low effective diffusivities determined and for the long half-lives of TRU radionuclides, waste form stability may be the primary determinant of activity release over the time period that must be considered for TRU waste disposal.

346 (BNWL--181) BURIED RADIOACTIVE WASTE STORAGE TANK TEMPERATURES AND SOIL TEMPERATURES NEAR LEAKS. Jansen, G. Jr.; Willingham, W.E.; DeMier, W.V. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Mar 1966. 69p. NTIS.

Temperatures and heat losses around waste tanks containing boiling solutions are compared for surrounding soil under dry conditions (low thermal conductivity) and under wet conditions (high thermal conductivity). Temperatures that might be attained in sludge-containing tanks resulting from liquid removal are also considered along with heat associated with leaks. A computer program is included for calculating heat rise around leaks.

347 (BNWL--1997) PUBLIC VALUES ASSOCIATED WITH NUCLEAR WASTE DISPOSAL. Maynard, W.S.; Nealey, S.M.; Hebert, J.A.; Lindell, M.K. (Battelle Human Affairs Research Center, Seattle, WA (USA)). Jun 1976. Contract EY-76-C-06-1830. 181p. Dep. NTIS, PC A09/MF A01.

This report presents the major findings from a study designed to assess public attitudes and values associated with nuclear waste disposal. The first objective was to obtain from selected individuals and organizations value and attitude information which would be useful to decision-makers charged with deciding the ultimate disposal of radioactive waste materials. A second research objective was to obtain information that could be structured and quantified for integration with technical data in a computer-assisted decision model. The third general objective of this research was to test several attitude-value measurement procedures for their relevance and applicability to nuclear waste disposal. The results presented in this report are based on questionnaire responses from 465 study participants. (LK)

348 (BNWL--2117, pp 1-9) U.S. GEOLOGIC STORAGE PROGRAM. Burkholder, H.C. 1976.

From Conference on actinide sediment reaction; Seattle, WA, USA (10 Feb 1976).

Outline and diagrams only. RADIOACTIVE WASTE DISPOSAL;UNDERGROUND DISPOSAL;USA;GEOLOGIC DEPOSITS

349 (BNWL--2381) RADIOLYTIC GAS GENERATION FROM CALCINED NUCLEAR WASTE. Tingey, G.L.; Felix, W.D. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Aug 1977. Contract EY-76-C-06-1830. 20p. Dep. NTIS, PC A02/MF A01.

The gas generation rates for both the gamma and alpha radiolysis experiments appear to be about the same, but there was a wider range of products from the α-radiolysis experiment. Although the major product was oxygen, significant quantities of gas condensable at -78°C, probably water, and fairly large fractions of nitrogen oxides are produced. By contrast, the gamma radiolysis experiment yielded nearly 100 percent O_2. A steady state level should be achieved before all of the NO_3^- salts are decomposed because of the reverse reaction $NO_2^- + {}^1/_2O_2 \rightarrow NO_3^-$. The rate of pressure increase is still substantial at a pressure well above 1 atmosphere. Furthermore, in practice the canister will probably be maintained at higher temperature; so the pressure will be further increased by the PV/T relationship. It is concluded that radiolysis of NO_3^- or H_2O will pose no insurmountable problems in the shipping and storing of waste calcines even though some additional pressurization will occur from the decomposition of residual nitrate salts. Release of gas by thermal decomposition or thermal desorption will produce gas pressures much more rapidly than radiolysis if the calcine is subjected to high temperatures from accident or fire. Such conditions are expected to lead to pressures of hundreds of atmospheres in closed containers. (DLC)

350 (BNWL-SA--6068) SAFETY ASSESSMENT OF GEOLOGIC REPOSITORIES FOR NUCLEAR WASTE. Bartlett, J.W.; Burkholder, H.C.; Winegardner, W.K. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 1977. Contract EY-76-C-06-1830. 27p. (CONF-770625--5). Dep. NTIS, PC A03/MF A01.

From International conference on nuclear systems reliability engineering and risk assessment; Gatlinburg, TN, USA (20 Jun 1977).

Consideration of geologic isolation for final disposition of radioactive wastes has led to the need for evaluation of the safety of the concept. Such evaluations require consideration of factors not encountered in conventional risk analysis: consequences at times and places far removed from the repository site; indirect, complex, and alternative pathways between the waste and the point of potential consequences; a highly limited data base; and limited opportunity for experimental verification of results. R and D programs to provide technical safety evaluations are under way. Three methods are being considered for the probabilistic aspects of the evaluations: fault tree analysis, repository simulation analysis, and system stability analysis. Nuclide transport models, currently in a relatively advanced state of development, are used to evaluate consequences of postulated loss of geologic isolation. This paper outlines the safety assessment methods, unique features of the assessment problem that affect selection of methods and reliability of results, and available results. It also discusses potential directions for future work.

351 (BNWL-SA--6158) WASTE ISOLATION SAFETY ASSESSMENT PROGRAM. Dau, G.J. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 1977. Contract EY-76-C-06-1830. 7p. (CONF-770221--2). Dep. NTIS, PC A02/MF A01.

From EPRI fuel cycle risk assessment workshop; Palo Alto, CA, USA (9 Feb 1977).

Objective is to assess the safety of the long-term disposal of high-level radioactive waste in a geologic formation. The program is divided into five tasks: safety assessment concepts and methods, disruptive event analysis, source characterization, transport modeling, and nuclide transport data. Goals and planned work are listed for each task. (DLC)

352 (BNWL-SA--6310) SAFETY ASSESSMENT AND GEOSPHERE TRANSPORT METHODOLOGY FOR THE GEOLOGIC ISOLATION OF NUCLEAR WASTE MATERIALS. Burkholder, H.C.; Stottlemyre, J.A.; Raymond, J.R. (Battelle Pacific Northwest Labs., Richland, WA (USA)). May 1977. Contract EY-76-C-06-1830. 16p. (CONF-770565--5). Dep. NTIS, PC A02/MF A01.

From Workshop on risk analysis and geologic modelling; Ispra, Italy (23 May 1977).

As part of the National Waste Terminal Storage Program in the United States, the Waste Isolation Safety Assessment Program (WISAP) is underway to develop and demonstrate the methods and obtain the data necessary to assess the safety of geologic isolation repositories and to communicate the assessment results to the public. This paper reviews past analysis efforts, discusses the WISAP technical approach to the problem, and points out areas where work is needed. The computer code GETOUT II, which models the nuclide transport in geologic media, is described. (DLC)

353 (BNWL-SA--6382) GLASS WASTE FORMS FOR RADIOACTIVE WASTE CONTAINMENT. Ross, W.A.; Mendel, J.E. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 1977. Contract EY-76-C-06-1830. 30p. (CONF-771102--28). Dep. NTIS, PC A03/MF A01.

From 70. annual AIChE meeting; New York, NY, USA (13 Nov 1977).

Glass appears to be a promising medium for storing and disposing of high-level radioactive waste. The waste can be readily vitrified into a chemically durable material that appears to be resistant to potential changes caused by the self-heating nature and the contained radioactivity of the waste. Investigations are continuing but the evidence to date indicates that the vitrified product is a highly stable material well-suited for geologic isolation. 15 figures, 3 tables.

354 (CONF-790420--, pp 4-8) MATERIALS ASPECTS OF NUCLEAR WASTE DISPOSAL IN CANADA. Cameron, D.J.; Strathdee, G.G. (Whiteshell Nuclear Research Establishment, Pinawa, Manitoba). May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

The concept of disposal of nuclear waste in a deep, hard-rock vault raises a number of questions concerning the behavior of materials. The man-made materials under consideration are the waste form, engineered containment, and the buffer and backfill materials. The general approach to developing each of these barriers is presented.

355 (CONF-790420--, pp 47-51) GLASSY AND CRYSTALLINE HIGH-LEVEL NUCLEAR WASTE FORMS: AN ATTEMPT AT CRITICAL EVALUATION. Lutze, W. (Hahn-Meitner-Institut fuer Kernforschung GmbH, Berlin, Germany). May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

Pros and cons of glassy and crystalline nuclear waste forms are discussed. An evaluation is given in terms of technological simplicity and safety-relevant product properties, i.e., thermal, chemical, mechanical, and radiation stability, showing that glasses are the first choice and alternatives are either the backup solution or second-generation products.

356 (CONF-790420--, pp 66-72) CHARACTERIZATION AND EVALUATION OF MULTIBARRIER NUCLEAR WASTE FORMS. Rusin, J.M.; Lokken, R.O.; Wald, J.W. (Battelle Pacific Northwest Labs., Richland, WA). May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

Four multibarrier concepts using coatings and metal matrices were developed and demonstrated on a 1-liter scale. The effect of volatility, leachability, and impact resistance on product integrity is discussed. Results of radiation stability tests are also included.

357 (CONF-790420--, pp 238-242) EVALUATION OF LONG-TERM LEACHING OF BOROSILICATE GLASS IN PURE WATER. Lanza, F.; Parnisari, E. (CEC, Joint Research Centre, Ispra, Italy). May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA

(30 Apr 1979).
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 leaching rate is followed by measurements of weight loss and analysis of the surface composition. An accumulation at the surface of the less soluble cations is observed.

358 (CONF-790420--, pp 243-247) DEVELOPMENT OF ALUMINOSILICATE AND BOROSILICATE GLASSES AS MATRICES FOR CANDU HIGH-LEVEL WASTE. Strathdee, G.G.; McIntyre, N.S.; Taylor, P. (Atomic Energy of Canada Ltd., Pinewa, Manitoba). May 1979. Dep. NTIS, PC A17/MF A01.
From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).
This paper covers the results of analyses of two radioactive nepheline syenite glass blocks recovered from in-ground leaching experiments at the Chalk River Nuclear Laboratories. Current research on borosilicate glasses for immobilization of high-level waste is also described.

359 (CONF-790420--, pp 248-255) LEACHING OF RADIOACTIVE WASTE STORAGE GLASSES. Boult, K.A.; Dalton, J.T.; Hall, A.R.; Hough, A.; Marples, J.A.C. (Atomic Energy Research Establishment, Harwell, England). May 1979. Dep. NTIS, PC A17/MF A01.
From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).
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 studies are waste composition, glass devitrification, the source and pH of the water, temperature, and radiation dose to the glass.

360 (CONF-790420--, pp 256-262) CORROSION BEHAVIOR OF ZINC BOROSILICATE SIMULATED NUCLEAR WASTE GLASS. Clark, D.E.; Yen-Bower, E.L.; Hench, L.L. (Univ. of Florida, Gainesville). May 1979. Dep. NTIS, PC A17/MF A01.
From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).
The corrosion behavior of a zinc borosilicate glass containing simulated nuclear wastes (SNWG) is evaluated in acidic, neutral, and basic solutions. Auger electron spectroscopy coupled with argon ion milling (AES-IM) and infrared reflection spectroscopy (IRRS) coupled with sequential polishing are used to determine the depth of the leached layers. Leaching rates based on solution analyses are calculated for several species including cesium and strontium. The leaching rates of some species from the glass are at least an order of magnitude greater in acidic than in neutral and basic solutions. Scanning electron microscopy in conjunction with energy dispersive X-ray spectroscopy (SEM-EDS) is used to monitor surface morphological changes that accompany glass corrosion.

361 (CONF-790420--, pp 263-268) CHEMICAL DURABILITY OF NUCLEAR WASTE GLASSES. Simmons, J.H.; Barkatt, A.; Macedo, P.B.; Pehrsson, P.E.; Simmons, C.J.; Barkatt, A.; Tran, D.C.; Sutter, H.; Saleh, M. (Catholic Univ. of America, Washington, DC). May 1979. Dep. NTIS, PC A17/MF A01.
From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).
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 the corrosion mechanism. Measurement on a model glass, a borosilicate, and a high-silica glass fixation medium are reported. The extrapolation of short-term laboratory tests to long-time storage is discussed, and the essential parameters for accurate long-term predictions are determined and evaluated.

362 (CONF-790420--, pp 269-273) CHEMICAL DURABILITY AND CHARACTERIZATION OF NUCLEAR WASTE FORMS IN A HYDROTHERMAL ENVIRONMENT. Braithwaite, J.W.; Johnstone, J.K. (Sandia Labs., Albuquerque, NM). May 1979. Dep. NTIS, PC A17/MF A01.
From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).
The chemical durability of a simulated copper borosilicate waste glass and titanate waste ceramic has been studied in hydrothermal environments which could possibly be encountered in a bedded salt or sub-seabed waste isolation repository. The major parameters investigated which affect matrix corrosion and cesium solubilization include solution saturation and equilibrium phenomena, solution composition (especially the Mg^{+2} ion concentration), pH, particle size, temperature, and time.

363 (CONF-790420--, pp 274-276) HYDROTHERMAL STABILITY OF SPENT FUEL AND HIGH-LEVEL WASTE CERAMICS IN THE GEOLOGIC REPOSITORY ENVIRONMENT. McCarthy, G.J.; White, W.B.; Komarneni, S.; Schaetz, B.E.; Freeborn, W.P.; Smith, D.K. (Pennsylvania State Univ., University Park). May 1979. Dep. NTIS, PC A17/MF A01.
From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).
The stability trends for four simulated waste forms (spent fuel, a reference glass, a supercalcine ceramic) under hydrothermal conditions (100 to 300°C; 400°C for the coated ceramic; 300 bars) with solutions alone and in contact with basalt and shale are summarizd. 2 tables.

364 (CONF-790420--, pp 277-283) DEVITRIFICATION AND LEACHING EFFECTS IN HLW GLASS: COMPARISON OF SIMULATED AND FULLY RADIOACTIVE WASTE GLASS. Wald, J.W.; Westsik, J.H. Jr. (Battelle Pacific Northwest Labs., Richland, WA). May 1979. Dep. NTIS, PC A17/MF A01.
From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).
A detailed examination of four fully radioactive nuclear waste glass compositions has been conducted and includes microstructural characterization and leaching behavior. All results have indicated that the fully radioactive glasses are quite stable under the temperature conditions studied, and their behavior can be typified by nonradioactive products of similar compositions. 2 figures, 6 tables.

365 (CONF-790420--, pp 284-288) LONG-TERM LEACH RATES OF GLASSES CONTAINING ACTUAL WASTE. Wiley, J.R.; LeRoy, J.H. (E.I. du Pont de Nemours and Co., Aiken, SC). May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

Leach rates of borosilicate glasses that contained actual Savannah River Plant waste were measured. Leaching was done by water and by buffer solutions of pH 4, 7, and 9. Leach rates were then determined from the amount of ^{137}Cs, ^{90}Sr, and plutonium released into the leach solutions. The cumulative fractions leached were fit to a mathematical model that included leaching by diffusion and glass dissolution. 5 figures, 3 tables.

366 (CONF-790420--, pp 349-353) IODIDE AND IODATE SODALITES FOR THE LONG-TERM STORAGE OF IODINE-129. Strachan, D.M.; Babad, H. (Rockwell Hanford Operations, Richland, WA). May 1979. Dep. NTIS, PC A17/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

There exist several proposals for the storage of ^{129}I. None of these propose the use of a mineral with demonstrated geologic stability. The work described in this paper has identified the minerals iodide and iodate sodalites $[Na_8(AlSiO_4)_6I_2/(IO_3)_2]$ as good candidates for the long-term storage of ^{129}I. 4 tables.

367 (DOE/ET--0028(Vol.3)) TECHNOLOGY FOR COMMERCIAL RADIOACTIVE WASTE MANAGEMENT. (Battelle Pacific Northwest Labs., Richland, WA (USA)). May 1979. Contract EY-76-C-06-1830. 492p. Dep. NTIS, PC A21/MF A01.

Conceptual facilities for interim storage of various treated transuranic (TRU) and gaseous wastes produced during fuel reprocessing and mixed oxide fuel fabrication are described in volume 3. Alternatives for interim storage of spent fuel prior to reprocessing or geologic isolation are also described. The storage concepts are based on available technology. They do not necessarily represent optimum designs, but are representative of what could be achieved with current capabilities. In actual applications it is reasonable to expect that there could be some improvements over these concepts, reflected in lower costs, lower environmental impacts, or both. These conceptual descriptions provide a reasonable basis for cost analysis and for development of estimates of environmental impacts. Sections are devoted to: storage of high-level liquid waste in large stainless steel tanks; two interim storage concepts for fuel residue waste (fuel hulls and hardware) waste storage; storage concepts for other nonhigh-level TRU waste; two alternatives for storage of solidified high-level waste; conceptual storage for large quantities of plutonium oxide; a concept for storing krypton gas cylinders; and alternatives for both short-term and extended storage of spent fuel.

368 (DOE/UMT--0200) EVALUATION OF LINERS FOR A URANIUM-MILL TAILINGS DISPOSAL SITE: A STATUS REPORT. Buelt, J.L.; Hale, V.Q.; Barnes, S.M.; Silviera, D.J. (Battelle Pacific Northwest Labs., Richland, WA (USA)). May 1981. Contract AC06-76RL01830. 67p. (PNL--3679). NTIS, PC A04/MF A01. Order Number DE81023671.

The United States Department of Energy is conducting a program designed to reclaim or stabilize inactive uranium-mill tailings sites. This report presents the status of the Liner Evaluation Program. The purpose of the study was to identify eight prospective lining materials or composites for laboratory testing. The evaluation was performed by 1) reviewing proposed regulatory requirements to define the material performance criteria; 2) reviewing published literature and communicating with industrial and government experts experienced with lining materials and techniques; and 3) characterizing the tailings at three of the sites for calcium concentration, a selection of anions, radionuclides, organic solvents, and acidity levels. The eight materials selected for laboratory testing are: natural soil amended with sodium-saturated montmorillonite (Volclay); locally available clay in conjunction with an asphalt emulsion radon suppression cover; locally available clay in conjunction with a multibarrier radon suppression cover; rubberized asphalt membrane; hydraulic asphalt concrete; chlorosulfonated polyethylene (hypalon) or high-density polyethylene; bentonite, sand and gravel mixture; and catalytic airblown asphalt membrane. The materials will be exposed in test units now being constructed to conditions such as wet/dry cycles, temperature cycles, oxidative environments, ion-exchange elements, etc. The results of the tests will identify the best material for field study. The status report also presents the information gathered during the field studies at Grand Junction, Colorado. Two liners, a bentonite, sand and gravel mixture, and a catalytic airblown asphalt membrane, were installed in a prepared trench and covered with tailings. The liners were instrumented and are being monitored for migration of moisture, radionuclides, and hazardous chemicals. The two liner materials will also be subjected to accelerated laboratory tests for a comparative assessment.

369 (EPRI-ER--469(Pt.A)) EVALUATIONS OF FUSION-FISSION (HYBRID) CONCEPTS: MARKET PENETRATION ANALYSIS FOR FUSION-FISSION HYBRIDS. PART A. Engel, R.L.; Deonigi, D.E. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jan 1976. 36p. Dep. NTIS, PC A03/MF A01.

This report summarizes findings of the fusion-fission studies conducted for the Electric Power Research Institute by Battelle, Pacific Northwest Laboratories. This particular study focused on the evaluation of fissile material producing hybrids. Technical results of the evaluation of actinide burning are presented in a companion volume, Part B.

370 (EPRI-ER--469(Pt.B)) EVALUATIONS OF FUSION-FISSION (HYBRID) CONCEPTS: TRANSMUTATION OF HIGH-LEVEL ACTINIDE WASTE IN HYBRIDS. PART B. Jenquin, U.P.; Leonard, B.R. Jr. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jun 1976. 60p. Dep. NTIS, PC A04/MF A01.

This report summarizes the findings of the fusion-fission studies conducted for the Electric Power Research Institute by Battelle, Pacific Northwest Laboratories. This particular study focused on the evaluation of actinide burning. Detailed results of an evaluation of fissile material producing hybrids are presented in a companion volume, Part A.

371 (HEDL-TC--1384(Rev.1)) SPENT FUEL CHARACTERIZATION FOR THE COMMERCIAL WASTE AND SPENT FUEL PACKAGING PROGRAM. Fish, R.L.; Davis, R.B.; Pasupathi, V.; Klingensmith, R.W. (Hanford Engineering Development Lab.,

Richland, WA (USA); Battelle Columbus Labs., OH (USA)). Mar 1980. Contract EY-76-C-14-2170. 55p. Dep. NTIS, PC A04/MF A01.

This document presents the rationale for spent fuel characterization and provides a detailed description of the characterization examinations. Pretest characterization examinations provide quantitative and qualitative descriptions of spent fuel assemblies and rods in their irradiated conditions prior to disposal testing. This information is essential in evaluating any subsequent changes that occur during disposal demonstration and laboratory tests. Interim examinations and post-test characterization will be used to identify fuel rod degradation mechanisms and quantify degradation kinetics. The nature and behavior of the spent fuel degradation will be defined in terms of mathematical rate equations from these and laboratory tests and incorporated into a spent fuel performance prediction model. Thus, spent fuel characterization is an essential activity in the development of a performance model to be used in evaluating the ability of spent fuel to meet specific waste acceptance criteria and in evaluating incentives for modification of the spent fuel assemblies for long-term disposal purposes.

372 (ONWI--88, pp 169-185) WASTE ISOLATION SAFETY ASSESSMENT PROGRAM WISAP RELEASE SCENARIO ANALYSIS. Benson, G.L. (Battelle Pacific Northwest Labs., Richland, WA). 1979. NTIS, PC A19/MF A01.

From US/FRG bilateral workshop on waste isolation performance assessment and in-situ testing; Berlin, F.R. Germany (1 Oct 1979).

This paper consists of viewgraphs which outline the release scenario analysis and disruptive event analysis methodologies for a simulated waste repository over a 10^6 year time frame. (DLC)

373 (PNL--1904) RESEARCH AND DEVELOPMENT PLANS FOR DISPOSAL OF HIGH-LEVEL AND TRANSURANIC WASTES. Bartlett, J.W.; Platt, A.M. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Sep 1978. Contract EY-76-C-06-1830. 75p. Dep. NTIS, PC A04/MF A01.

This plan recommends a 20-year, 206 million (1975 $'s) R and D program on geologic structures in the contiguous U.S. and on the midplate Pacific seabed with the objective of developing an acceptable method for disposal of commercial high-level and transuranic wastes by 1997. No differentiation between high-level and transuranic waste disposal is made in the first 5 years of the program. A unique application of probability theory to R and D planning establishes, at a 95% confidence level, that the program objective will be met if at least fifteen generic options and five specific disposal sites are explored in detail and at least two pilot plants are constructed and operated. A parallel effort on analysis and evaluation maximizes information available for decisions on the acceptability of the disposal techniques. Based on considerations of technical feasibility, timing and technical risk, the other disposal concepts, e.g., ice sheets, partitioning, transmutation and space disposal cited in BNWL-1900 are not recommended for near future R and D.

374 (PNL--1948) RADIOACTIVITY ASSOCIATED WITH BIOTA AND SOILS OF THE 216-A-24 CRIB. Klepper, E.L.; Rogers, L.E.; Hedlund, J.D.; Schreckhise, R.G. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Mar 1979. Contract EY-76-C-06-1830. 62p. Dep. NTIS, PC A04/MF A01.

The 216-A-24 Crib was used from 1958 to 1966 to receive condensate (^{137}Cs, ^{90}Sr) from the 241-A and 241-AX Tank Farms. In 1975, rabbitbrush plants (Chrysothamnus nauseosus) growing on the crib were found to contain radioactive materials. Highest levels of activity and densest stands of rabbitbrush plants were in the center of the second section of the crib where a Geiger--Mueller count rate meter showed surface exposure rates of certain plants to be as high as 125 times background. Of the 519 shrubs on the second section, 364 were at background, 62 were up to 10 times background, and 93 were over 10 times background. Contaminated shrubs were restricted to the center of the crib; all shrubs more than 6 meters away from the centerline were at background levels. The shrubs appeared to absorb ^{137}Cs and trace amounts of other fission products from within or below the gravel layers. The gravel appeared to retain significant amounts of ^{137}Cs. Soil above the gravel layers was not contaminated. Cesium-137 was detectable in the upper cm of soil and in the litter, especially beneath canopies of plants with high levels of ^{137}Cs in their leaves. Some animal samples collected on the crib contained ^{137}Cs. Insect species associated with a rabbitbrush shrub containing ^{137}Cs and its litter showed higher levels of ^{137}Cs than other wider-ranging species caught in pitfall traps and by hand. Two out of seven pocket mice contained detectable amounts of ^{137}Cs with 70% of the total body burden in the muscle and skeleton. Recommendations for restoration of the crib surface include eradication of rabbitbrush plants, removal of the surface centimeter of soil on the central 12 meters of the crib, removal of the cobble layer from the surface, installation of a one-foot layer of clean soil and revegetation of the surface with cheatgrass. 12 figures, 14 tables.

375 (PNL--2247) DEVITRIFICATION BEHAVIOR IN A ZINC BOROSILICATE NUCLEAR WASTE GLASS. Turcotte, R.P.; Wald, J.W. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Mar 1978. Contract EY-76-C-06-1830. 43p. Dep. NTIS, PC A03/MF A01.

Experimental studies of thermally induced changes in a simulated nuclear waste glass were conducted to define the composition and concentration of all phases formed over a broad range of time (up to one year) and temperature (<1200°C) conditions. Depending on time and temperature, a steady-state condition is achieved in which a number of crystalline phases coexist with a glass phase which is partially depleted of some elements. Concentrations of the phases increase with decreasing temperature but usually reach only a fraction of their maximum theoretical concentration. Considering the major phase formed (Zn_2SiO_4), this fractional concentration is about 10% at 900°C and 45% at 700°C, when equilibrium is achieved. Under these unfavorable time/temperature conditions, the glass is about one-third crystalline. Because of melt insolubles (RuO_2,Pd) the most homogeneous glass will contain approximately 3% crystalline phases. Crystallization occurs at rates in agreement with those estimated from theory, based on a knowledge of the glass viscosity and an estimated heat of crystallization for the Zn_2SiO_4 phase. The times to reach steady-state concentrations range from a few hours at 900°C to approximately 1 year at 700°C. No crystallization at 500°C was observed after one year. The crystal ingrowth rates follow a reasonable 1/T dependence for Zn_2SiO_4 and give an activation energy for the process of approximately 40 kcal/mole. Growth rates for the other phases are of the same order of magnitude as for Zn_2SiO_4. Results of the present work are used to evaluate processing/storage conditions needed to maintain a vitreous product. It is shown that crystallization over very long time periods (10^6 years) will likely occur only if ambient temperatures exceed approximately 225°C for the entire time period.

376 (PNL--2278) HISTORY OF PROTOTYPE HIGH LEVEL WASTE CANISTER SS-9 WHILE IN AIR AND WATER STORAGE. Bradley, D.J. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Nov 1977. Contract EY-76-C-06-1830. 39p. Dep. NTIS, PC A03/MF A01.

Canister SS-9 was filled with high-level phosphate ceramic waste material in March 1969. Following 1.2 years water storage at 50°C, 3.5 years hot air storage at 400 to 500°C, and 10 months water storage at 50°C, the canister failed. The canister has three visible cracks, one of which is 13 in. long. 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. Cores were taken from Canister SS-9 and the leach rate of the material in deionized water was determined to be 5.1×10^{-4} g/cm^2-day for the first day. Averaged over 90 days, the material leach rate was 3.1×10^{-5} g/cm^2-day. Since it was known that the failure occurred sometime between quarterly canister storage water sampling periods, these leach rates were used to calculate an effective waste surface area presented by the canister cracks. Thus, the leach rates for the first day and the average for 90 days represent the extremes that could have occurred. The effective waste surface area contacted via the canister cracks was calculated to be between 1 and 20 m^2. Based on the calculated effective surface area and the above leach rates of the phosphate ceramic material, 6.1 g of waste were dissolved per day. This is related to the activity of the canister as follows: Curies of ^{137}Cs in Canister SS-9 (corrected to May 1977) = 1.78×10^4; Curies of ^{137}Cs leached in one day, based on above leach rates = 5.5×10^{-1}; Percent of total cesium in canister = 3.1×10^{-3}; Curies of ^{137}Cs leached in one day based on averaged 90th day leach rate and total canister surface area of 1.35 m^2 = 1.41×10^{-2}; Percent of total cesium in canister = 7.9×10^{-5}. 15 figures, 14 tables.

377 (PNL--2444) NATIONAL WASTE TERMINAL STORAGE PROGRAM: POTENTIAL PROBLEMS IN THE WASTE TRANSPORTATION SYSTEM. DeSteese, J.G.; Rhoads, R.E. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Dec 1977. EY-76-C-06-1830;W-7405-ENG-26. 30p. Dep. NTIS, PC A03/MF A01.

Potential problems are identified which may impact the planning, organization, and operation of nuclear waste transportation systems serving federal repositories. These system-level problems have the potential of seriously interfering with the overall OWI Transportation/Logistics Study objective of having a viable nuclear waste transportation system in 1985. This report includes recommended action and priority judgments to address these problems and minimize their impact. The potential problems identified as most important have consequences which may impact the overall state of future preparedness for transporting nuclear waste. Other important concerns relate to the imposition of unnecessarily severe and costly restrictions on nuclear waste transportation, public and carrier acceptance, and the involvement of interested parties in planning and decision-making. The major recommendation of this report is that the planning and development of the waste transportation system should be controlled by a central planning activity which anticipates the impact of uncertainties and undesirable events.

378 (PNL--2448) BATCH KD MEASUREMENTS OF NUCLIDES TO ESTIMATE MIGRATION POTENTIAL AT THE PROPOSED WASTE ISOLATION PILOT PLANT IN NEW MEXICO. Serne, R.J.; Rai, D.; Mason, M.J.; Molecke, M.A. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 7 Oct 1977. Contract EY-76-C-06-1830. 92p. Dep. NTIS, PC A05/MF A01.

Laboratory measurements to determine the sorption distribution coefficients, Kd, of radionuclides present in, and potentially leached from, radioactive wastes, in contact with representative geologic media, have been conducted. The nuclides studied include Cs, Sr, Tc, Ru, Sb, Ce, Eu, Pu, Np, Cm, Am, U, and Pa. The crushed rock materials used were from the vicinity of the Waste Isolation Pilot Plant, a project to isolate radioactive wastes in a bedded salt facility, near Carlsbad, New Mexico. Solutions used consist of salt brine and groundwater, specific to the WIPP site, plus distilled water, for laboratory intercomparisons. The batch Kd data reported, plus data from sorption and migration measurements being conducted or planned elsewhere, will be used to evaluate the potential for radionuclide migration from the bedded salt WIPP facility. The data can be used for transport modeling and for safety assessment determinations.

379 (PNL--2451) WASTE ISOLATION SAFETY ASSESSMENT PROGRAM. SUMMARY OF FY-77 PROGRESS. Burkholder, H.C.; Greenborg, J.; Stottlemyre, J.A.; Bradley, D.J.; Raymond, J.R.; Serne, R.J. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Nov 1977. Contract EY-76-C-06-1830. 37p. Dep. NTIS, PC A03/MF A01.

Objective is to provide long-term safety information for the National Waste Terminal Storage Program. Work in FY 77 supported the development of the generic assessment method (release scenario analysis, release consequence analysis) and of the generic data base (waste form release rate data, radionuclide geochemical interaction data). (DLC)

380 (PNL--2480) ULTRASONIC INSPECTION TECHNIQUES FOR TWO WELD CLOSURES PROPOSED FOR RSSF WASTE STORAGE CASKS. Doctor, S.R.; Morris, C.J. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jan 1978. Contract EY-76-C-06-1830. 28p. Dep. NTIS, PC A03/MF A01.

One method being considered for interim storage of high-level radioactive waste materials is to place these materials in large sealed stainless steel canisters and subsequently store these canisters in a second sealed steel storage cask. Weld procedures are proposed as the closure or seal for these vessels. Inspection of these closures to assure initial and long-term integrity of the closure welds presents a challenge to nondestructive testing. The environment is thermally (400 to 1000°F) and radioactively (10^5 R/hr) hot necessitating remote inspection procedures. As a result of research work, ultrasonic test techniques were developed for inspecting the final weld closure of the waste cask. Special transducers, coupling techniques and fixturing were developed and demonstrated in a mockup test facility by remotely examining a 2-in. full penetration weld closure. The examination was performed at room ambient and at a temperature of 200°F. Testing at the desired temperature of 400°F was not completed due to a loss in transducer performance at temperatures in excess of 200°F. Upon completion of the mockup test demonstration, the cask was subjected to a drop test. The ultrasonic results of the pre- and post-examination of two weld closures (the 2-in. full penetration weld and the threaded plug with seal weld) are presented. After the completion of the drop test, both weld closures were radiographed. The radiographs verified the ultrasonic examination and the presence of weld defects in the same areas. Sectioning of the cask closure welds with metallographic verification was not completed at the time of this writing. As a result of the experience gained from the Retrievable Surface Storage Facility (RSSF) storage cask program, recommendations pertaining to the nondestructive engineering development program for Spent Unreprocessed Fuel (SURF) storage casks are presented.

381 (PNL--2496) RELATIONSHIPS BETWEEN PROPERTIES OF HANFORD AREA SOILS AND THE AVAILABILITY OF ^{134}CS AND ^{85}SR FOR UPTAKE BY CHEATGRASS AND TUMBLEWEED. Cataldo, D.A.; Routson, R.C.; Paine, D.;Garland, T.R. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Oct 1978. Contract EY-77-C-06-1030. 43p. Dep. NTIS, PC A03/MF A01.

The relationships between plant root absorption mechanisms and basic soil parameters which effect the concentration of Sr and Cs, and analog ions in soil solution are reviewed. Based on these relationships, studies were undertaken to determine the relative availability of Cs and Sr for uptake by plants grown on Hanford area soils, and the feasibility of developing a predictive relationship between readily measurable soil parameters and plant availability of Cs and Sr. Concentration ratios were determined for cheatgrass and tumbleweed grown on Hanford area soils representing a range in properties (Burbank, Ritzville, Lickskillet, Rupert and Warden). Concentration ratios ranged from 0.0078 to 0.066 and 3.5 to 16 for ^{134}Cs and ^{85}Sr, respectively. Soils were analyzed for physical properties, mineralogy, extractable cations and extractable ^{134}Cs and ^{85}Sr. Simple correlation analyses showed Cs and Sr uptake in cheatgrass and Sr uptake by tumbleweed to be related to cation exchange capacity, and extractable Sr, Ba and Mg. However, in the case of Cs, this correlation may be coincidental since these divalent cations are not chemical analogs of Cs. Uptake of Cs by tumbleweed showed weak correlations with extractable and exchangeable K. Factor analysis and principle components analysis did not assist in further quantitation of the relationships between plant uptake and soil parameters.

382 (PNL--2557) CHARACTERIZATION OF THE HANFORD 300 AREA BURIAL GROUNDS. FINAL REPORT: DECONTAMINATION AND DECOMMISSIONING. Phillips, S.J.; Ames, L.L.; Fitzner, R.E.; Gee, G.W.; Sandness, G.A.; Simmons, C.S. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jan 1980. Contract EY-76-C-06-1830. 255p. Dep. NTIS, PC A12/MF A01.

Pacific Northwest Laboratory conducted a series of investigations at the Hanford Site to develop technologies for characterizing and monitoring radioactive waste burial facilities that could be used in determining appropriate decommissioning alternatives. Specific objectives were to develop unique functional geophysics, geochemical, soil physics, numerical modeling, and biological methodologies needed to better characterize and monitor buried radioactive waste disposal sites. To meet these objectives the project was divided into four tasks: Task I, Geophysical Evaluation - Geophysical surveys were taken to locate and define the gross composition of waste materials. Task II, Geochemical Analysis - The interaction of disposed radionuclides with geologic media was analyzed through an integrated radiochemical procedure. Task III, Fluid Transport and Modeling - Computer modeling of water migration in partially saturated groundwater systems was verified with actual data collected at a field test facility used to monitor micrometeorological and geohydrological energy and mass transfer factors. Task IV, Biological Transport - Several biological organisms were evaluated for potential radionuclide uptake and transport. Along with the four tasks, the project included a review of pertinent literature and regulatory issues that might affect the alternatives selected. Surveys were taken of the surrounding area and specific sites and operations. The overall results indicated that the 300 Area Burial Grounds have been adequate in containing radioactive waste. Based on the results of the project, the alternatives identified for decommissioning these sites are exhumation and translocation, entombment, perpetual care, and abandonment. Perpetual care (currently used) appears to be the best decommissioning alternative for these burial grounds at this time. However, another alternative may be selected depending on future waste management policies, plans, or activities.

383 (PNL--2563) ORGANIC COMPONENTS OF NUCLEAR WASTES AND THEIR POTENTIAL FOR ALTERING RADIONUCLIDE DISTRIBUTION WHEN RELEASED TO SOIL. McFadden, K.M. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Aug 1980. Contract AC06-76RL01830. 34p. NTIS, PC A03/MF A01.

Normal waste processing at the Hanford operations requires the use of many organic materials, chiefly in the form of complexing agents and diluents. These organic materials and their chemical and radiolytic degradation products, have potential for complexing fission products and transuranium elements, both in the waste streams and upon infiltration into soil, perhaps influencing future sorption or migration of the nuclides. Particular complexation characteristics of various nuclides which constitute the major fission products, long-lived isotopes, and the most mobile in radioactive wastes are discussed briefly with regards to their anticipated sorption or mobility in soils. Included in the discussion are Am, Sb, Ce, Cs, Co, Cm, Eu, I, Np, Pm, Pu, Ra, Ru, Sr, Tc, U, and Zr. 107 references.

384 (PNL--2643) WASTE ISOLATION SAFETY ASSESSMENT PROGRAM SCENARIO ANALYSIS METHODS FOR USE IN ASSESSING THE SAFETY OF THE GEOLOGIC ISOLATION OF NUCLEAR WASTE. Greenborg, J.; Winegardner, W.K.; Pelto, P.J.; Voss, J.W.; Stottlemyre, J.A.; Forbes, I.A.; Fussell, J.B.; Burkholder, H.C. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Nov 1978. Contract EY-76-C-06-1830. 49p. Dep. NTIS, PC A03/MF A01.

The relative utility of the various safety analysis methods to scenario analysis for a repository system was evaluated by judging the degree to which certain criteria are satisfied by use of the method. Six safety analysis methods were reviewed in this report for possible use in scenario analysis of nuclear waste repositories: expert opinion, perspectives analysis, fault trees/event trees, Monte Carlo simulation, Markov chains, and classical systems analysis. Four criteria have been selected. The criteria suggest that the methods: (1) be quantitative and scientifically based; (2) model the potential disruptive events and processes, (3) model the system before and after failure (sufficiently detailed to provide for subsequent consequence analysis); and (4) be compatible with the level of available system knowledge and data. Expert opinion, fault trees/event trees, Monte Carlo simulation and classical systems analysis were judged to have the greatest potential appliation to the problem of scenario analysis. The methods were found to be constrained by limited data and by knowledge of the processes governing the system. It was determined that no single method is clearly superior to others when measured against all the criteria. Therefore, to get the best understanding of system behavior, a combination of the methods is recommended. Monte Carlo simulation was judged to be the most suitable matrix in which to incorporate a combination of methods.

385 (PNL--2651) SOLID PHASES AND SOLUTION SPECIES OF DIFFERENT ELEMENTS IN GEOLOGIC ENVIRONMENTS. Rai, D.; Serne, R.J. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Mar 1978. Contract EY-76-C-06-1830. 140p. Dep. NTIS, PC A07/MF A01.

An investigation was conducted to predict from thermodynamic data the nature of the solid phases and solution species in various weathering environments of different elements (Am, Sb, Ce, Cs, Co, Cm, Eu, I, Np, Pu, Pm, Ra, Ru, Sr, Tc, T, U, and Zr) that are present in radioactive wastes, to predict the degree of adsorption of different elements by the solid matrices and to compare these predictions with observed results, and to determine the influence of different factors (such as Ph, Eh, complexing ligands) on total pore-water concentration and the nature of solution species of selected elements. Based on the nature of the predominant solution species, qualitative predictions regarding the adsorption and movement of various elements can be made. Soils and sediments mainly show cation exchange capacities (since these materials carry a large net negative charge) and to a limited extent, anion exchange capacities. Thus, most cations migrate through the soil or rock column at speeds slower than the groundwater. Relative to each other, the trivalent cations generally move the slowest, the divalent cations at intermediate velocities and the monovalent cations most rapidly. Tritium is unique in that it readily substitutes for hydrogen in water and migrates, therefore, at the same velocity as water. The simple anions tend to migrate through soils and rocks with little reaction because usually a pH of less than 4 is required to activate a significant soil anion exchange capacity. The migration and retention of inorganic complex species (mononuclear and polynuclear) would also be dependent upon the charge and size of the species. The behavior of organic complexed species of elements is difficult to predict because of the lack of knowledge regarding the exact nature of the organic ligands, a wide variation in amounts and types of organic ligands, and the size and solubility of these organics.

386 (PNL--2658) CHARACTERIZATION OF THE HANFORD 300 AREA BURIAL GROUNDS. TASK II. GEOCHEMICAL ANALYSIS. Ames, L.L.; Phillips, S.J. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Apr 1979. Contract EY-76-06-1830. 68p. Dep. NTIS, PC A04/MF A01.

The geochemical interaction of radionuclides with geologic materials must be understood in order to assess radionuclide migration from waste burial structures. In Task II, Geochemical Analysis, soil samples were taken from a representative disposal site to provide information that could be applied to the retired 300 Area Burial Grounds on the Department of Energy's Hanford Site, Washington. The site chosen for evaluation was the 316-4 Crib, which received liquid waste containing hexone and a total of 939 kg of uranium between 1948 and 1956. The methodology developed to analyze the soil samples combined neutron-enhanced autoradiography, x-ray fluorescence, and electron microprobe and radiochemical data. Results showed that the uranium content of the soil immediately underlying the crib is small. Furthermore, the uranium in the 316-4 Crib generally formed an insoluble phosphorus compound that was not readily leached under ambient conditions. However, uranium was easily leached in dilute acid solutions. Consequently, acidic solutions should not be added to waste burial structures that contain uranium compounds. The buffering and cation exchange capacity of the soil clay fraction tends to retard migration of the water-soluble forms, and uranium from the 316-4 Crib has not been detected in the Columbia River. No significant health hazard is apparent at present as a result ofuranium disposal at the 316-4 Crib; however, efforts should continue to restrict access to this site.

387 (PNL--2743) PUBLIC POLICY ISSUES IN NUCLEAR WASTE MANAGEMENT. Nealey, S.M.; Radford, L.M. (Battelle Human Affairs Research Center, Seattle, WA (USA)). Oct 1978. Contract EW-78-C-06-1076. 98p. Dep. NTIS, PC A05/MF A01.

This document aims to raise issues and to analyze them, not resolve them. The issues were: temporal equity, geographic and socioeconomic equity, implementation of a nuclear waste management system, and public involvement.

388 (PNL--2759) HYDROTHERMAL REACTIONS OF NUCLEAR WASTE SOLIDS . A PRELIMINARY STUDY. Westsik, J.H., Jr.; Turcotte, R.P. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Sep 1978. Contract EY-76-C-06-1830. 49p. Dep. NTIS, PC A03/MF A01.

A simulated high-level waste glass, Supercalcine, and some common ceramic and metallic solids were exposed to hydrothermal conditions at 250 and 350°C for time periods ranging from three days to three weeks. Most of the experiments were done in salt brine, but the glass study did include deionized water tests so that the influence of salt could be better understood. Under the extreme hydrothermal conditions of these tests, all of the materials examined underwent measurable changes. The glass is converted to a mixture of crystalline phases, depending upon conditions, giving $NaFeSi_2O_6$ as the primary alteration product. The rate of alteration is higher in deionized water than in salt brine; however, under equivalent test conditions, 66% of the Cs originally in the glass is released to the salt brine, while only 6% is released to deionized water. Rb and Mo are the only other fission product elements significantly leached from the glass. Evidence is presented which shows that sintered Supercalcine undergoes chemical changes in salt brine that are qualitatively similar to those experienced by glass samples. High concentrations of Cs enter the aqueous phase, and Sn and Mo are mobilized. Scouting tests were made with a variety of materials including commercial glasses, granite, UO_2, Al_2O_3, steel, and waste glasses. Weight losses under hydrothermal conditions are in a relatively narrow band, with glass and ceramic materials showing 3 to 20 times greater weight losses than 304L stainless steel in the 250°C test used. The conclusion from these studies is that virtually all solid materials show hydrothermal reactivity at temperatures between 250 and 350°C, and that these extreme conditions are not desirable.

389 (PNL--2776) NATURALLY OCCURRING GLASSES: ANALOGUES FOR RADIOACTIVE WASTE FORMS. Ewing, R.C.; Haaker, R.F. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Apr 1979. Contract EY-76-C-06-1830. 82p. Dep. NTIS, PC A05/MF A01.

Volcanic glasses are very often altered by weathering and leaching and recrystallize to their fine-grained equivalents (rhyolites, felsites). The oldest volcanic glasses are dated at 40 million years before the present, but the majority are much younger. Devitrification textures was produced experimentally; and hydration rates for volcanic glasses were determined as a function of composition, temperature, and climate. Presence of water and temperature are the most important rate controlling variables. Even material that may still be described as glassy often exhibits evidence of alteration and recrystallization. Of the volcanic glasses that are preserved in the geologic record, it would be rare to describe such a glass as pristine. Despite the common alteration and recrystallization effects observed in volcanic glasses, glasses formed as a result of impact, tektites and lunar glasses, may occur in substantially unaltered form. In the case of tektites, their resistance to alteration is a result of their high SiO_2 content and low alkali content. Lunar glasses have been preserved for hundreds of millions of years because they exist in an environment with a low oxygen fugacity and an extremely low water vapor partial presssure. Thus one might expect glasses of particular compositions or in specific types of environment to be stable for long periods of time. These conclusions are applied to radioactive waste disposal over several time periods (0-30h, 30h-20y, 20-200y).

390 (PNL--2782) ASSESSMENT OF EFFECTIVENESS OF GEOLOGIC ISOLATION SYSTEMS. TEST CASE RELEASE CONSEQUENCE ANALYSIS FOR A SPENT FUEL REPOSITORY IN BEDDED SALT. Raymond, J.R.; Bond, F.W.; Cole, C.R.; Nelson, R.W.; Reisenauer, A.E.; Washburn, J.F.; Norman, N.A.; Mote, P.A.; Segol, G. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jan 1980. Contract EY-76-C-06-1830. 82p. Dep. NTIS, PC A05/MF A01.

Geologic and geohydrologic data for the Paradox Basin have been used to simulate movement of ground water and radioacrtive contaminants from a hypothetical nuclear reactor spent fuel repository after an assumed accidental release. The pathlines, travel times and velocity of the ground water from the repository to the discharge locale (river) were determined after the disruptive event by use of a two-dimensional finite difference hydrologic model. The concentration of radioactive contaminants in the ground water was calculated along a series of flow tubes by use of a one-dimensional mass transport model which takes into account convection, dispersion, contaminant/media interactions and radioactive decay. For the hypothetical site location and specific parameters used in this demonstration, it is found that Iodine-129 (I-129) is tthe only isotope reaching the Colorado River in significant concentration. This concentration occurs about 8.0 x 10^5 years after the repository has been breached. This I-129 groundwater concentration is about 0.3 of the drinking water standard for uncontrolled use. The groundwater concentration would then be diluted by the Colorado River. None of the actinide elements reach more than half the distance from the repository to the Colorado River in the two-million year model run time. This exercise demonstrates that the WISAP model system is applicable for analysis of contaminant transport. The results presented in this report, however, are valid only for one particular set of parameters. A complete sensitivity analysis must be performed to evaluate the range of effects from the release of contaminants from a breached repository.

391 (PNL--2797) CONTROLLED SAMPLE PROGRAM PUBLICATION NO. 1: CHARACTERIZATION OF ROCK SAMPLES. Ames, L.L. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Oct 1978. Contract EY-76-C-06-1830. 116p. Dep. NTIS, PC A06/MF A01.

A description is presented of the methodology used and the geologic parameters measured on several rocks which are being used in round-robin laboratory and nuclide adsorption methodology experiments. Presently investigators from various laboratories are determining nuclide distribution coefficients utilizing numerous experimental techniques. Unfortunately, it appears that often the resultant data are dependent not only on the type of groundwater and rock utilized, but also on the experimentor or method used. The Controlled Sample Program is a WISAP (Waste Isolation Safety Assessment Program) attempt to resolve the apparent method and dependencies and to identify individual experimenter's bias. The rock samples characterized in an interlaboratory Kd methodology comparison program include Westerly granite, Argillaceous shale, Oolitic limestone, Sentinel Gap basalt, Conasauga shale, Climax Stock granite, anhydrite, Magenta dolomite and Culebra dolomite. Techniques used in the characterization include whole rock chemical analysis, X-ray diffraction, optical examination, electron microprobe elemental mapping, and chemical analysis of specific mineral phases. Surface areas were determined by the B.E.T. and ethylene glycol sorption methods. Cation exchange capacities were determined with ^{85}Sr, but were of questionable value for the high calcium rocks. A quantitative mineralogy was also estimated for each rock. Characteristics which have the potential of strongly affecting radionuclide Kd values such as the presence of sulfides, water-soluble, pH-buffering carbonates, glass, and ferrous iron were listed for each rock sample.

392 (PNL--2808) RADIOACTIVE WASTES: PUBLIC ATTITUDES TOWARD DISPOSAL FACILITIES. Lindell, M.K.; Earle, T.C.; Hebert, J.A.; Perry, R.W. (Battelle Human Affairs Research Center, Seattle, WA (USA)). Oct 1978. Contract EW-78-C-06-1076. 65p. Dep. NTIS, PC A04/MF A01.

Seventeen geographically widespread, established groups were selected which were expected to vary in their attitudes from strongly pronuclear to strongly antinuclear.

People who tend to be politically active were chosen. The highest level of consensus was found on the need for site monitoring, site control, and information transfer in a waste repository. Overall, the results indicate that pronuclear respondents believe that the hazards of nuclear waste are similar to other industrial risks, while antinuclear respondents are less optimistic about safe storage of nuclear wastes and believe that nuclear power is different.

393 (PNL--2851) SUMMARY OF FY-1978 CONSULTANT INPUT FOR SCENARIO METHODOLOGY DEVELOPMENT. Scott, B.L.; Benson, G.L.; Craig, R.A. (eds.); Harwell, M.A. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Nov 1979. Contract EY-76-C-06-1830. 180p. Dep. NTIS, PC A04/MF A01.

Associated with commercial nuclear power production in the United States is the generation of potentially hazardous radioactive waste products. The Department of Energy (DOE), through the National Waste Terminal Storage (NWTS) Program, is seeking to develop nuclear waste isolation systems in geologic formations. These underground waste isolation systems will preclude contact with the biosphere of waste radionuclides in concentrations which are sufficient to cause deleterious impact on humans or their environments. Comprehensive analyses of specific isolation systems are needed to assess the postclosure expectations of the systems. Assessment of Effectiveness of Geologic Isolation Systems (AEGIS) program has been established for developing the capability of making these analyses. The assessment of repository post-closure safety has two basic components: identification and analyses of breach scenarios and the pattern of events and processes causing each breach, and identification and analyses of the environmental consequences of radionuclide transport and interactions subsequent to a repository breech. Specific processes and events which might affect potential repository sites and, the rates and probabilities for those phenomena are presented. The description of the system interactions and synergisms and of the repository system as an evolving and continuing process are included. Much of the preliminary information derived from the FY-1978 research effort is summarized in this document. This summary report contains information pertaining to the following areas of study: climatology, geomorphology, glaciology, hydrology, meteorites, sea level fluctuations, structural geology and volcanology.

394 (PNL--2851) SUMMARY OF FY-1978 CONSULTATION INPUT FOR SCENARIO METHODOLOGY DEVELOPMENT. Scott, B.L.; Benson, G.L.; Craig, R.A.; Harwell, M.A. (eds.). (Battelle Pacific Northwest Labs., Richland, WA (USA)). Nov 1979. Contract EY-76-C-06-1830. 180p. Dep. NTIS, PC A09/MF A01.

The Scenario Methodology Development task is concerned with evaluating the geologic system surrounding an underground repository and describing the phenomena (volcanic, seismic, meteorite, hydrologic, tectonic, climate, etc.) which could perturb the system and possibly cause loss of repository integrity. This document includes 14 individual papers. Separate abstracts were prepared for all 14 papers. (DLC)

395 (PNL--2851, pp I.1-I.9) DISRUPTIVE EVENT ANALYSIS: VOLCANISM AND IGNEOUS INTRUSION. Crowe, B.M. (Los Alamos Scientific Lab., NM). Nov 1979. Dep. NTIS, PC A09/MF A01.

Summary of FY-1978 consultation input for Scenario Methodology Development.

Three basic topics are addressed for the disruptive event analysis. first, the range of disruptive consequences of a radioactive waste repository by volcanic activity; second, the possible reduction of the risk of disruption by volcanic activity through selective siting of a repository; and third, the quantification of the probability of repository disruption by volcanic activity.

396 (PNL--2854) ASSESSMENT OF EFFECTIVENESS OF GEOLOGIC ISOLATION SYSTEMS. GEOLOGIC FACTORS IN THE ISOLATION OF NUCLEAR WASTE: EVALUATION OF LONG-TERM GEOMORPHIC PROCESSES AND CATASTROPHIC EVENTS. Mara, S.J. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Mar 1980. Contract EY-76-C-06-1830. 54p. Dep. NTIS, PC A04/MF A01.

SRI International has projected the rate, duration, and magnitude of geomorphic processes and events in the Southwest and Gulf Coast over the next million years. This information will be used by the Department of Energy's Pacific Northwest Laboratory (PNL) as input to a computer model, which will be used to simulate possible release scenarios and the consequences of the release of nuclear waste from geologic containment. The estimates in this report, although based on best scientific judgment, are subject to considerable uncertainty. An evaluation of the Quaternary history of the two study areas revealed that each had undergone geomorphic change in the last one million years. Catastrophic events were evaluated in order to determine their significance to the simulation model. Given available data, catastrophic floods are not expected to occur in the two study areas. Catastrophic landslides may occur in the Southwest, but because the duration of the event is brief and the amount of material moved is small in comparison to regional denudation, such events need not be included in the simulation model. Ashfalls, however, could result in removal of vegetation from the landscape, thereby causing significant increases in erosion rates. Because the estimates developed during this study may not be applicable to specific sites, general equations were presented as a first step in refining the analysis. These equations identify the general relationships among the important variables and suggest those areas of concern for which further data are required. If the current model indicates that geomorphic processes (taken together with other geologic changes) may ultimately affect the geologic containment of nuclear waste, further research may be necessary to refine this analysis for application to specific sites.

397 (PNL--2858) HYDROGEOLOGIC EFFECTS OF NATURAL DISRUPTIVE EVENTS ON NUCLEAR WASTE REPOSITORIES. Davis, S.N. (Arizona Univ., Tucson (USA)). Jun 1980. Contract AC06-76RL01830. 41p. NTIS, PC A03/MF A01.

Some possible hydrogeologic effects of disruptive events that may affect repositories for nuclear wastte are described. A very large number of combinations of natural events can be imagined, but only those events which are judged to be most probable are covered. Waste-induced effects are not considered. The disruptive events discussed above are placed into four geologic settings. Although the geology is not specific to given repository sites that have been considered by other agencies, the geology has been generalized from actual field data and is, therefore, considered to be physically reasonable. The geologic settings considered are: (1) interior salt domes of the Gulf Coast, (2) bedded salt of southeastern New Mexico, (3) argillaceous rocks of southern Nevanda, and (4) granitic stocks of the Basin and Range Province. Log-normal distributions of permeabilities of rock units are given for each region. Chapters are devoted to: poresity and permeability of natural materials, regional flow patterns, disruptive events (faulting, dissolution of rock forming minerals, fracturing from various causes, rapid changes of hydraulic regimen); possible hydrologic effects of disruptive events; and hydraulic fracturing.

398 (PNL--2859) PRELIMINARY SUBSURFACE HYDROLOGIC CONSIDERATIONS: COLUMBIA RIVER PLATEAU PHYSIOGRAPHIC PROVINCE. ASSESSMENT OF EFFECTIVENESS OF GEOLOGIC ISOLATION SYSTEMS. Veatch, M.D. (Battelle Pacific Northwest Labs., Richland, WA (USA); Shannon and Wilson, Inc., Seattle, WA (USA)). Apr 1980. Contract EY-76-C-06-1830. 43p. NTIS, PC A03/MF A01.

This report contains a discussion of the hydrologic conditions of the Columbia River Plateau physiographic province. The Columbia River Plateau is underlain by a thick basalt sequence. The Columbia River basalt sequence contains both basalt flows and sedimentary interbeds. These sedimentary interbeds, which are layers of sedimentary rock between lava flows, are the main aquifer zones in the basalt sequence. Permeable interflow zones, involving the permeable top and/or rubble bottom of a flow, are also water-transmitting zones. A number of stratigraphic units are present in the Pasco Basin, which is in the central part of the Columbia River Plateau. At a conceptual level, the stratigraphic sequence from the surface downward can be separated into four hydrostratigraphic systems. These are: (1) the unsaturated zone, (2) the unconfined aquifer, (3) the uppermost confined aquifers, and (4) the lower Yakima basalt hydrologic sequence. A conceptual layered earth model (LEM) has been developed. The LEM represents the major types of porous media (LEM units) that may be encountered at a number of places on the Columbia Plateau, and specifically in the Pasco Basin. The conceptual LEM is not representative of the actual three-dimensional hydrostratigraphic sequence and hydrologic conditions existing at any specific site within the Columbia Plateau physiographic province. However, the LEM may be useful for gaining a better understanding of how the hydrologic regime may change as a result of disruptive events that may interact with a waste repository in geologic media.

399 (PNL--2862) ASSESSMENT OF EFFECTIVENESS OF GEOLOGIC ISOLATION SYSTEMS: THE FEASIBILITY OF COMPUTER INTERROGATION OF EXPERTS FOR WISAP. Wight, L.H. (Battelle Pacific Northwest Labs., Richland, WA (USA); TERA Corp., Berkeley, CA (USA)). May 1980. Contract EY-76-C-06-1830. 101p. NTIS, PC A06/MF A01.

Simulation of the response of a waste repository to events that could initiate a fault tree to breech and failure is currently a keystone to the Battelle Waste Isolation Safety Assessment Program (WISAP). The repository simulation, which is part of the Disruptive Event Analysis Task, models the repository for its entire design life, one million years. This is clearly a challenging calculation, requiring input unlike any other response analysis by virtue of the long design life of the facility. What technology will provide design criteria for a million year design life. Answers to questions like this can, to some extent, be based on data, but always require some subjective judgments. The subjectivity, which is sometimes driven by inadequate or incomplete data or by a lack of understanding of the physical process, is therefore a crucial ingredient in an analysis of initiating events. Because of the variety of possible initiating events (glaciation, man-caused disruption, volcanism, etc.), many expert opinions will be solicited as input. The complexity of the simulation, the variety of experts involved, and the volume of applicable data all suggest that there may be a more direct, economical method to solicit the expert opinion. This report addresses the feasibility of such a system. Background information is presented that demonstrates the advantages of a computer interrogation system over conventional interrogation and assessment techniques. In the subsequent three sections the three elements - structure and decomposition, scaling, and synthesis - that are basic to any interrogation and assessment technique are reviewed. The interrelationship are schematically illustrated between these three fundamental elements and, therefore, serves as a useful guide to these three sections. Each of these three sections begins with a recommended approach to the particular element and ends with an illustration of representative dialogue.

400 (PNL--2863) GLACIOLOGICAL PARAMETERS OF DISRUPTIVE EVENT ANALYSIS. Bull, C. (Battelle Pacific Northwest Labs., Richland, WA (USA); Ohio State Univ., Columbus (USA)). Apr 1980. Contract EY-76-C-06-1830. 55p. Dep. NTIS, PC A04/MF A01.

The possibility of complete glaciation of the earth is small and probably need not be considered in the consequence analysis by the Assessment of Effectiveness of Geologic Isolation Systems (AEGIS) Program. However, within a few thousand years an ice sheet may well cover proposed waste disposal sites in Michigan. Those in the Gulf Coast region and New Mexico are unlikely to be ice covered. The probability of ice cover at Hanford in the next million years is finite, perhaps about 0.5. Sea level will fluctuate as a result of climatic changes. As ice sheets grow, sea level will fall. Melting of ice sheets will be accompanied by a rise in sea level. Within the present interglacial period there is a definite chance that the West Antarctic ice sheet will melt. Ice sheets are agents of erosion, and some estimates of the amount of material they erode have been made. As an average over the area glaciated by late Quaternary ice sheets, only a few tens of meters of erosion is indicated. There were perhaps 3 meters of erosion per glaciation cycle. Under glacial conditions the surface boundary conditions for ground water recharge will be appreciably changed. In future glaciations melt-water rivers generally will follow pre-existing river courses. Some salt dome sites in the Gulf Coast region could be susceptible to changes in the course of the Mississippi River. The New Mexico site, which is on a high plateau, seems to be immune from this type of problem. The Hanford Site is only a few miles from the Columbia River, and in the future, lateral erosion by the Columbia River could cause changes in its course. A prudent assumption in the AEGIS study is that the present interglacial will continue for only a limited period and that subsequently an ice sheet will form over North America. Other factors being equal, it seems unwise to site a nuclear waste repository (even at great depth) in an area likely to be glaciated.

401 (PNL--2874) WASTE ISOLATION SAFETY ASSESSMENT PROGRAM. TECHNICAL PROGRESS REPORT FOR FY-1978. Brandstetter, A.; Harwell, M.A.; Howes, B.W.; Benson, G.L.; Bradley, D.J.; Raymond, J.R.; Serne, R.J.; Schilling, A.H. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jul 1979. Contract EY-76-C-06-1830. 338p. NTIS, PC A15/MF A01.

Associated with commercial nuclear power production in the United States is the generation of potentially hazardous radioactive wastes. The Department of Energy (DOE) is seeking to develop nuclear waste isolation systems in geologic formations that will preclude contact with the biosphere of waste radionuclides in concentrations which are sufficient to cause deleterious impact on humans or their environments. Comprehensive analyses of specific isolation systems are needed to assess the expectations of meeting that objective. The Waste Isolation Safety Assessment Program (WISAP) has been established at the Pacific Northwest Laboratory (operated by Battelle Memorial Institute) for developing the capability of making those analyses. Progress on the following tasks is reported: release scenario analysis, waste form release rate analysis, release consequence analysis, sorption-desorption analysis, and societal acceptance analysis. (DC)

402 (PNL--2875) CHARACTERIZATION OF THE HANFORD 300 AREA BURIAL GROUNDS. DECONTAMINATION AND DECOMMISSIONING REGULATORY ISSUES. Morris, F.A.; Smith, R.F.; Phillips, S.J. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Mar 1979. Contract EY-76-C-06-1830. 33p. Dep. NTIS, PC A03/MF A01.

The Hanford 300 Area Burial Grounds characterization project has identified four management alternatives for disposition of the burial grounds. These alternatives are: (1) abandonment, (2) entombment, (3) perpetual care, and (4) exhumation and translocation. Major Federal statutes and regulations that could apply to management alternatives are identified along with the constraints that applicable laws could impose. This analysis includes explicit attention to the uncertainty surrounding various legal constraints. Also specified are legislative developments as well as trends in other agencies and the courts, obtained by review of legislative proceedings, statutes and regulations, that could result in legislation or policies posing additional constraints.

403 (PNL--2879) ENTRAPMENT OF KRYPTON IN SPUTTER DEPOSITED METALS: A STORAGE MEDIUM FOR RADIOACTIVE GASES. Tingey, G.L.; McClanahan, E.D.; Bayne, M.A.; Moss, R.W. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Apr 1979. Contract EY-76-C-06-1830. 37p. Dep. NTIS, PC A03/MF A01.

Sputter deposition of metals with a negative substrate bias results in a deposit containing relatively large concentrations of the sputtering gas. This phenomenon has been applied as a technique for storage of the radioactive gas, ^{85}Kr, which is generated in nuclear fuels for power production. Alloys which sputter to yield an amorphous product have been shown to contain up to 12 atom % Kr [42 cm^3 of Kr(STP)/g of deposit: concentration equivalent to a gas at 4380 psi pressure]. Release from these metals occurs at so low a rate that extrapolation to long times yields a ^{85}Kr release at 300°C of about 0.06% in 100 years. A preliminary evaluation of the engineering feasibility and economics of the sputtering process indicates that ^{85}Kr can be effectively trapped in a solid matrix with currently available techniques on a scale required for handling DOE-generated waste or commercial reprocessed fuels and that the cost should not be a limiting factor.

404 (PNL--2903) EVALUATION OF KRIGING TECHNIQUES FOR HIGH LEVEL RADIOACTIVE WASTE REPOSITORY SITE CHARACTERIZATION. Doctor, P.G. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jan 1979. Contract EY-76-C-06-1830. 87p. Dep. NTIS, PC A05/MF A01.

Kriging is a statistical method for estimating functions that describe spatially distributed phenomena such as groundwater elevation and depth to basalt. It produces a contour model of the geologic formation of a potential site with an associated measure of uncertainty, and it can be used to optimize the selection of additional sampling locations. Kriging was applied to water potential data and top-of-basalt elevations from the Hanford site; the computer code BLUEPACK was used to perform the computations. The water potential contours were in close agreement with a hand-drawn contour map which is used as a standard. It is concluded that kriging can be a useful tool for geologic waste repository site characterization. (DLC)

405 (PNL--2918) BIOBARRIERS USED IN SHALLOW-BURIAL GROUND STABILIZATION. Cline, J.F. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Mar 1979. Contract EY-76-C-06-1830. 24p. Dep. NTIS, PC A02/MF A01.

These data show that cobblestone can be effective as a barrier to burrowing animals and insects, but not totally effective as a barrier to plant roots. Because of variable weather patterns at Hanford, five to six year studies are recommended for further evaluation of the effectiveness of different materials as biobarriers to radioactive substances. The following criteria must be met to present plant roots from entering buried waste and transporting radioactive or other elements to the soil surface where they can enter the food web: (1) the burial zone beneath the cover should be kept dry; (2) enough soil or other water-retaining substance should be placed in the cover to hold annual precipitation; (3) plants or other substances should be placed in the cover to remove soil moisture from site each year via evaporation and plant transpiration; and (4) different additions to the cover should be designed and placed over the buried waste to prevent burrowing animals from causing channelization of water through the cover to the lower levels. Stone size appeared to affect the plants' rate of root growth since root growth slowed in the air spaces between stones. Root toxin was 100% effective as a means of keeping roots out of the buried waste; this method could be used as a barrier modification where no plant cover is needed. 9 figures, 2 tables.

406 (PNL--2921) CHARACTERIZATION OF THE HANFORD 300 AREA BURIAL GROUNDS. TASK III: FLUID TRANSPORT AND MODELING. Gee, G.W.; Simmons, C.S. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Aug 1979. Contract EY-76-C-06-1830. 126p. Dep. NTIS, PC A07/MF A01.

In Task III, Fluid Transport and Modeling, a computer model was developed and applied to the 300 Area Burial Grounds to analyze the influence of potential evaporation and rainfall patterns on drainage. The model describes one-dimensional unsaturated flow. Fluid transport equations were evaluated to describe the driving forces of fluid flow. The data indicate that the major processes are evaporative drying, capillarity, and gravity flow. Thermally induced transport does not appear significant in the subsurface sediments of the area. Several empirical evaporation methods are available for assessing potential evaporation/evapotranspiration. Four methods were used with the unsaturated flow model. Ultimately, the Blaney-Criddle method was chosen for subsequent simulation examples because it relies only on the climatic data available and gave results comparable to the other methods tested. Simulations showed that a dry layer formation is important in controlling the soil-water balance in the profile. The surface dry layer acts as a mulch to retard the evaporative water losses and increase water storage. The most important climatic factor in determining drainage appears to be yearly rainfall distribution. When rainfall is distributed in fall or winter, during periods of low potential evaporation, both water storage and drainage are increased. Summer showers, on the other hand, were shown to add little to the annual water storage. Rainfall occurring in one year influences the subsequent annual drainage for several succeeding years because of annual changes in water storage capacity and the transient nature of unsaturated flow in the storage zone. 47 figures, 9 tables.

407 (PNL--2928) ASSESSMENT OF EFFECTIVENESS OF GEOLOGIC ISOLATION SYSTEMS. PERSPECTIVES ON THE GEOLOGICAL AND HYDROLOGICAL ASPECTS OF LONG-TERM RELEASE SCENARIO ANALYSES. Stottlemyre, J.A.; Wallace, R.W.; Benson, G.L.; Zellmer, J.T. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jun 1980. Contract AC06-76RL01830. 123p. NTIS, PC A06/MF A01.

Information that may be relevant to individuals involved with analyzing long-term release scenarios of specific repositories for nuclear waste is presented. The bulk of the information is derived from recent studies in West Germany and the United States. Emphasis is on the specific geological and hydrological phenomena that, alone or in concert, could potentially perturb the area around specific repository sites. Research is continuing on most of the topics discussed within this report. Because research is ongoing, statements and conclusions described in this document are subject to change. The main topics of this report are: (1) fracturing, (2) geohydrology, (3) magmatic activity, and (4) geomorphology. Therefore, the site-specific nature of the problem cannot be overemphasized. As an example of how one might combine the many synergistic and time-dependent parameters into a concise format the reader is referred to A Conceputal Simulation Model for Release Scenario Analysis of a Hypothetical Site in Columbia Plateau Basalts, PNL-2892. For additional details on the topics in this report, the reader is referred to the Pacific Northwest Laboratory (PNL) consultant report listed in the bibliography.

408 (PNL--2939) FINITE-ELEMENT THREE-DIMENSIONAL GROUND-WATER (FE3DGW) FLOW MODEL - FORMULATION, PROGRAM LISTINGS AND USERS' MANUAL. Gupta, S.K.; Cole, C.R.; Bond, F.W. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Dec 1979. Contract EY-76-C-06-1830. 283p. Dep. NTIS, PC A13/MF A01.

The Assessment of Effectiveness of Geologic Isolation Systems (AEGIS) Program is developing and applying the methodology for assessing the far-field, long-term post-closure safety of deep geologic nuclear waste repositories. AEGIS is being performed by Pacific Northwest Laboratory (PNL) under contract with the Office of Nuclear Waste Isolation (ONWI) for the Department of Energy (DOE). One task within AEGIS is the development of methodology for analysis of the consequences (water pathway) from loss of repository containment as defined by various release scenarios. Analysis of the long-term, far-field consequences of release scenarios requires the application of numerical codes which simulate the hydrologic systems, model the transport of released radionuclides through the hydrologic systems to the biosphere, and, where applicable, assess the radiological dose to humans. Hydrologic and transport models are available at several levels of complexity or sophistication. Model selection and use are determined by the quantity and quality of input data. Model development under AEGIS and related programs provides three levels of hydrologic models, two levels of transport models, and one level of dose models (with several separate models). This document consists of the description of the FE3DGW (Finite Element, Three-Dimensional Groundwater) Hydrologic model third level (high complexity) three-dimensional, finite element approach (Galerkin formulation) for saturated groundwater flow.

409 (PNL--2954) TESTS FOR DETERMINING IMPACT RESISTANCE AND STRENGTH OF GLASS USED FOR NUCLEAR WASTE DISPOSAL. Bunnell, L.R. (Battelle Pacific Northwest Labs., Richland, WA (USA)). May 1979. Contract EY-76-C-06-1830. 41p. Dep. NTIS, PC A03/MF A01.

Tests are described for determining the impact resistance (Section A) and static tensile strength (Section B) 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. For clarity this report is divided into two sections, Impact Resistance and Tensile Strength.

410 (PNL--2990) PRELIMINARY CONCEPTUAL DESIGNS FOR ADVANCED PACKAGES FOR THE GEOLOGIC DISPOSAL OF SPENT FUEL. Westerman, R.E. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Apr 1979. Contract AC06-76RL01830. 109p. NTIS, PC A06/MF A01.

The present study assumes that the spent fuel will be disposed of in mined repositories in continental geologic formations, and that the post-emplacement control of the radioactive species will be accomplished independently by both the natural barrier, i.e., the geosphere, and the engineered barrier system, i.e., the package components consisting of the stabilizer, the canister, and the overpack; and the barrier components external to the package consisting of the hole sleeve and the backfill medium. The present document provides an overview of the nature of the spent fuel waste; the general approach to waste containment, using the defense-in-depth philosophy; material options, both metallic and nonmetallic, for the components of the engineered barrier system; a set of strawman criteria to guide the development of package/engineered barrier systems; and four preliminary concepts representing differing approaches to the solution of the containment problem. These concepts use: a corrosion-resistant meta canister in a special backfill (2 barriers); a mild steel canister in a corrosion-resistant metallic or nonmetallic hole sleeve, surrounded by a special backfill (2 barriers); a corrosion-resistant canister and a corrosion-resistant overpack (or hole sleeve) in a special backfill (3 barriers); and a mild steel canister in a massive corrosion-resistant bore sleeve surrounded by a polymer layer and a special backfill (3 barriers). The lack of definitive performance requirements makes it impossible to evaluate these concepts on a functional basis at the present time.

411 (PNL--3070) COMPARISON OF INTERA AND WISAP CONSEQUENCE MODEL APPLICATION. ASSESSMENT OF EFFECTIVENESS OF GEOLOGIC ISOLATION SYSTEMS. Cole, C.R.; Bond, F.W. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jan 1980. Contract EY-76-C-06-1830. 66p. Dep. NTIS, PC A04/MF A01.

The Waste Isolation Safety Assessment Program (WISAP) is being conducted to develop, for the Office of Nuclear Waste Isolation (ONWI), the methodology necessary to perform long-term safety assessments of deep geologic repositories. The Waste Isolation Pilot Plant (WIPP) program is developing a nuclear waste storage facility and is performing assessments of that site. WISAP and WIPP have similar, though independent, methodologies for assessing the consequences of a repository breach subsequent to closure. Intera Environmental Consultants are under contract to Sandia Laboratories to conduct the hydrologic and transport modeling for the WIPP Site Release Consequence Analysis (WIPP EIS/ER 1978). To provide a mutual benchmark check of the radionuclide and ground-water transport models of these two programs, ONWI has requested WISAP to perform a release consequence analysis based on the WIPP site, utilizing the same data and conceptual model which the WIPP program used for its environmental assessments. Therefore, only a portion of the WISAP methodology was used; specifically, only WISAP geotransport models were exercised. The other important parts of WISAP assessment methodology were not used, so that WISAP did not develop the scenario nor did WISAP interpret the field data to develop the conceptual model of the geohydrology of the WIPP site. The results of the comparative assessment are presented. Although the different models required slightly different input parameters, the results of the hydrologic simulations show a very close correspondence between the WISAP and WIPP predictions. This was as expected, since the various hydrologic codes available essentially utilize and solve the same basic flow equations. In addition, this report presents the results of the WISAP radionuclide transport model simulations. These results will provide the basis for comparison with WIPP results when these become available.

412 (PNL--3139) STATUS OF SORPTION INFORMATION RETRIEVAL SYSTEM. Hostetler, D.D.; Serne, R.J.; Brandstetter, A. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Sep 1979. Contract EY-76-C-06-1830. 64p. Dep. NTIS, PC A04/MF A01.

A Sorption Information Retrieval System (SIRS) is being designed to provide an efficient, computerized, data base for information on radionuclide sorption in geologic media. The data bank will include Kd values for a large number of radionuclides occurring in radioactive wastes originating from the commercial nuclear power industry. Kd values determined to date span several groundwater compositions and a wide variety of rock types and minerals. The data system will not only include Kd values, but also background information on the experiments themselves. This will allow the potential user to retrieve not only the Kd values of interest but also sufficient information to evaluate the accuracy and usefulness of the data. During FY-1979, the logic structure of the system was designed, the software programmed, the data categories selected, and the data format specified. About 40% of the approximately 5000 Kd experiments performed by the Waste Isolation Safety Assessment Program (WISAP) and its subcontractors during FY-1977 and FY-1978 have been evaluated, coded and keypunched. Additional software improvements and system testing are needed before the system will be fully operational. A workshop requested by the NEA was held to discuss potential international participation in the data system.

413 (PNL--3146) BASALT-RADIONUCLIDE DISTRIBUTION COEFFICIENT DETERMINATIONS. FY-1979 ANNUAL REPORT. Ames, L.L.; McGarrah, J.E. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Sep 1980. Contract AC06-76RL01830. 70p. NTIS, PC A04/MF A01.

Experimental radionuclide distribution coefficients (Kd') were determined for Pomona, Flow E, Umtanum basalts, and secondary mineralization associated with Pomona basalt at 23°, 60° and 150°C. Radionuclides used were ^{75}Se, ^{85}Sr, ^{99}Tc, ^{125}I, ^{135}Cs, ^{226}Ra, ^{237}Np, ^{238}U, ^{241}Am, and ^{241}Pu. Solution oxygen contents were controlled by the basalt/groundwater system (Eh = 600 to 700 mV), and were high (8.2 to 8.4 mg/l) at 23°C. Oxygen contents and pH changed little in contact with basalt. The effects of temperature changes on radionuclide Kd' results varied depending upon the radionuclide involved, solution-solid reactions, and the relationship of the radionuclide to these reactions. For example, cesium Kd' values decreased from 3100 ml/g for Umtanum basalt at 23°C to 120 ml/g at 150°C. At the same time, strontium Kd' values increased for Umtanum basalt from 105 ml/g at 23°C to complete removal at 150°C and 40 days. Radionuclide adsorption coefficient measurements at higher temperatures and pressures were made in addition to the 23°C, solution-solid contact time-conditional Kd (Kd') measurements. These include Kd' measurements with Umtanum basalt, Pomona basalt, Flow E basalt and secondary mineralization and radioisotopes of americium, cesium, iodine, neptunium, plutonium, radium, selenium, strontium, technetium and uranium. The additional temperatures involved were 60°C, 150°C, and 300°C. At 150°C, argon pressures of 6.9, 13.8, 20.7, and 27.6 MPa will be used to ascertain the effects of pressure changes on Kd' values. So far only the 6.9 MPa argon pressure has been investigated. The upper temperature of 250°C is where thermal breakdown of dioctahedral smectites (secondary mineralization) begins.

414 (PNL--3160-2) ASSESSMENT OF EFFECTIVENESS OF GEOLOGIC ISOLATION SYSTEMS. VARIABLE THICKNESS TRANSIENT GROUND-WATER FLOW MODEL. VOLUME 2. USERS' MANUAL. Reisenauer, A.E. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Dec 1979. Contract EY-76-C-06-1830. 157p. Dep. NTIS, PC A08/MF A01.

A system of computer codes to aid in the preparation and evaluation of ground-water model input, as well as in the computer codes and auxillary programs developed and adapted for use in modeling major ground-water aquifers is described. The ground-water model is interactive, rather than a batch-type model. Interactive models have been demonstrated to be superior to batch in the ground-water field. For example, looking through reams of numerical lists can be avoided with the much superior graphical output forms or summary type numerical output. The system of computer codes permits the flexibility to develop rapidly the model-required data files from engineering data and geologic maps, as well as efficiently manipulating the voluminous data generated. Central to these codes is the Ground-water Model, which given the boundary value problem, produces either the steady-state or transient time plane solutions. A sizeable part of the codes available provide rapid evaluation of the results. Besides contouring the new water potentials, the model allows graphical review of streamlines of flow, travel times, and detailed comparisons of surfaces or points at designated wells. Use of the graphics scopes provide immediate, but temporary displays which can be used for evaluation of input and output and which can be reproduced easily on hard copy devices, such as a line printer, Calcomp plotter and image photographs.

415 (PNL--3160-3) VARIABLE THICKNESS TRANSIENT GROUND-WATER FLOW MODEL. VOLUME 3. PROGRAM LISTINGS. Reisenauer, A.E. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Dec 1979. Contract EY-76-C-06-1830. 236p. Dep. NTIS, PC A11/MF A01.

The Assessment of Effectiveness of Geologic Isolation Systems (AEGIS) Program is developing and applying the methodology for assessing the far-field, long-term post-closure safety of deep geologic nuclear waste repositories. AEGIS is being performed by Pacific Northwest Laboratory (PNL) under contract with the Office of Nuclear Waste Isolation (ONWI) for the Department of Energy (DOE). One task within AEGIS is the development of methodology for analysis of the consequences (water pathway) from loss of repository containment as defined by various release scenarios. Analysis of the long-term, far-field consequences of release scenarios requires the application of numerical codes which simulate the hydrologic systems, model the transport of released radionuclides through the hydrologic systems to the biosphere, and, where applicable, assess the radiological dose to humans. Hydrologic and transport models are available at several levels of complexity or sophistication. Model selection and use are determined by the quantity and quality of input data. Model development under AEGIS and related programs provides three levels of hydrologic models, two levels of transport models, and one level of dose models (with several separate models). This is the third of 3 volumes of the description of the VTT (Variable Thickness Transient) Groundwater Hydrologic Model - second level (intermediate complexity) two-dimensional saturated groundwater flow.

416 (PNL--3161-1) ASSESSMENT OF EFFECTIVENESS OF GEOLOGIC ISOLATION SYSTEMS. CIRMIS DATA SYSTEM. VOLUME 1. INITIALIZATION, OPERATION, AND DOCUMENTATION. Friedrichs, D.R. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jan 1980. Contract EY-76-C-06-1830. 164p. Dep. NTIS, PC A08/MF A01.

The Assessment of Effectiveness of Geologic Isolation Systems (AEGIS) Program is developing and applying the methodology for assessing the far-field, long-term post-closure safety of deep geologic nuclear waste repositories. AEGIS is being performed by Pacific Northwest Laboratory (PNL) under contract with the Office of Nuclear Waste Isolation (ONWI) for the Department of Energy (DOE). One task within AEGIS is the development of methodology for analysis of the consequences (water pathway) from loss of repository containment as defined by various release scenarios. The various input parameters required in the analysis are compiled in data systems. The data are organized and prepared by various input subroutines for use by the hydrologic and transport codes. The hydrologic models simulate the groundwater flow systems and provide water flow directions, rates, and velocities as inputs to the transport models. Outputs from the transport models are basically graphs of radionuclide concentration in the groundwater plotted against time. After dilution in the receiving surface-water body (e.g., lake, river, bay), these data are the input source terms for the dose models, if dose assessments are required. The dose models calculate radiation dose to individuals and populations. CIRMIS (Comprehensive Information Retrieval and Model Input Sequence) Data System, a storage and retrieval system for model input and output data, including graphical interpretation and display is described. This is the first of four volumes of the description of the CIRMIS Data System.

417 (PNL--3161-2) CIRMIS DATA SYSTEM. VOLUME 2. PROGRAM LISTINGS. Friedrichs, D.R. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jan 1980. Contract EY-76-C-06-1830. 291p. Dep. NTIS, PC A13/MF A01.

The Assessment of Effectiveness of Geologic Isolation Systems (AEGIS) Program is developing and applying the methodology for assessing the far-field, long-term post-closure safety of deep geologic nuclear waste repositories. AEGIS is being performed by Pacific Northwest Laboratory (PNL) under contract with the Office of Nuclear Waste Isolation (ONWI) for the Department of Energy (DOE). One task within AEGIS is the development of methodology for analysis of the consequences (water pathway) from loss of repository containment as defined by various release scenarios. Analysis of the long-term, far-field consequences of release scenarios requires the application of numerical codes which simulate the hydrologic systems, model the transport of released radionuclides through the hydrologic systems, model the transport of released radionuclides through the hydrologic systems to the biosphere, and, where applicable, assess the radiological dose to humans. The various input parameters required in the analysis are compiled in data systems. The data are organized and prepared by various input subroutines for utilization by the hydraulic and transport codes. The hydrologic models simulate the groundwater flow systems and provide water flow directions, rates, and velocities as inputs to the transport models. Outputs from the transport models are basically graphs of radionuclide concentration in the groundwater plotted against time. After dilution in the receiving surface-water body (e.g., lake, river, bay), these data are the input source terms for the dose models, if dose assessments are required.The dose models calculate radiation dose to individuals and populations. CIRMIS (Comprehensive Information Retrieval and Model Input Sequence) Data System is a storage and retrieval system for model input and output data, including graphical interpretation and display. This is the second of four volumes of the description of the CIRMIS Data System.

418 (PNL--3161-3) ASSESSMENT OF EFFECTIVENESS OF GEOLOGIC ISOLATION SYSTEMS. CIRMIS DATA SYSTEM. VOLUME 3. GENERATOR ROUTINES. Friedrichs, D.R.; Argo, R.S. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jan 1980. Contract EY-76-C-06-1830.

157p. Dep. NTIS, PC A08/MF A01.

The Assessment of Effectiveness of Geologic Isolation Systems (AEGIS) Program is developing and applying the methodology for assessing the far-field, long-term post-closure safety of deep geologic nuclear waste repositories. AEGIS is being performed by Pacific Northwest Laboratory (PNL) under contract with the Office of Nuclear Waste Isolation (ONWI) for the Department of Energy (DOE). One task within AEGIS is the development of methodology for analysis of the consequences (water pathway) from loss of repository containment as defined by various release scenarios. The various input parameters required in the analysis are compiled in data systems. The data are organized and prepared by various input subroutines for utilization by the hydraulic and transport codes. The hydrologic models simulate the groundwater flow systems and provide water flow directions, rates, and velocities as inputs to the transport models. Outputs from the transport models are basically graphs of radionuclide concentration in the groundwater plotted against time. After dilution in the receiving surface-water body (e.g., lake, river, bay), these data are the input source terms for the dose models, if dose assessments are required. The dose models calculate radiation dose to individuals and populations. CIRMIS (Comprehensive Information Retrieval and Model Input Sequence) Data System, a storage and retrieval system for model input and output data, including graphical interpretation and display is described. This is the third of four volumes of the description of the CIRMIS Data System.

419 (PNL--3161-4) ASSESSMENT OF EFFECTIVENESS OF GEOLOGIC ISOLATION SYSTEMS. CIRMIS DATA SYSTEM. VOLUME 4. DRILLER'S LOGS, STRATIGRAPHIC CROSS SECTION AND UTILITY ROUTINES. Friedrichs, D.R. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jan 1980. Contract EY-76-C-06-1830. 242p. 11.

The Assessment of Effectiveness of Geologic Isolation Systems (AEGIS) Program is developing and applying the methodology for assessing the far-field, long-term post-closure safety of deep geologic nuclear waste repositories. AEGIS is being performed by Pacific Northwest Laboratory (PNL) under contract with the Office of Nuclear Waste Isolation (ONWI) for the Department of Energy (DOE). One task within AEGIS is the development of methodology for analysis of the consequences (water pathway) from loss of repository containment as defined by various release scenarios. Analysis of the long-term, far-field consequences of release scenarios requires the application of numerical codes which simulate the hydrologic systems, model the transport of released radionuclides through the hydrologic systems to the biosphere, and, where applicable, assess the radiological dose to humans. The various input parameters required in the analysis are compiled in data systems. The data are organized and prepared by various input subroutines for use by the hydrologic and transport codes. The hydrologic models simulate the groundwater flow systems and provide water flow directions, rates, and velocities as inputs to the transport models. Outputs from the transport models are basically graphs of radionuclide concentration in the groundwater plotted against time. After dilution in the receiving surface-water body (e.g., lake, river, bay), these data are the input source terms for the dose models, if dose assessments are required. The dose models calculate radiation dose to individuals and populations. CIRMIS (Comprehensive Information Retrieval and Model Input Sequence) Data System is a storage and retrieval system for model input and output data, including graphical interpretation and display. This is the fourth of four volumes of the description of the CIRMIS Data System.

420 (PNL--3162) PATHS GROUNDWATER HYDROLOGIC MODEL. Nelson, R.W.; Schur, J.A. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Apr 1980. Contract AC06-76RL01830. 156p. NTIS, PC $10.50/MF $3.50.

A preliminary evaluation capability for two-dimensional groundwater pollution problems was developed as part of the Transport Modeling Task for the Waste Isolation Safety Assessment Program (WISAP). Our approach was to use the data limitations as a guide in setting the level of modeling detail. PATHS Groundwater Hydrologic Model is the first level (simplest) idealized hybrid analytical/numerical model for two-dimensional, saturated groundwater flow and single component transport; homogeneous geology. This document consists of the description of the PATHS groundwater hydrologic model. The preliminary evaluation capability prepared for WISAP, including the enhancements that were made because of the authors' experience using the earlier capability is described. Appendixes A through D supplement the report as follows: complete derivations of the background equations are provided in Appendix A. Appendix B is a comprehensive set of instructions for users of PATHS. It is written for users who have little or no experience with computers. Appendix C is for the programmer. It contains information on how input parameters are passed between programs in the system. It also contains program listings and test case listing. Appendix D is a definition of terms.

421 (PNL--3173) STATUS REPORT ON LWR SPENT FUEL IAEA LEACH TESTS. Katayama, Y.B.; Bradley, D.J.; Harvey, C.O. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Mar 1980. Contract EY-76-C-06-1830. 59p. NTIS, PC A04/MF A01.

Spent light-water-reactor (LWR) fuel with an average burnup of 28,000 MWd/MTU was leach-tested at 25°C using a modified version of the International Atomic Energy Agency procedure. Leach rates were determined from tests conducted in five different solutions: deionized water, sodium chloride (NaCl), sodium bicarbonate ($NaHCO_3$), calcium chloride ($CaCl_2$) and Waste Isolation Pilot Plant B brine solutions. Elemental leach rates are reported based on the release of ^{90}Sr + ^{90}Y, ^{106}Ru, ^{137}Cs, ^{144}Ce, ^{154}Eu, /sup 239 + 240/Pu, ^{244}Cm and total uranium. After 467 days of cumulative leaching, the elemental leach rates are highest in deionized water. The elemental leach rates uin the different solutions generally decreased from deionized water to the 0.03M NaCl solution to the WIPP B brine solution to the 0.03M $NaHCO_3$ solution and was a factor of 20 lower in 0.015M $CaCl_2$ solution than in deionized water. The leach rates of spent fuel and borosilicate waste-glass were also compared. In sodium bicarbonate solution, the leach rates of the two waste forms were nearly equal, but the glass was increasingly more resistant than spent fuel in calcium chloride solution, followed by sodium chloride solution, WIPP B brine solution and deionized water. In deionized water the glass, based on the elemental release of plutonium and curium, was 50 to 400 times more leach resistant than spent fuel.

422 (PNL--3179) MULTICOMPONENT MASS

TRANSPORT MODEL: A MODEL FOR SIMULATING MIGRATION OF RADIONUCLIDES IN GROUND WATER. Washburn, J.F.; Kaszeta, F.E.; Simmons, C.S.; Cole, C.R. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jul 1980. Contract AC06-76RL01830. 120p. NTIS, PC A06/MF A01.

This report presents the results of the development of a one-dimensional radionuclide transport code, MMT2D (Multicomponent Mass Transport), for the AEGIS Program. Multicomponent Mass Transport is a numerical solution technique that uses the discrete-parcel-random-wald (DPRW) method to directly simulate the migration of radionuclides. MMT1D accounts for: convection;dispersion; sorption-desorption; first-order radioactive decay; and n-membered radioactive decay chains. Comparisons between MMT1D and an analytical solution for a similar problem show that: MMT1D agrees very closely with the analytical solution; MMT1D has no cumulative numerical dispersion like that associated with solution techniques such as finite differences and finite elements; for current AEGIS applications, relatively few parcels are required to produce adequate results; and the power of MMT1D is the flexibility of the code in being able to handle complex problems for which analytical solution cannot be obtained. Multicomponent Mass Transport (MMT1D) codes were developed at Pacific Northwest Laboratory to predict the movement of radiocontaminants in the saturated and unsaturated sediments of the Hanford Site. All MMT models require ground-water flow patterns that have been previously generated by a hydrologic model. This report documents the computer code and operating procedures of a third generation of the MMT series: the MMT differs from previous versions by simulating the mass transport processes in systems with radionuclide decay chains. Although MMT is a one-dimensional code, the user is referred to the documentation of the theoretical and numerical procedures of the three-dimensional MMT-DPRW code for discussion of expediency, verification, and error-sensitivity analysis.

423 (PNL--3226) FIELD TEST FACILITY FOR MONITORING WATER/RADIONUCLIDE TRANSPORT THROUGH PARTIALLY SATURATED GEOLOGIC MEDIA: DESIGN, CONSTRUCTION, AND PRELIMINARY DESCRIPTION. Phillips, S.J.; Campbell, A.C.; Campbell, M.D.; Gee, G.W.; Hoober, H.H.; Schwarzmiller, K.O. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Nov 1979. Contract EY-76-C-06-1830. 52p. Dep. NTIS, PC A04/MF A01.

Shallow land burial has been a common practice for disposing radioactive waste materials since the beginning of plutonium production operations. Accurate monitoring of radionuclide transport and factors causing transport within the burial sites is essential to minimizing risks associated with disposal. However, monitoring has not always been adequate. Consequently, the Department of Energy (DOE) has begun a program aimed at better assuring and evaluating containment of radioactive wastes at shallow land burial sites. This program includes a technological base for monitoring transport. As part of the DOE program, Pacific Northwest Laboratory (PNL) is developing geohydrologic monitoring systems to evaluate burial sites located in arid regions. For this project, a field test facility was designed and constructed to assess monitoring systems for near-surface disposal of radioactive waste and to provide information for evaluating site containment performance. The facility is an integrated network of monitoring devices and data collection instruments. This facility is used to measure water and radionuclide migration under field conditions typical of arid regions. Monitoring systems were developed to allow for measurement of both mass and energy balance. Work on the facility is ongoing. Continuing work includes emplacement of prototype monitoring instruments, data collection, and data synthesis. At least 2 years of field data are needed to fully evaluate monitoring information.

424 (PNL--3226(App.1)) FIELD TEST FACILITY FOR MONITORING WATER/RADIONUCLIDE TRANSPORT THROUGH PARTIALLY SATURATED GEOLOGIC MEDIA: DESIGN, CONSTRUCTION, AND PRELIMINARY DESCRIPTION. APPENDIX I. ENGINEERING DRAWINGS. Phillips, S.J.; Campbell, A.C.; Campbell, M.D.; Gee, G.W.; Hoober, H.H.; Schwarzmiller, K.O. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Nov 1979. Contract EY-76-C-06-1830. 20p. Dep. NTIS, PC A02/MF A01.

The engineering plans for a test facility to monitor radionuclide transport in water through partially saturated geological media are included. Drawings for the experimental set-up excavation plan and details, lysimeter, pad, access caisson, and caisson details are presented. (DC)

425 (PNL--3227) GEOPHYSICAL SURVEYS FOR BURIED WASTE DETECTION AT THE IDAHO NATIONAL ENGINEERING LABORATORY. Sandness, G.A.; Rising, J.L.; Kimbrough, J.R. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Dec 1979. Contract EY-76-C-06-1830. 77p. Dep. NTIS, PC A05/MF A01.

This report describes a series of geophysical surveys performed at the Idaho National Engineering Laboratory (INEL). The main purpose of the surveys was to evaluate techniques, principally ground-penetrating radar, for detecting and mapping radioactive wastes buried in shallow trenches and pits. A second purpose was to determine the feasibility of using ground-penetrating radar to measure the depth of basalt bedrock. A prototype geophyscal survey system developed by the US Department of Energy's Pacific Northwest Laboratory was used for this study. Radar, magnetometer, and metal detector measurements were made at three sites in the Radioactive Waste Management Complex (RWMC) at INEL. Radar measurements were made at fourth site adjacent to the RWMC. The combination of three geophysical methods was shown to provide considerable information about the distribution of buried waste materials. The tests confirmed the potential effectiveness of the radar method, but they also pointed out the need for continued research and development in ground-penetrating radar technology. The radar system tested in this study appears to be capable of measuring the depth to basalt in the vicinity of the RWMC.

426 (PNL--3318) MATERIALS CHARACTERIZATION CENTER WORKSHOP ON LEACHING OF RADIOACTIVE WASTE FORMS. SUMMARY REPORT. Ross, W.A.; Strachan, D.M.; Turcotte, R.P.; Westsik, J.H. Jr. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Apr 1980. Contract EY-76-C-06-1830. 104p. Dep. NTIS, PC A06/MF A01.

At the first Materials Characterization Center (MCC) workshop, on the leaching of radioactive waste forms, there was general agreement that, after certain revisions, the proposed leach test plan set forth by the MCC can be expected to meet most of the nuclear waste community's waste form durability data requirements. The revisions give a clearer definition of the purposes of each test and the end uses of the data. As a result of the workshop, the format of the test program has been recast to clarify the purposes, limitations, and interrelationships of the individual tests. There was also a recognition that the leach test program must be based on an understanding of the mechanistic principles of leaching, and that further study is needed to ensure that the approved data from the MCC leach tests will be compatible with mechanistic research needs. It was agreed that another meeting of the participants in Working Groups 3 and 4, and perhaps some other experts, should be held as soon as possible to focus just on the definition of leach test requirements for mechanistic research. The MCC plans to hold this meeting in April 1980. Many of the tests that will lead to increased understanding of mechanisms will of necessity be long-term tests, sometimes lasting for several years. But the MCC also faces pressing needs to produce approved data that can be used for the comparison of waste forms in the relative near-term, i.e., in the next 1 to 3 yr. Therefore, it was decided to initiate a round-robin test of the MCC short-term static leach procedure as soon as practicable. The MCC has tentative plans for organization of the round robin in May 1980.

427 (PNL--3326) LEACH TEST METHODOLOGY FOR THE WASTE/ROCK INTERACTIONS TECHNOLOGY PROGRAM. Bradley, D.J.; McVay, G.L.; Coles, D.G. (Battelle Pacific Northwest Labs., Richland, WA (USA)). May 1980. Contract AC06-76RL01830. 49p. NTIS, PC A03/MF A01.

Experimental leach studies in the WRIT Program have two primary functions. The first is to determine radionuclide release from waste forms in laboratory environments which attempt to simulate repository conditions. The second is to elucidate leach mechanisms which can ultimately be incorporated into nearfield transport models. The tests have been utilized to generate rates of removal of elements from various waste forms and to provide specimens for surface analysis. Correlation between constituents released to the solution and corresponding solid state profiles is invaluable in the development of a leach mechanism. Several tests methods are employed in our studies which simulate various proposed leach incident scenarios. Static tests include low temperature (below 100°C) and high temperature (above 100°C) hydrothermal tests. These tests reproduce nonflow or low-flow repository conditions and can be used to compare materials and leach solution effects. The dynamic tests include single-pass, continuous-flow(SPCF) and solution-change (IAA)-type tests in which the leach solutions are changed at specific time intervals. These tests simulate repository conditions of higher flow rates and can also be used to compare materials and leach solution effects under dynamic conditions. The modified IAEA test is somewhat simpler to use than the one-pass flow and gives adequate results for comparative purposes. The static leach test models the condition of near-zero flow in a repository and provides information on element readsorption and solubility limits. The SPCF test is used to study the effects of flowing solutions at velocities that may be anticipated for geologic groundwaters within breached repositories. These two testing methods, coupled with the use of autoclaves, constitute the current thrust of WRIT leach testing.

428 (PNL--3333) SUMMARY OF NON-US NATIONAL AND INTERNATIONAL RADIOACTIVE WASTE MANAGEMENT PROGRAMS 1980. Harmon, K.M.; Kelman, J.A.; Stout, L.A.; Hsieh, K.A. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Mar 1980. Contract EY-76-C-06-1830. 110p. Dep. NTIS, PC A06/MF A01.

Many nations and international agencies are working to develop improved technology and industrial capability for nuclear fuel cycle and waste management operations. The effort in some countries is limited to research in university laboratories on treating low-level waste from reactor plant operations. In other countries, national nuclear research institutes are engaged in major programs in all phases of the fuel cycle and waste management, and there is a national effort to commercialize fuel cycle operations. Since late 1976, staff members of Pacific Northwest Laboratory have been working under US Department of Energy sponsorship to assemble and consolidate openly available information on foreign and international nuclear waste management programs and technology. This report summarizes the information collected on the status of fuel cycle and waste management programs in selected countries making major efforts in these fields as of the end of January 1980.

429 (PNL--3349) METHODS FOR DETERMINING RADIONUCLIDE RETARDATION FACTORS: STATUS REPORT. Relyea, J.F.; Serne, R.J.; Rai, D. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Apr 1980. Contract AC06-76RL01830. 65p. NTIS, PC A04/MF A01.

This report identifies a number of mechanisms that retard radionuclide migration, and describes the static and dynamic methods that are used to study such retardation phenomena. Both static and dynamic methods are needed for reliable safety assessments of underground nuclear-waste repositories. This report also evaluates the extent to which the two methods may be used to diagnose radionuclide migration through various types of geologic media, among them unconsolidated, crushed, intact, and fractured rocks. Adsorption is one mechanism that can control radionuclide concentrations in solution and therefore impede radionuclide migration. Other mechanisms that control a solution's radionuclide concentration and radionuclide migration are precipitation of hydroxides and oxides, oxidation-reduction reactions, and the formation of minerals that might include the radionuclide as a structural element. The retardation mechanisms mentioned above are controlled by such factors as surface area, cation exchange capacity, solution pH, chemical composition of the rock and of the solution, oxidation-reduction potential, and radionuclide concentration. Rocks and ground waters used in determining retardation factors should represent the expected equilibrium conditions in the geologic system under investigation. Static test methods can be used to rapidly screen the effects of the factors mentioned above. Dynamic (or column) testing, is needed to assess the effects of hydrodynamics and the interaction of hydrodynamics with the other important parameters. This paper proposes both a standard method for conducting batch Kd determinations, and a standard format for organizing and reporting data. Dynamic testing methods are not presently developed to the point that a standard methodology can be proposed. Normal procedures are outlined for column experimentation and the data that are needed to analyze a column experiment are identified.

430 (PNL--3356) ANALYSIS ON THE USE OF ENGINEERED BARRIERS FOR GEOLOGIC ISOLATION OF

SPENT FUEL IN A REFERENCE SALT SITE REPOSITORY. Cloninger, M.O.; Cole, C.R.; Washburn, J.F. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Dec 1980. Contract AC06-76RL01830. 254p. NTIS, PC A12/MF A01. Order Number DE81027620.

A perspective on the potential durability and effectiveness requirements for the waste form, container and other engineered barriers for geologic disposal of spent nuclear fuel has been developed. This perspective is based on calculated potential doses to individuals who may be exposed to radioactivity released from a repository via a groundwater transport pathway. These potential dose commitments were calculated with an integrated geosphere transport and biosphere transport model. A sensitivity analysis was accomplished by varying four important system parameters, namely the waste radionuclide release rate from the repository, the delay prior to groundwater contact with the waste (leach initiation), aquifer flow velocity and flow path length. The nuclide retarding capacity of the geologic media, a major determinant of the isolation effectiveness, was not varied as a parameter but was held constant for a particular reference site. This analysis is limited to looking only at engineered barriers whose net effect is either to delay groundwater contact with the waste form or to limit the rate of release of radionuclides into the groundwater once contact has occurred. The analysis considers only leach incident scenarios, including a water well intrusion into the groundwater near a repository, but does not consider other human intrusion events or catastrophic events. The analysis has so far been applied to a reference salt site repository system and conclusions are presented.Basically, in nearly all cases, the regional geology is the most effective barrier to release of radionuclides to the biosphere; however, for long-lived isotopes of carbon, technetium and iodine, which were poorly sorbed on the geologic media, the geology is not very effective once a leach incident is initiated.

431 (PNL--3365) HANFORD TRANSURANIC STORAGE CORROSION REVIEW. Nelson, J.L.; Divine, J.R. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Dec 1980. Contract AC06-76RL01830. 30p. NTIS, PC A03/MF A01.

The rate of atmospheric corrosion of the transuranic (TRU) waste drums at the US Department of Energy's Hanford Project, near Richland, Washington, was evaluated by Pacific Northwest Laboratory (PNL). The rate of corrosion is principally contingent upon the effects of humidity, airborne pollutants, and temperature. Results of the study indicate that actual penetration of barrels due to atmospheric corrosion will probably not occur within the 20-year specified recovery period. Several other US burial sites were surveyed, and it appears that there is sufficient uncertainty in the available data to prevent a clearcut statement of the corrosion rate at a specific site. Laboratory and site tests are recommended before any definite conclusions can be made. The corrosion potential at the Hanford TRU waste site could be reduced by a combination of changes in drum materials (for example, using galvanized barrels instead of the currently used mild steel barrels), environmental exposure conditions (for example, covering the barrels in one of numerous possible ways), and storage conditions (for example, separating the layers of barrels with slats of wood instead of sheets of plywood).

432 (PNL--3398) ASSESSMENT OF EFFECTIVENESS OF GEOLOGIC ISOLATION SYSTEMS: A SHORT DESCRIPTION OF THE AEGIS APPROACH. Silviera, D.J.; Harwell, M.A.; Napier, B.A.; Zellmer, J.T.; Benson, G.L. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Sep 1980. Contract AC06-76RL01830. 59p. NTIS, PC A04/MF A01.

To meet licensing criteria and protection standards for HLW disposal, research programs are in progress to determine acceptable waste forms, canisters, backfill materials for the repository, and geological formations. Methods must be developed to evaluate the effectiveness of the total system. To meet this need, methods are being developed to assess the long-term effectiveness of isolating nuclear wastes in geologic formations. This work was started in 1976 in the Waste Isolation Safety Assessment Program (WISAP) and continues in the Assessment of Effectiveness of Geologic Isolation Systems (AEGIS) Program. The evaluation of this long-term effectiveness involves a number of distinct steps. AEGIS currently has the methods for performing these evaluation steps. These methods are continuously being improved to meet the inreasing level of sophistication which will be required. AEGIS develops a conceptual description of the geologic systems and uses computer models to simulate the existing groundwater pathways. AEGIS also uses a team of consulting experts, with the assistance of a computer model of the geologic processes, to develop and evaluate plausible release scenarios. Then other AEGIS computer models are used to simulate the transport of radionuclides to the surface and the resultant radiation doses to individuals and populations. (DLC)

433 (PNL--3432) SORPTION OF REDOX-SENSITIVE ELEMENTS: CRITICAL ANALYSIS. Strickert, R.G. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Dec 1980. Contract AC06-76RL01830. 59p. NTIS, PC A04/MF A01.

The redox-sensitive elements (Tc, U, Np, Pu) discussed in this report are of interest to nuclear waste management due to their long-lived isotopes which have a potential radiotoxic effect on man. In their lower oxidation states these elements have been shown to be highly adsorbed by geologic materials occurring under reducing conditions. Experimental research conducted in recent years, especially through the Waste Isolation Safety Assessment Program (WISAP) and Waste/Rock Interaction Technology (WRIT) program, has provided extensive information on the mechanisms of retardation. In general, ion-exchange probably plays a minor role in the sorption behavior of cations of the above three actinide elements. Formation of anionic complexes of the oxidized states with common ligands (OH^-, CO^{--}_3) is expected to reduce adsorption by ion exchange further. Pertechnetate also exhibits little ion-exchange sorption by geologic media. In the reduced (IV) state, all of the elements are highly charged and it appears that they form a very insoluble compound (oxide, hydroxide, etc.) or undergo coprecipitation or are incorporated into minerals. The exact nature of the insoluble compounds and the effect of temperature, pH, pe, other chemical species, and other parameters are currently being investigated. Oxidation states other than Tc (IV,VII), U(IV,VI), Np(IV,V), and Pu(IV,V) are probably not important for the geologic repository environment expected, but should be considered especially when extreme conditions exist (radiation, temperature, etc.). Various experimental techniques such as oxidation-state analysis of tracer-level isotopes, redox potential measurement and control, pH measurement, and solid phase identification have been used to categorize the behavior of the various valence states.

434 (PNL--3447) USE OF CERAMIC MATERIALS IN WASTE-PACKAGE SYSTEMS FOR GEOLOGIC DISPOSAL OF NUCLEAR WASTES. Fullam, H.T. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Dec 1980. Contract AC06-76RL01830. 81p. NTIS, PC A05/MF A01.

A study to investigate the potential use of ceramic materials as components in the waste package systems was conducted. The initial objective of the study was to screen and compare a large number of ceramic materials and identify the best materials for the proposed application. The principal method used to screen the candidates was to subject samples of each material to a series of leaching tests and to determine their relative resistence to attack by the leach solutions. A total of 14 ceramic materials, plus graphite and basalt were evaluated using three different leach solutions: demineralized water, a synthetic Hanford ground water, and a synthetic WIPP brine solution. The ceramic materials screened were Al_2O_3 (99%), Al_2O_3 (99.8%), mullite ($2Al_2O_3.SiO_2$), vitreous silica (SiO_2), $BaTiO_3$, $CaTiO_3$, $CaTiSiO_5$, TiO_2, ZrO_2, $ZrSiO_4$, Pyroceram 9617, and Marcor Code 9658 machinable glass-ceramic. Average leach rates for the materials tested were determined from analyses of the leach solutions and/or sample weight loss measurements. Because of the limited scope of the present study, evaluation of the specimens was limited to ceramographic examination. Based on an overall evaluation of the leach rate data, five of the materials tested, namely graphite, TiO_2, ZrO_2, and the two grades of alumina, exhibited much greater resistance to leaching than did the other materials tested. Based on all the experimental data obtained, and considering other factors such as cost, availability, fabrication technology, and mechanical and physical properties, graphite and alumina are the preferred candidates for the barrier application. The secondary choices are TiO_2 and ZrO_2.

435 (PNL--3484) INVESTIGATION OF METALLIC, CERAMIC, AND POLYMERIC MATERIALS FOR ENGINEERED BARRIER APPLICATIONS IN NUCLEAR-WASTE PACKAGES. Westerman, R.E. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Oct 1980. Contract AC06-76RL01830. 119p. NTIS, PC A06/MF A01.

An effort to develop licensable engineered barrier systems for the long-term (about 1000 yr) containment of nuclear wastes under conditions of deep continental geologic disposal has been underway at Pacific Northwest Laboratory since January 1979, under the auspices of the High-Level Waste Immobilization Program. In the present work, the barrier system comprises the hard or structural elements of the package: the canister, the overpack(s), and the hole sleeve. A number of candidate metallic, ceramic, and polymeric materials were put through mechanical, corrosion, and leaching screening tests to determine their potential usefulness in barrier-system applications. Materials demonstrating adequate properties in the screening tests will be subjected to more detailed property tests, and, eventually, cost/benefit analyses, to determine their ultimate applicability to barrier-system design concepts. The following materials were investigated: two titanium alloys of Grade 2 and Grade 12; 300 and 400 series stainless steels, Inconels, Hastelloy C-276, titanium, Zircaloy, copper-nickel alloys and cast irons; total of 14 ceramic materials, including two grades of alumina, plus graphite and basalt; and polymers such as polyamide-imide, polyarylene, polyimide, polyolefin, polyphenylene sulfide, polysulfone, fluoropolymer, epoxy, furan, silicone, and ethylene-propylene terpolymer (EPDM) rubber. The most promising candidates for further study and potential use in engineered barrier systems were found to be rubber, filled polyphenylene sulfide, fluoropolymer, and furan derivatives.

436 (PNL--3496) DETERMINING CRITERIA FOR THE DISPOSAL OF IODINE-129. Burger, L.L. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Oct 1980. Contract AC06-76RL01830. 33p. NTIS, PC A03/MF A01.

The basic consideration in the disposal of the ^{129}I produced by the nuclear power industry is that humans must be protected from unacceptable radiation risks. Existing standards prescribe maximum concentrations in air and water and, more recently, a maximum release per unit of electrical power production. The global quantity, distribution, and rate of movement of ^{127}I (natural iodine), naturally produced ^{129}I, and anthropogenic ^{129}I are examined. The ^{129}I released earlier as a result of nuclear activities over the past few decades is not uniformly dispersed. But the possibility of much greater dispersion exists and, therefore, of much greater dilution than was previously attempted. The potential for dilution with respect to either the ^{129}I concentration or the $^{129}I/^{127}I$ ratio far exceeds the minimum required for acceptable exposure to mankind. For utilizing the dilution principle, it is preferable to package and dispose of ^{129}I separately from other fission products. The deep ocean is seen to be the logical location for ultimate disposal. A set of 14 basic items is described that can be used to set criteria for storage and disposal of ^{129}I. It is suggested that preliminary standards be developed on these and perhaps other items to apply to (1) temporary storage and transportation, (2) disposal to a dry environment with a time limitation on calculated behavior, and (3) disposal to the deep ocean with complete release permitted in 10^3 yr. Early quantification of some of these items will permit better decisions on further research and development needed for iodine removal or control, fixation, and disposal.

437 (PNL--3542) GEOLOGIC SIMULATION MODEL FOR A HYPOTHETICAL SITE IN THE COLUMBIA PLATEAU. Petrie, G.M.; Zellmer, J.T.; Lindberg, J.W.; Foley, M.G. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Apr 1981. Contract AC06-76RL01830. 603p. NTIS, PC A99/MF A01. Order Number DE81027687.

This report describes the structure and operation of the Assessment of Effectiveness of Geologic Isolation Systems (AEGIS) Geologic Simulation Model, a computer simulation model of the geology and hydrology of an area of the Columbia Plateau, Washington. The model is used to study the long-term suitability of the Columbia Plateau Basalts for the storage of nuclear waste in a mined repository. It is also a starting point for analyses of such repositories in other geologic settings. The Geologic Simulation Model will aid in formulating design disruptive sequences (i.e. those to be used for more detailed hydrologic, transport, and dose analyses) from the spectrum of hypothetical geological and hydrological developments that could result in transport of radionuclides out of a repository. Quantitative and auditable execution of this task, however, is impossible without computer simulation. The computer simulation model aids the geoscientist by generating the wide spectrum of possible future evolutionary paths of the areal geology and hydrology, identifying those that may affect the repository integrity. This allows the geoscientist to focus on potentially disruptive processes, or series of events. Eleven separate submodels are used in the simulation portion of the model: Climate, Continental Glaciation, Deformation, Geomorphic Events, Hydrology, Magmatic Events, Meteorite Impact, Sea-Level Fluctuations, Shaft-Seal Failure, Sub-Basalt Basement Faulting, and Undetected Features. Because of the modular construction of the model, each submodel can easily be replaced with an updated or modified version as new information or developments in the state of the art become available. The model simulates the geologic and hydrologic systems of a hypothetical repository site and region for a million years following repository decommissioning. The Geologic Simulation Model operates in both single-run and Monte Carlo modes.

438 (PNL--3556) GRAPHITE MATRIX MATERIALS FOR NUCLEAR WASTE ISOLATION. Morgan, W.C. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jun 1981. Contract AC06-76RL01830. 42p. NTIS, PC A03/MF A01. Order Number DE81026322.

At low temperatures, graphites are chemically inert to all but the strongest oxidizing agents. The raw materials from which artificial graphites are produced are plentiful and inexpensive. Morover, the physical properties of artificial graphites can be varied over a very wide range by the choice of raw materials and manufacturing processes. Manufacturing processes are reviewed herein, with primary emphasis on those processes which might be used to produce a graphite matrix for the waste forms. The approach, recommended herein, involves the low-temperature compaction of a finely ground powder produced from graphitized petroleum coke. The resultant compacts should have fairly good strength, low permeability to both liquids and gases, and anisotropic physical properties. In particular, the anisotropy of the thermal expansion coefficients and the thermal conductivity should be advantageous for this application. With two possible exceptions, the graphite matrix appears to be superior to the metal alloy matrices which have been recommended in prior studies. The two possible exceptions are the requirements on strength and permeability; both requirements will be strongly influenced by the containment design, including the choice of materials and the waste form, of the multibarrier package. Various methods for increasing the strength, and for decreasing the permeability of the matrix, are reviewed and discussed in the sections in Incorporation of Other Materials and Elimination of Porosity. However, it would be premature to recommend a particular process until the overall multi-barrier design is better defined. It is recommended that increased emphasis be placed on further development of the low-temperature compacted graphite matrix concept.

439 (PNL--3559) INITIAL WASTE PACKAGE INTERACTION TESTS: STATUS REPORT. Shade, J.W.; Bradley, D.J. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Dec 1980. Contract AC06-76RL01830. 83p. NTIS, PC A05/MF A01.

This report describes the results of some initial investigations of the effects of rock media on the release of simulated fission products from a sngle waste form, PNL reference glass 76-68. All tests assemblies contained a minicanister prepared by pouring molten, U-doped 76-68 glass into a 2-cm-dia stanless steel tube closed at one end. The tubes were cut to 2.5 to 7.5 cm in length to expose a flat glass surface rimmed by the canister wall. A cylindrical, whole rock pellet, cut from one of the rock materials used, was placed on the glass surface then both the canister and rock pellet were packed in the same type of rock media ground to about 75 μm to complete the package. Rock materials used were a quartz monzonite basalt and bedded salt. These packages were run from 4 to 6 weeks in either 125 ml digestion bombs or 350 ml autoclaves capable of direct solution sampling, at either 250 or 150°C. Digestion bomb pressures were the vapor pressure of water, 600 psig at 250°C, and the autoclaves were pressurized at 2000 psig with an argon overpressure. In general, the solution chemistry of these initial package tests suggests that the rock media is the dominant controlling factor and that rock-water interaction may be similar to that observed in some geothermal areas. In no case was uranium observed in solution above 15 ppB. The observed leach rates of U glass not in contact with potential sinks (rock surfaces and alteration products) have been observed to be considerably higher. Thus the use of leach rates and U concentrations observed from binary leach experiments (waste-form water only) to ascertain long-term environmental consequences appear to be quite conservative compared to actual U release in the waste package experiments. Further evaluation, however, of fission product transport behavior and the role of alteration phases as fission product sinks is required.

440 (PNL--3569) CORROSION RESISTANCE OF CAST IRONS AND TITANIUM ALLOYS AS REFERENCE ENGINEERED METAL BARRIERS FOR USE IN BASALT GEOLOGIC STORAGE: A LITERATURE ASSESSMENT. Charlot, L.A.; Westerman, R.E. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jul 1981. Contract AC06-76RL01830. 72p. NTIS, PC A04/MF A01. Order Number DE81026840.

A survey and assessment of the literature on the corrosion resistance of cast irons and low-alloy titanium are presented. Selected engineering properties of cast iron and titanium are briefly described; however, the corrosion resistance of cast iron and titanium in aqueous solutions or in soils and their use in a basalt repository are emphasized. In evaluating the potential use of cast iron and titanium as structural barrier materials for long-lived nuclear waste packages, it is assumed that titanium has the general corrosion resistance to be used in relatively thin cross sections whereas the cost and availability of cast iron allows its use even in very thick cross sections. Based on this assumption, the survey showed that: The uniform corrosion of low-alloy titanium in a basalt environment is expected to be extremely low. A linear extrapolation of general corrosion rates with an added corrosion allowance suggests that a 3.2- to 6.4-mm-thick wall may have a life of 1000 yr. Pitting and crevice corrosion are not likely corrosion modes in basalt ground waters. It is also unlikely that stress corrosion cracking (SCC) will occur in the commercially pure (CP) titanium alloy or in palladiumor molybdenum-alloyed titanium materials. Low-alloy cast irons may be used as barrier metals if the environment surrounding the metal keeps the alloy in the passive range. The solubility of the corrosion product and the semipermeable nature of the oxide film allow significant uniform corrosion over long time periods. A linear extrapolation of high-temperature corrosion rates on carbon steels and corrosion rates of cast irons in soils gives an estimated metal penetration of 51 to 64 mm after 1000 yr. A corrosion allowance of 3 to 5 times that suggests that an acceptable cast iron wall may be from 178 to 305 mm thick. Although they cannot be fully assessed, pitting and crevice corrosion should not affect cast iron due to the ground-water chemistry of basalt.

441 (PNL--3678) NUCLEAR WASTE MANAGEMENT AND ENVIRONMENTAL MEDIATION: AN EXPLORATORY ANALYSIS. TOPICAL REPORT. Greene, M.R.; Lindell, M.K.; Nealey, S.M.; Drexler, J.A. Jr. (Battelle Human Affairs Research Center, Seattle, WA (USA)). Sep 1980. Contract AC06-76RL01830. 45p. NTIS, PC A03/MF A01. Order Number DE81024016.

Two types of mediation efforts are identified: settlement-oriented mediation and participation-oriented mediation. A range of environmental mediation efforts that have taken place to date are discussed. Within the context of the two identified types of mediation, these characteristics are discussed for the waste management controversy. Emphasis was placed on the complexity of the issues and the range of participants. Also discussed are the relationship between an environmental mediation effort and alternative mechanisms for conflict resolution, such as NEPA based litigation, consultation and concurrence and state veto or federal preemption. Participation-oriented mediation may be more suitable than settlement-oriented mediation for encouraging constructive communication and reducing conflict among participants in the controversy. Several limitations to participation-oriented mediation need to be considered. One is that environmental mediation is such a very new field that it might not be possible to find an experienced mediator willing to attempt such a complex problem. Two is the compatibility of participation-oriented and settlement-oriented mediation.

442 (PNL--3698) RADIONUCLIDE DISEQUILIBRIA STUDIES FOR INVESTIGATING THE INTEGRITY OF POTENTIAL NUCLEAR WASTE DISPOSAL SITES: SUBSEABED STUDIES. Laul, J.C.; Thomas, C.W.; Petersen, M.R.; Perkins, R.W. (Pacific Northwest Lab., Richland, WA (USA)). Sep 1981. Contract AC06-76RL01830. 24p. NTIS, PC A02/MF A01. Order Number DE82010339.

This study of subseabed sediments indicates that natural radionuclides can be employed to define past long-term migration rates and thereby evaluate the integrity of potential disposal sites in ocean sediments. The study revealed the following conclusions: (1) the sedimentation rate of both the long and short cores collected in the North Pacific is 2.5 mm/1000 yr or 2.5 m/m.yr in the upper 3 meters; (2) the sedimentation rate has been rather constant over the last one million years; and (3) slow diffusive processes dominate within the sediment. Reworking of the sediment by physical processes or organisms is not observed.

443 (PNL--3700(Pt.2), pp 115) ASSESSMENT OF EFFECTIVENESS OF GEOLOGIC ISOLATION SYSTEMS. Soldat, J.K.; Napier, B.A.; Strenge, D.L.; Schreckhise, R.G.; Zimmerman, M.G. Feb 1981. NTIS, PC A09/MF A01.

Pacific Northwest Laboratory annual report for 1980 to the DOE Assistant Secretary for Environment. Part 2. Ecological sciences.

The program for Assessment of Effectiveness of Geologic Isolation Systems (AEGIS) is managed through PNL's Water and Land Resources Department and is funded through the Battelle Office of Nuclear Waste Isolation (ONWI). The Ecological Sciences Department was involved in two subtasks under AEGIS: Dose Methodology Development and Reference Site Initial Analysis (RSIA) for a Salt Dome.

444 (PNL--3700(Pt.2), pp 117) HANFORD DEFENSE WASTE STUDIES. Napier, B.A.; Zimmerman, M.G.; Soldat, J.K. Feb 1981. NTIS, PC A09/MF A01.

Pacific Northwest Laboratory annual report for 1980 to the DOE Assistant Secretary for Environment. Part 2. Ecological sciences.

PNL is assisting Rockwell Hanford Operations to prepare a programmatic environmental impact statement for the management of Hanford defense nuclear waste. The Ecological Sciences Department is leading the task of calculation of public radiation doses from a large matrix of potential routine and accidental releases of radionuclides to the environment.

445 (PNL--3700(Pt.2), pp 123-124) ROCKWELL SUPPORT STUDIES. Cadwell, L.L.; Cline, J.F.; Gano, K.A.; Warren, J.L.; Rogers, L.E. Feb 1981. NTIS, PC A09/MF A01.

Pacific Northwest Laboratory annual report for 1980 to the DOE Assistant Secretary for Environment. Part 2. Ecological sciences.

Studies performed for the Rockwell Hanford Operations (RHO) were designed to either identify the role of biota in the uptake and transport of radionuclides from low-level waste management areas or design and/or evaluate methods for reducing biological transport of radionuclides away from waste management areas.

446 (PNL--3720) MATERIALS CHARACTERIZATION CENTER WORKSHOP ON CORROSION OF ENGINEERED BARRIERS. Merz, M.D.; Zima, G.E.; Jones, R.H.; Westerman, R.E. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Mar 1981. Contract AC06-76RL01830. 72p. NTIS, PC A04/MF A01.

A workshop on corrosion test procedures for materials to be used as barriers in nuclear waste repositories was conducted August 19 and 20, 1980, at the Battelle Seattle Research Center. The purpose of the meeting was to obtain guidance for the Materials Characterization Center in preparing test procedures to be approved by the Materials Review Board. The workshop identified test procedures that address failure modes of uniform corrosion, pitting and crevice corrosion, stress corrosion, and hydrogen effects that can cause delayed failures. The principal areas that will require further consideration beyond current engineering practices involve the analyses of pitting, crevice corrosion, and stress corrosion, especially with respect to quantitative predictions of the lifetime of barriers. Special techniques involving accelerated corrosion testing for uniform attack will require development.

447 (PNL--3798) RISK PERCEPTION, RISK EVALUATION AND HUMAN VALUES: COGNITIVE BASES OF ACCEPTABILITY OF A RADIOACTIVE WASTE REPOSITORY. Earle, T.C.; Lindell, M.K.; Rankin, W.L. (Battelle Human Affairs Research Center, Seattle, WA (USA)). Jul 1981. Contract AC06-76RL01830. 100p. (BHARC--411/81/007). NTIS, PC A05/MF A01. Order Number DE82001044.

Public acceptance of radioactive waste management alternatives depends in part on public perception of the associated risks. Three aspects of those perceived risks were explored in this study: (1) synthetic measures of risk perception based on judgments of probability and consequences; (2) acceptability of hypothetical radioactive waste policies, and (3) effects of human values on risk perception. Both the work on synthetic measures of risk perception and on the acceptability of hypothetical policies included investigations of three categories of risk: (1) Short-term public risk (affecting persons living when the wastes are created), (2) Long-term public risk (affecting persons living after the time the wastes were created), and (3) Occupational risk (affecting persons working with the radioactive wastes). The human values work related to public risk perception in general, across categories of persons affected. Respondents were selected according to a purposive sampling strategy.

448 (PNL--3799) PUBLIC PERCEPTIONS OF INDUSTRIAL RISKS: THE CONTEXT OF PUBLIC ATTITUDES TOWARD RADIOACTIVE WASTE. Earle, T.C. (Battelle Human Affairs Research Center, Seattle, WA (USA)). Jun 1981. Contract AC06-76RL01830. 76p. (BHARC--411). NTIS, PC A05/MF A01. Order Number DE81030676.

A survey was made to determine the public risk perception of several industrial hazards. A free response approach was used in order for respondents to generate their own alternatives. The general class of hazard investigated here included all hazardous industrial facilities. The free response survey was used to study public perception of: (a) the closeness of the nearest hazardous industrial facility (as estimated by the respondent); (b) the sort of facility it is; (c) the sorts of risk associated with it; and (d) the persons placed at risk by it. Respondents also identified the risks of, and the persons placed at risk by, both a toxic chemical disposal facility and a nuclear waste disposal facility. Results of this study thus can inform us of the unprompted concerns of the public regarding a wide variety of industrial facilities.

449 (PNL--3805) MIGRATION OF STRONTIUM-90 FROM A STRONTIUM-90 FLUORIDE DEEP OCEAN SOURCE. Yabusaki, S. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jun 1981. Contract AC06-76RL01830. 27p. NTIS, PC A03/MF A01. Order Number DE81023670.

A hypothetical rupture of a heat source capsule on the ocean floor is analyzed for strontium-90 migration and attenuation. The evolution of the three-dimensional contaminant plume is simulated by a modified version of the Okubo-Pritchard radially symmetrical, diffusion velocity dispersion model. Results from this study indicate that released solutes are confined vertically to a layer near the level of introduction. Along the plume centerline, however, water quality is affected for considerably distances downstream from the source, with the maximum effect occurring after one year.

450 (PNL--3844) STRONTIUM AND CESIUM RADIONUCLIDE LEAK DETECTION ALTERNATIVES IN A CAPSULE STORAGE POOL. Larson, D.E.; Crawford, T.W.; Joyce, S.M. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Aug 1981. Contract AC06-76RL01830. 93p. NTIS, PC A05/MF A01. Order Number DE81023611.

A study was performed to assess radionuclide leak-detection systems for use in locating a capsule leaking strontium-90 or cesium-137 into a water-filled pool. Each storage pool contains about 35,000 L of water and up to 715 capsules, each of which contains up to 150 kCi strontium-90 or 80 kCi cesium-137. Potential systems assessed included instrumental chemical analyses, radionuclide detection, visual examination, and other nondestructive nuclear-fuel examination techniques. Factors considered in the assessment include: cost, simplicity of maintenance and operation, technology availability, reliability, remote operation, sensitivity, and ability to locate an individual leaking capsule in its storage location. The study concluded that an adaption of the spent nuclear-fuel examination technique of wet sipping be considered for adaption. In the suggested approach, samples would be taken continuously from pool water adjacent to the capsule(s) being examined for remote radiation detection. In-place capsule isolation and subsequent water sampling would confirm that a capsule was leaking radionuclides. Additional studies are needed before implementing this option. Two other techniques that show promise are ultrasonic testing and eddy-current testing.

451 (PNL--3873) DEVELOPMENT OF BACKFILL MATERIAL AS AN ENGINEERED BARRIER IN THE WASTE PACKAGE SYSTEM. INTERIM TOPICAL REPORT. Wheelwright, E.J.; Hodges, F.N.; Bray, L.A.; Westsik, J.H. Jr.; Lester, D.H.; Nakai, T.L.; Spaeth, M.E.; Stula, R.T. (Pacific Northwest Lab., Richland, WA (USA)). Sep 1981. Contract AC06-76RL01830. 105p. NTIS, PC A06/MF A01. Order Number DE82000937.

A backfill barrier, emplaced between the containerized waste and the host rock, can both protect the other engineered barriers and act as a primary barrier to the release of radionuclides from the waste package. Attributes that a backfill should provide in order to carry out its required function have been identified. Primary attributes are those that have a direct effect upon the release and transport of radionuclides from the waste package. Supportive attributes do not directly affect radionuclide release but are necessary to support the primary attributes. The primary attributes, in order of importance, are: minimize (retard or exclude) the migration of ground water between the host rock and the waste canister system; retard the migration of selected chemical species (corrosive species and radionuclides) in the ground water; control the Eh and pH of the ground water within the waste-package environment. The supportive attributes are: self-seal any cracks or discontinuities in the backfill or interfacing host geology; retain performance properties at all repository temperatures; retain peformance properties during and after receiving repository levels of gamma radiation; conduct heat from the canister system to the host geology; retain mechanical properties and provide resistance to applied mechanical forces; retain morphological stability and compatibility with structural barriers and with the host geology for required period of time. Screening and selection of candidate backfill materials has resulted in a preliminary list of materials for testing. Primary emphasis has been placed on sodium and calcium bentonites and zeolites used in conjunction with quartz sand or crushed host rock. Preliminary laboratory studies have concentrated on permeability, sorption, swelling pressure, and compaction properties of candidate backfill materials.

452 (PNL--3898) APPLICATION OF A SITE-BINDING, ELECTRICAL, DOUBLE-LAYER MODEL TO NUCLEAR WASTE DISPOSAL. Relyea, J.F.; Silva, R.J. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Sep 1981. Contract AC06-76RL01830. 28p. NTIS, PC A03/MF A01.

Order Number DE81030231.

A site-binding, electrical, double-layer adsorption model has been applied to adsorption of Cs for both a montmorillonite clay and powdered SiO_2. Agreement between experimental and predicted results indicates that C/sub s/+ is adsorbed by a simple cation-exchange mechanism. Further application of a combination equilibrium thermodynamic model and site-binding, electrical, double-layer adsorption model has been made to predict the behavior of U(VI) in solutions contacting either the montmorillonite clay or powdered SiO_2. Experimentally determined U solution concentrations have been used to select what is felt to be the best available thermodynamic data for U under oxidizing conditions. Given the existing information about the probable U solution species, it was possible to determine that UO_2^{+2} is most likely adsorbed by cation-exchange at pH 5. At higher values (pH 7 and 9), it was shown that $UO_2(OH)_2^0$ is probably the most strongly adsorbed U solution species. It was also found that high NaCl solution concentrations at higher pH values lowered U concentrations (either because of enhanced sorption or lowered solubility); however, the mechanism responsible for this behavior has not been determined.

453 (PNL--3913) ASSESSMENT OF RADIATION EFFECTS IN DEFENSE TRANSURANIC WASTE FORMS. Roberts, F.P. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jul 1981. Contract AC06-76RL01830. 43p. NTIS, PC A03/MF A01. Order Number DE81029432.

The actinide concentrations of the defense transuranic (TRU) wastes were analyzed and the potential effects of the radiation on the properties of the wastes after conversion to immobile forms were assessed. The study focused on the contact-handled retrievably-stored wastes. The major components of the current inventory are defense plutonium-contaminated wastes containing various amounts of ^{241}Am. The wastes stored at Idaho National Engineering Laboratory (INEL) are typical of the wastes in this category. There is also a substantial amount of wastes contaminated with plutonium enriched in ^{238}Pu arising from the Department of Energy's isotopic heat-source programs. Most of these wastes are stored at the Los Alamos National Laboratory (LANL) and the Savannah River Plant (SRP) sites. Four reference wastes were selected representing a credible range of actinide activities and were used for estimating radiation doses to the final waste forms based on: INEL first stage sludge, a composite of all wastes at INEL, a composite of all wastes at LANL including both defense and heat source plutonium wastes, and a composite of all heat-source plutonium wastes at SRP free from defense plutonium. From integrated alpha and beta-gamma doses over a 10^5y storage period, it is concluded that: accumulated beta-gamma doses of 10^6 to 10^8 rad over a 10^5 y storage period will not significantly change physical properties of the waste form. The alpha decay doses are 3-30 x 10^{18} decays/cm^3 (up to 3 x 10^{10} rad ionizing radiation from alpha particles) accumulated over a 10^5 y storage period. Radiolytic gas generation can be substantial in waste forms containing water or organic materials. The effect of leachant radiolysis on the leachabilities of TRU waste forms is not sufficiently understood to rule out the need for testing with actinide-doped specimens.

454 (PNL--3915) INVESTIGATION OF SUSCEPTIBILITY OF TITANIUM-GRADE 2 AND TITANIUM-GRADE 12 TO ENVIRONMENTAL CRACKING IN A SIMULATED BASALT REPOSITORY ENVIRONMENT. Pitman, S.G. (Pacific Northwest Lab., Richland, WA (USA)). Oct 1981. Contract AC06-76RL01830. 53p. NTIS, PC A04/MF A01. Order Number DE82003233.

Slow strain rate tests and fatigue crack growth rate tests were used to evaluate the relative susceptibility of titanium (Ti)-grade 2 and -grade 12 to cracking in a simulated repository environment. Slow strain rate tests were performed in air and simulated Hanford basaltic groundwater at 250°C. Fatigue crack growth rate tests were done in Hanford groundwater, fluoride-ion-enhanced Hanford groundwater, and high-purity water at 90°C. The following conclusions can be drawn: Ti-grade 2 and Ti-grade 12 exhibited strain-rate-dependent degradation of ductility in slow strain rate tests at 250°C. In-air tests confirmed that the loss of ductility is not limited to the simulated repository environment and may be caused by an internal strengthening or embrittlement mechanism such as dynamic strain aging. The ductility degradation in Ti-grade 12 was found to be highly orientation dependent. (Only one orientation of grade 2 was tested). No fractographic evidence of an environmental cracking mechanism was found in the slow strain rate tests; all specimens failed by micro-void coalescence. The fatigue crack growth rate of Ti-grade 2 and -grade 12 was not affected relative to air or high-purity water by any of the environments used, and no frequency dependence was observed. This may indicate that no environmental cracking mechanism relative to a Hanford basalt repository is operating significantly under the conditions (high oxygen fugacity and a temperature of 90°C).

455 (PNL--3954) RADIATION DOSE RATES FROM COMMERCIAL PWR AND BWR SPENT FUEL ELEMENTS. Willingham, C.E. (Pacific Northwest Lab., Richland, WA (USA)). Oct 1981. Contract AC06-76RL01830. 88p. NTIS, PC A05/MF A01. Order Number DE82000697.

Data on measurements of gamma dose rates from commercial reactor spent fuel were collected, and documented calculated gamma dose rates were reviewed. As part of this study, the gamma dose rate from spent fuel was estimated, using computational techniques similar to previous investigations into this problem. Comparison of the measured and calculated dose rates provided a recommended dose rate in air versus distance curve for PWR spent fuel.

456 (PNL--4022) LEACHING BEHAVIOR OF GLASS CERAMIC NUCLEAR WASTE FORMS. Lokken, R.O. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Nov 1981. Contract AC06-76RL01830. 72p. NTIS, PC A04/MF A01. Order Number DE82005595.

Glass ceramic waste forms have been investigated as alternatives to borosilicate glasses for the immobilization of high-level radioactive waste at Pacific Northwest Laboratory (PNL). Three glass ceramic systems were investigated, including basalt, celsian, and fresnoite, each containing 20 wt % simulated high-level waste calcine. Static leach tests were performed on seven glass ceramic materials and one parent glass (before recrystallization). Samples were leached at 90°C for 3 to 28 days in deionized water and silicate water. The results, expressed in normalized elemental mass loss, (g/m^2), show comparable releases from celsian and fresnoite glass ceramics. Basalt glass ceramics demonstrated the lowest normalized elemental losses with a nominal release less than 2 g/m^2 when leached in polypropylene containers. The releases from basalt glass ceramics when leached in silicate water were nearly identical with those in deionized water. The overall leachability of celsian and fresnoite glass ceramics was improved when silicate water was used as the leachant.

457 (PNL--4100-Pt.2, pp 145) ASSESSMENT OF EFFECTIVENESS OF GEOLOGIC ISOLATION SYSTEMS. Soldat, J.K.; Napier, B.A. Feb 1982. NTIS, PC A08/MF A01.

Annual Report for 1981 to the DOE Office of the Assistant Secretary for Environmental Protection, Safety, and Emergency Preparedness. Part 2. Ecological Sciences.

The program for Assessment of Effectiveness of Geologic Isolation Systems (AEGIS) is managed through Pacific Northwest Laboratory's Water and Land Resources Department and is funded through the Battelle Office of Nuclear Waste Isolation (ONWI). The Ecological Sciences Department was involved in two subtasks under AEGIS: Dose Methodology Development and Reference Site Initial Analysis (RSIA) for a Salt Dome.

458 (PNL--4100-Pt.2, pp 147) HANFORD DEFENSE WASTE STUDIES. Napier, B.A.; Watson, E.C. Feb 1982. NTIS, PC A08/MF A01.

Annual Report for 1981 to the DOE Office of the Assistant Secretary for Environmental Protection, Safety, and Emergency Preparedness. Part 2. Ecological Sciences.

PNL is assisting DOE in preparation of a programmatic environmental documentation on the potential strategies for managing Hanford Defense nuclear waste. The Ecological Sciences Department is performing the subtasks of developing guidance on allowable amounts of residual environmental contamination and calculating potential public health and safety impacts from proposed operations.

459 (PNL--4100-Pt.2, pp 153) STANDARDIZED INPUT FOR HANFORD ENVIRONMENTAL IMPACT STATEMENTS. Mueller, M.A.; Jamison, J.D.; Napier, B.A. Feb 1982. NTIS, PC A[illegible]/MF A01.

Annual Report for 1981 to the DOE Office of the Assistant Secretary for Environmental Protection, Safety, and Emergency Preparedness. Part 2. Ecological Sciences.

The DOE Waste Management Program has required that environmental impact statements (EIS) be prepared for a variety of waste management activities at the Hanford site. These EISs are prepared in accordance with the Council on Environmental Quality (CEQ) and DOE guidelines by several different lead contractors. The objective of this project, sponsored by the DOE Waste Management Program, is to reduce redundancy of effort and promote uniformity in EISs by preparing standardized sections and methods that are acceptable to DOE's Richland Operations Office and the Hanford contractors.

460 (PNL--4100-Pt.2, pp 159) PRELIMINARY RADIATION DOSE ANALYSIS FOR REFERENCE FUSRAP WASTE SITES. Kennedy, W.E. Jr.; Aaberg, R.L. Feb 1982. NTIS, PC A08/MF A01.

Annual Report for 1981 to the DOE Office of the Assistant Secretary for Environmental Protection, Safety, and Emergency Preparedness. Part 2. Ecological Sciences.

The purpose of this study is to provide Bechtel National at Oak Ridge, Tennessee, with a preliminary radiation dose analysis of buried radioactive waste for the formerly utilized site remedial action program (FUSRAP). Bechtel National will use the preliminary results from this analysis to identify important scenarios, radionuclides, and exposure pathways for site-specific cost/benefit analyses.

461 (PNL-SA--6693) KRYPTON ENTRAPMENT IN PULSE BIASED SPUTTER-DEPOSITED METALS. Bayne, M.A.; Moss, R.W.; McClanahan, E.D. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 1978. Contract EY-76-C-06-1830. 27p. (CONF-780430--7). Dep. NTIS, PC A03/MF A01.

From Conference on metallurgical coatings; San Francisco, CA, USA (3 Apr 1978).

A supported gas discharge sputtering system was constructed to investigate krypton entrapment in high-rate sputter-deposited thick films. Krypton entrapment was studied as a function of substrate and target voltage for nickel, aluminum, titanium and iron deposits. Control of substrate and target voltages was achieved with pulse circuits capable of adjusting the pulse duration and repetition rate. A water-cooled cylindrical copper substrate 15 cm in diameter was used to collect the sputtered metal from a 9-cm diameter cylindrical target. To observe the immediate effect of changes in sputtering parameters on gas trapping, as well as measure the total gas trapped, a sensitive mass flow indicator was installed in the krypton supply of the dynamically pumped system. Data relating krypton content to substrate bias conditions and deposition rate are discussed in light of the theory of Carter, Colligon and Leck on ion absorption in the presence of sputtering. Krypton contents > 2 at.% were found in the metals studied, all of which were deposited at rates exceeding 1 nm/sec. Computed krypton concentrations agreed with concentrations in samples that were analyzed chemically.

462 (PNL-SA--6957) WASTE ISOLATION SAFETY ASSESSMENT PROGRAM. TASK 4: CONTRACTOR INFORMATION MEETING PROCEEDINGS. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 1977. Contract EY-76-C-03-1830. 606p. (CONF-7709157--). Dep. NTIS, PC A99/MF A01.

From Contractor information meeting; Seattle, WA, USA (20 Sep 1977).

The predominant topic is radionuclide migration and absorption-desorption from a deep geological repository. Research programs and the development of experimental methods are described. Separate abstracts were prepared for the 15 technical reports which are included in full; there are also discussions of guest lectures, informal presentations, discussions, and a summary of the meeting. (DLC)

463 (PNL-SA--6957, pp 6-43) OVERVIEW OF THE OFFICE OF WASTE ISOLATION'S TECHNICAL SUPPORT PROJECTS TO SUPPORT THE DESIGN, CONSTRUCTION, AND LICENSING OF A DEEP GEOLOGICAL REPOSITORY FOR NUCLEAR WASTE. Dole, L.R. (Union Carbide Corp., Oak Ridge, TN). 1977.

From Symposium on power plant heat rejection; Washington, DC, USA (12 Sep 1977).

Twenty-nine viewgraphs outline goals for deep geological disposal, schedule for repository No. 1, technical support projects, and quality assurance requirements.

464 (PNL-SA--6957, pp 44-56) OVERVIEW OF PNL'S WASTE ISOLATION SAFETY ASSESSMENT PROGRAM (WISAP). Burkholder, H.C. 1977.

From Contractor information meeting; Seattle, WA, USA (20 Sep 1977).

The major objectives of the PNL Waste Isolation Safety Assessment Program (WISAP) were described. In FY 77, WISAP was divided into six specific but interrelated tasks. Conceptually the interrelations and final products from each task are shown. Major activities within each task are described. 8 figs.

465 (PNL-SA--6957, pp 57-87) OVERVIEW OF TASK 4 NUCLIDE TRANSPORT DATA. Serne, R.J. 1977.
From Contractor information meeting; Seattle, WA, USA (20 Sep 1977).
A multi-year program plan is presented to collect the necessary data on nuclide sorption-desorption interactions with geologic media. Detailed activities which need to be performed in each of the six subtasks are described. The general areas in which each subcontractor performed work in FY 77 were presented in the overview. Detailed technical discussions of each subcontractor's work will be presented in ensuing presentations.

466 (PNL-SA--7184) METHODOLOGY FOR ASSESSING THE LONG-TERM SAFETY OF RADIOACTIVE WASTE STORAGE IN GEOLOGIC FORMATIONS. Brandstetter, A. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Sep 1978. Contract EY-76-C-06-1830. 4p. (CONF-7809198--1). Dep. NTIS, PC A02/MF A01.
From Meeting on migration of radionuclides in crystalline rocks; Studsvik, Sweden (11 Sep 1978).
The development of the safety assessment methodology currently consists of four major tasks: release scenario analysis, waste release studies, release consequence analysis, and radionuclide sorption studies. Each of these tasks and plans for each are discussed. International cooperation on these tasks is urged. (DLC)

467 (PNL-SA--7197) WORKSHOP ON POTENTIALLY DISRUPTIVE PHENOMENA FOR NUCLEAR WASTE REPOSITORIES, JULY 27-29, 1977. Jacobson, J.J. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 1977. Contract EY-76-C-06-1830. 37p. Dep. NTIS, PC A03/MF A01.
The workshop on Potentially Disruptive Phenomena for Nuclear Waste Repositories brought together experts in the geosciences to identify and evaluate potentially disruptive events and processes and to contribute ideas on how to extrapolate data from the past into the next one million years. The analysis is to be used to model a repository in geologic media for long-term safety assessments of nuclear waste storage. The workshop included invited presentations on the following items: an overview of the Waste Isolation Safety Assessment Program (WISAP), simulation techniques, subjective probabilities and methodology of obtaining data, similar modeling efforts at Lawrence Livermore and Sandia Laboratories, and geologic processes or events.

468 (PNL-SA--7243) WASTE ISOLATION SAFETY ASSESSMENT PROGRAM. Brandstetter, A.; Harwell, M.A. (Battelle Pacific Northwest Labs., Richland, WA (USA)). May 1979. Contract EY-76-C-06-1830. 15p. (IAEA-SM--243/35; CONF-790711--4). Dep. NTIS, PC A02/MF A01.
From International symposium on underground disposal of radioactive wastes; Otaniemi, Finland (2 Jul 1979).
Associated with commercial nuclear power production in the United States is the generation of potentially hazardous radioactive wastes. The Department of Energy (DOE), through the National Waste Terminal Storage (NWTS) Program, is seeking to develop nuclear waste isolation systems in geologic formations that will preclude contact with the biosphere of waste radionuclides in concentrations which are sufficient to cause deleterious impact on humans or their environments. Comprehensive analyses of specific isolation systems are needed to assess the expectations of meeting that objective. The Waste Isolation Safety Assessment Program (WISAP) has been established at the Pacific Northwest Laboratory (operated by Battelle Memorial Institute) for developing the capability of making those analyses. Among the analyses required for isolation system evaluation is the detailed assessment of the post-closure performance of nuclear waste repositories in geologic formations. This assessment is essential, since it is concerned with aspects of the nuclear power program which previously have not been addressed. Specifically, the nature of the isolation systems (e.g., involving breach scenarios and transport through the geosphere), and the time-scales necessary for isolation, dictate the development, demonstration and application of novel assessment capabilities. The assessment methodology needs to be thorough, flexible, objective, and scientifically defensible. Further, the data utilized must be accurate, documented, reproducible, and based on sound scientific principles.

469 (PNL-SA--7352(Vol.1), pp 231-329) SYSTEMATIC STUDY OF METAL ION SORPTION ON SELECTED GEOLOGIC MEDIA. SESQUIANNUAL REPORT, APRIL 1977-SEPTEMBER 1978. Meyer, R.E. (comp.). (Oak Ridge National Lab., TN). 1978. Dep. NTIS, PC A22/MF A01.
From 2. contractor information meeting; Seattle, WA, USA (1 Oct 1978).
Results of the first year and a half of a research program on equilibria of inorganic ions between selected minerals and aqueous solutions covering a wide range of composition are summarized. The work is motivated by needs of information for enhanced oil recovery and isolation of nuclear waste, as well as by interest in the fundamental chemistry. A method using an axial filter configuration was adapted to these adsorbents. A technique of supporting clay on filteraid (diatomaceous earth) of low adsorptive properties was developed to allow measurements with columns of dispersible adsorbents, such as montmorillonite. These approaches, which are advantageous under some conditions, were shown to give distribution coefficients (D: adsorbate per weight of clay/adsorbate concentration in equilibrium solution) concordant with values obtained by batch equilibrations. Equilibria of non-hydrolyzed, non-complexed cations and clays were found to be in accord with ion-exchange predictions to a good approximation over a wide range of conditions. As expected on this basis, distribution coefficients of ions present in low concentration declined rapidly as salinity increased. Values of D for many ions at trace levels were the same as at moderate fractional loadings of the ion-exchange capacity. Exceptions were alkali metal ions on the calcium form of montmorillonite, for which decreases of D with loading were found to occur at unexpectedly low loadings, and the dependence of D's of rare earth and actinide (III) ions on salinity, for which a minimum or levelling was found at moderate ionic strength. Adsorption of Cs(I) and Sr(II) on rock samples were much lower from solutions of high salinity than from solutions of low salinity; D-values of Pu, present in unknown species, were erratic.

470 (PNL-SA--7352(Vol.1)) PROCEEDINGS OF THE TASK 4 WASTE ISOLATION SAFETY ASSESSMENT

PROGRAM SECOND CONTRACTOR INFORMATION MEETING. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 1979. Contract EY-76-C-06-1830. 522p. (CONF-7810227--(Vol.1)). Dep. NTIS, PC A22/MF A01.

From 2. contractor information meeting; Seattle, WA, USA (1 Oct 1978).

Volume I contains the following papers: Solution Species of ^{239}Pu in Oxidizing Environments; Solution Species of ^{239}Pu in the Environment; Theoretical and Experimental Evaluation of Waste Transport in Selected Rocks; Studies of Radionuclide Availability and Migration at the Nevada Test Site Relevant to Radioactive Waste Disposal; Systematic Study of Metal Ion Sorption on Selected Geologic Media; Chromatographic K/sub d/ values of Radionuclides; Effects of Redox Potentials on Sorption of Radionuclides by Geologic Media; and Transport Properties of Nuclear Waste in Geologic Media. Individual papers were processed.

471 (PNL-SA--7352(Vol.1), pp 97-185) THEORETICAL AND EXPERIMENTAL EVALUATION OF WASTE TRANSPORT IN SELECTED ROCKS. Silva, R.J.; Benson, L.V.; Apps, J.A. (Univ. of California, Berkeley). 1978. Dep. NTIS, PC A22/MF A01.

From 2. contractor information meeting; Seattle, WA, USA (1 Oct 1978).

The experimental results obtained in FY 1978 in the study of rock interaction with U, Np, Pu, Am, and Cm are discussed, as well as the progress in the development and application of mass transfer codes to simulate groundwater conditions in low permeability rocks. Basalt, granite, and shale were used in the study.

472 (PNL-SA--7352(Vol.1), pp 187-230) STUDIES OF RADIONUCLIDE AVAILABILITY AND MIGRATION AT THE NEVADA TEST SITE RELEVANT TO RADIOACTIVE WASTE DISPOSAL. Tewhey, J.D.; Weed, H.C.; Coles, D.G.; Ramspott, L.D.; MacLean, S.C. (Univ. of California Livermore). 1978. Dep. NTIS, PC A22/MF A01.

From 2. contractor information meeting; Seattle, WA, USA (1 Oct 1978).

The sorption distribution coefficient, K/sub d/, of a nuclide is defined as the concentration of a nuclide on a solid phase divided by its concentration in the liquid phase subsequent to equilibrium being attained. In the case of batch K/sub d/ measurements done in the laboratory, it is difficult to ascertain the extent of equilibration, and experiments may not reflect true K/sub d/ values. Kurt Wolfsberg of Los Alamos Scientific Laboratory has referred to laboratory K/sub d/ measurements as distribution ratios or R/sub d/'s. An R/sub d/ is obtained in the same manner as a K/sub d/ but does not imply equilibrium. The term R/sub d/ is used in this study. The results from the batch R/sub d/ studies are listed in tabular form. The standard deviation of the counting data for each R/sub d/ is given. The R/sub d/ values can be arranged according to the following inequalities: Pu > Cs, except for biotite; Cs > Sr, except for limestone; and Sr > Tc. The exceptions are considered significant for Pu vs Cs on biotite, but not for Cs vs Sr on limestone. The low R/sub d/ values for Tc indicate that within the limits of measurement it is not sorbed under the experimental conditions in this study. One possible cause for the negative values is preferential sorption of the sample on the walls of the sample container, so that the sample container sorbs less Tc than the blank container does.

473 (PNL-SA--7352(Vol.1), pp 405-508) TRANSPORT PROPERTIES OF NUCLEAR WASTES IN GEOLOGIC MEDIA. ANNUAL REPORT, OCTOBER 1, 1977-SEPTEMBER 30, 1978. Seitz, M.G.; Rickert, P.G.; Fried, S.M.; Friedman, A.M.; Steindler, M.J. (Argonne National Lab., IL). 1978. Dep. NTIS, PC A22/MF A01.

From 2. contractor information meeting; Seattle, WA, USA (1 Oct 1978).

Laboratory experiments were performed with nuclides of strontium, tin, cesium, plutonium, and americium to determine their migration characteristics in rocks from formations that may be suitable for geologic repositories for nuclear waste. Infiltration experiments have been performed at high pressures and at different solution flows, using columns of massive porous rock, porous beds of granulated minerals, or impermeable rock having fissures. Experimental results obtained this year confirm results obtained in the previous year - that nuclides migrating by fluid flow in rock often exhibit complex behavior (bleeding, dispersion of nuclide velocities, etc.) not predicted by simple chromatographic-type models. Experimental results this year, obtained using two radically different geologic media (mineral sand and rock fissures), were in agreement with a migration model which considers the kinetics of nuclides reacting with the solution and geologic media.

474 (PNL-SA--7352(Vol.1), pp 59-68) SOLUTION SPECIES OF ^{239}PU IN OXIDIZING ENVIRONMENTS: I. POLYMERIC PU(IV). Rai, D.; Serne, R.J. (Battelle Pacific Northwest Labs., Richland, WA). 1978. Dep. NTIS, PC A22/MF A01.

From 2. contractor information meeting; Seattle, WA, USA (1 Oct 1978).

The uncertainties regarding the presence of Pu(IV) polymer in environmental studies is a source of error in evaluation of adsorption and in estimation of Pu migration rates. Our data suggest that Pu(IV) polymer is similar to $Pu(OH)_4$ in its solubility and that the presence of the polymer may be predicted from pH and total Pu in solution.

475 (PNL-SA--7352(Vol.1), pp 69-95) SOLUTION SPECIES OF ^{239}PU [V] IN THE ENVIRONMENT. Rai, D.; Serne, R.J.; Swanson, J.L. (Battelle Pacific Northwest Labs., Richland, WA). 1978. Dep. NTIS, PC A22/MF A01.

From 2. contractor information meeting; Seattle, WA, USA (1 Oct 1978).

Information regarding the oxidation states of Pu in environmental samples is needed for estimating its migration through the geologic media. Thermodynamic data were used to develop stability fields for different Pu species. The data indicate that in the Eh-pH range of natural aqueous environments, the dominant species of Pu is likely to be Pu[V] in relatively oxidizing environments and Pu[III] in reducing environments. Because of the lack of methods of determining Pu[V] in environmental samples containing trace concentrations of Pu, Pu[V] has not been previously identified in these samples. Plutonium [VI] is generally assumed to be the dominant species in relatively oxidizing environments. However, a combination of solvent extraction and spectrophotometric techniques used in this study show that solutions (> $10^{-5.4}$ M Pu) in equilibrium with ^{239}Pu[IV] hydroxide contain Pu[V], which is in agreement with the thermodynamic predictions. Although this method could not be used conclusively with the remaining solutions (< $10^{-5.4}$ M Pu) contacting ^{239}Pu[IV] hydroxide and $^{239}PuO_2$, the solvent extraction and Eh-pH results are similar for all the samples suggesting the strong possibility that all samples contain Pu[V]. Thus the possibility, ignored in the past, that Pu[V] may be the dominant species in relatively oxidizing environments should be considered.

476 (PNL-SA--7352(Vol.2)) PROCEEDINGS OF THE TASK 4 WASTE ISOLATION SAFETY ASSESSMENT PROGRAM SECOND CONTRACTOR INFORMATION MEETING. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 1978. Contract EY-76-C-06-1830. 494p. (CONF-7810227--(Vol.2)). Dep. NTIS, PC A21/MF A01.

From 2. contractor information meeting; Seattle, WA, USA (1 Oct 1978).

Volume II contains the following papers: Laboratory Studies of Radionuclide Distributions between Selected Groundwaters and Geologic Media; Applicability of Microautoradiography to Sorption Studies; The Kinetics and Reversibility of Radionuclide Sorption Reactions with Rocks; Mobility and Sorption Processes of Radioactive Waste Materials in Subsurface Migration; Batch K/sub d/ Experiments with Common Minerals and Representative Groundwaters; Cesium and Strontium Migration in Unconsolidated Geologic Material; Statistical Investigation of the Mechanics Controlling Radionuclide Sorption; The ARDISC Model - A Computer Program to Calculate the Distribution of Trace Element Migration in Partially-Equilibrating Media; and Some Geochemical Aspects of the Canadian Nuclear Waste Disposal System. Individual papers were processed.

477 (PNL-SA--7352(Vol.2), pp 161-218) KINETICS AND REVERSIBILITY OF RADIONUCLIDE SORPTION REACTIONS WITH ROCKS. PROGRESS REPORT FOR FISCAL YEAR 1978. Barney, G.S.; Anderson, P.D. (Rockwell International, Richland, WA). 1978. Dep. NTIS, PC A21/MF A01.

From 2. contractor information meeting; Seattle, WA, USA (1 Oct 1978).

Sorption reactions of environmentally significant radionuclides with rocks from proposed waste repository sites were examined. The kinetics and reversibility of sorption were studied by measuring the distribution of radioactive tracers between synthetic groundwaters and crushed basalt, granite, and argillite in batch experiments. The initial rapid sorption reaction is followed by a slow reaction which results from changes in the composition of the rock surface and bulk solution. Equilibrium concentrations of radionuclides were not reached even after 91 days. Dissolution of some rock components is also incomplete after this time. A relatively large amount of silica is dissolved in groundwaters contacted with basalt due to dissolution of the glassy phase of the basalt sample. This silica appears to increase the solubility of plutonium and americium. Artificially weathering basalt greatly increases the sorption capacity for each of the radionuclides studied. Weathering of granite had no observable effect on sorption. However, weathered argillite sorbs cesium, plutonium, and americium more strongly than unweathered argillite. Several irreversible sorption reactions were identified. Cesium is irreversibly sorbed on granite and ruthenium is irreversibly sorbed on basalt, granite, and argillite. Strontium and neptunium sorption are reversible for each rock type.

478 (PNL-SA--7352(Vol.2), pp 219-258) MOBILITY AND SORPTION PROCESSES OF RADIOACTIVE WASTE MATERIALS IN SUBSURFACE MIGRATION. Eichholz, G.G.; Wahlig, B.G. (Georgia Inst. of Tech., Atlanta). 1978. Dep. NTIS, PC A21/MF A01.

From 2. contractor information meeting; Seattle, WA, USA (1 Oct 1978).

The present work has been directed to the study of two related phenomena: the movement of submicron particles of a type expected near disposal sites through aquifers of appropriate characteristics, and the sorption processes associated with the uptake of waste ions in trace concentrations on submicron particles and the measurement of subsequent partition coefficients leading to competitive transfer from the particles to surrounding media. Because of the late date of finalization of the research contract and, hence, the limited manpower devoted to this project until recently, the results obtained so far are largely preliminary in nature. Work done has included some of the following: setting up of two types of columns with detector systems; determination of sensitivity and spatial resolution of the collimated G* and scintillation detectors; preparation of suspensions of kaolin particles; measurement of particulate movements through the bed with neutron-activated particles; determination of the Pu-237 γ-ray spectrum; and planning of the next phases of the work, using several columns and a larger research group. The results obtained so far illustrate the kind of differential movement between soluble tracers and active particulates that takes place in a permeable rock medium. It is hoped to obtain more definite results on the parameters concerned in the near future.

479 (PNL-SA--7352(Vol.2), pp 259-332) BATCH KD DETERMINATIONS WITH COMMON MINERALS AND REPRESENTATIVE GROUNDWATERS. Relyea, J.F.; Ames, L.L.; Serne, R.J.; Fulton, R.W.; Washburne, C.D. (Battelle Pacific Northwest Labs., Richland, WA). 1978. Dep. NTIS, PC A21/MF A01.

From 2. contractor information meeting; Seattle, WA, USA (1 Oct 1978).

Adsorption and desorption Kd's have been determined for seven elements (Pu, Am, I, Tc, Cs, Np, Sr) in tracer concentrations using four simulated groundwaters and twelve minerals (illite, kaolinite, montmorillonite, vermiculite, biotite, quartz, albite, anorthite, microcline, hornblende, enstatite, augite) in all their 336 combinations. System parameters have been measured in an attempt to characterize each system in as complete a manner as possible. The mineralogy and chemical composition of each mineral used have been determined (both before and after crushing bulk samples). Eh and pH have been measured for each system. The mineral-solution effluent is presently being analyzed using effluent from experiments without tracers. The data generated will be analyzed by Adaptronics to produce empirical equations for Kd's which will be used in models to predict nuclide migration in geologic media. Multiple oxidation state elements have been found to behave differently than single oxidation state elements in certain situations. Research beyond the scope of this study is required to determine the cause of this anomalous behavior (specifically: Tc, Np and Pu). To obtain an understanding of adsorption phenomenon under reducing conditions in geologic media, the batch Kd determinations just described will be carried out under anoxic and reducing conditions with the same nuclides, minerals and solutions.

480 (PNL-SA--7352(Vol.2), pp 333-425)

STATISTICAL INVESTIGATION OF THE MECHANICS CONTROLLING RADIONUCLIDE SORPTION. PART II. FINAL TECHNICAL REPORT. Mucciardi, A.N.; Booker, I.J.; Orr, E.C.; Cleveland, D. (Adaptronics, Inc., McLean, VA). 1978. Dep. NTIS, PC A21/MF A01.

From 2. contractor information meeting; Seattle, WA, USA (1 Oct 1978).

The results in this report demonstrate the feasibility of modeling empirically the transport of radionuclides (sorption coefficient, K/sub d/) in several geologic media/groundwater solution environments. Nonlinear Adaptive Learning Network models were synthesized to predict adsorption and desorption K/sub d/'s for seven radionuclides as a function of 16 minerals and four groundwaters. The major results reported here are comparisons between the actual K/sub d/'s and those predicted by the models, and a rank-ordering of the importance of the geologic medium/solution variables per nuclide K/sub d/ model.

481 (PNL-SA--7352(Vol.2), pp 429-435) ARDISC MODEL: A COMPUTER PROGRAM TO CALCULATE DISTRIBUTION OF TRACE ELEMENT MIGRATION IN PARTIALLY-EQUILIBRATING MEDIA. Strickert, R. (Battelle Pacific Northwest Labs., Richland, WA). 1978. Dep. NTIS, PC A21/MF A01.

From 2. contractor information meeting; Seattle, WA, USA (1 Oct 1978).

The computer model used is similar to the standard mixing model, in which a one-dimensional box is divided into zones. Each zone can sorb a solute such that, at equilibrium, the distribution of the species between the surface of the zone and the volume of the solution is α/β, where α = the relative amount of species on the surface, and $\beta = 1 - \alpha$. Following this distribution, the solution from one zone is transferred to the next zone and another equilibrium distribution cycle can occur, and so on for additional cycles and zones. To incorporate a kinetic effect into this migrational model, the α/β distribution was modified.

482 (PNL-SA--7352(Vol.2), pp 437-445) SOME GEOCHEMICAL ASPECTS OF THE CANADIAN NUCLEAR WASTE DISPOSAL PROGRAM. Vandergraaf, T. (Atomic Energy of Canada, Pinawa, Manitoba). 1978. Dep. NTIS, PC A21/MF A01.

From 2. contractor information meeting; Seattle, WA, USA (1 Oct 1978).

The Canadian disposal program concept calls for the disposal of radioactive wastes in a deep underground repository in an igneous intrusion (pluton) in the Canadian Precambrian Shield. The experiment program involves exploratory drilling and radionuclides adsorption studies. The main interest is in measuring surface adsorption coefficients for 20 radioactive elements on force rock types as a function of time, temperature, groundwater composition, and aging of the rock.

483 (PNL-SA--7897) MATHEMATICAL SIMULATION FOR SAFETY ASSESSMENT OF NUCLEAR WASTE REPOSITORIES. Brandstetter, A.; Raymond, J.R.; Benson, G.L. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 1979. Contract EY-76-C-06-1830. 24p. (CONF-790994--1). Dep. NTIS, PC A02/MF A01.

From 26. Pacific Northwest regional meeting of the American Geophysical Union; Bend, OR, USA (17 Sep 1979).

Mathematical models are being developed as part of the Waste Isolation Safety Assessment Program (WISAP) for assessing the post-closure safety of nuclear waste storage in geologic formations. The objective of this program is to develop the methods and data necessary to determine potential events that might disrupt the integrity of a waste repository and provide pathways for radionuclides to reach the bioshpere, primarily through groundwater transport. Four categories of mathematical models are being developed to assist in the analysis of potential release scenarios and consequences: (1) release scenario analysis models; (2) groundwater flow models; (3) contaminant transport models; and (4) radiation dose models. The development of the release scenario models is in a preliminary stage; the last three categories of models are fully operational. The release scenario models determine the bounds of potential future hydrogeologic changes, including potentially disruptive events. The groundwater flow and contaminant transport models compute the flowpaths, travel times, and concentrations of radionuclides that might migrate from a repository in the event of a breach and potentially reach the biosphere. The dose models compute the radiation doses to future populations. Reference site analyses are in progress to test the models for application to different geologies, including salt domes, bedded salt, and basalt.

484 (PNL-SA--7917) VITRIFICATION OF HIGH LEVEL WASTES: A REVIEW OF THE COMPUTER THERMAL ANALYSES FOR STORAGE CANISTERS. Wescott, R.I.; Slate, S.C. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Aug 1979. Contract AC06-76RL01830. 36p. (CONF-790822--17). NTIS, PC A03/MF A01.

From 87. AICHE national meeting; Boston, MA, USA (19 Aug 1979).

CANIST, a two-dimensional (r and THETA) computer program that solves the unsteady-state, heat conduction equation was used to model the thermal behavior of canisters filled with waste glass. CANIST has been found to be a valuable analytical tool for predicting the temperature profile of a waste storage canister as a function of several variables, including the diameter of the canister, the placement of internal fins, the heat generation rate of the waste glass, and the thermophysical properties of the canister and the waste glass. Thus, temperature dependent processes that may affect the integrity of the glass/canister unit, for example cracking, can be investigated using an analytical approach. In the present study, the canister temperature profiles predicted by CANIST were compared to canister temperatures measured during full-scale non-radioactive waste immobilization tests conducted at Pacific Northwest Laboratory. The agreement between experimental and predicted temperatures was good, particularly considering the fact that the thermophysical properties of the waste glass modeled have not yet been accurately determined. Examination of some glass-filled canisters has revealed cracking to have occurred in the glass. However, the comparison between measured and CANIST predicted temperatures suggests that cracking does not significantly influence the heat-transfer process. CANIST was also used to evaluate different ways of reducing the centerline temperature of a canister, and to predict the centerline temperature as a function of the heat generation rate of the waste glass and the type of interim storage, i.e., air or water.

485 (PNL-SA--7953) NATURE OF GLASS LEACHING. McVay, G.L.; Buckwalter, C.Q. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jul 1979. Contract EY-76-C-06-1830. 40p. (CONF-7909102--2). Dep. NTIS, PC A03/

MF A01.

From Argonne specialists' workshop on basic research needs for nuclear waste management; Argonne, IL, USA (5 Sep 1979).

Several models have been proposed in the literature to describe glass leaching behavior in aqueous solutions, but they are primarily applicable to relatively simple glasses. Their shortcoming is that they do not incorporate enough of the phenomena occurring during leaching. Leaching characteristics of a complex waste containment glass are discussed. Gamma irradiation increases the leach rates. Since bulk diffusion does not play a major role in leaching, the existing models are totally indequate for complex glasses. 4 figures, 3 tables. (DLC)

486 (PNL-SA--7992) PROCEEDINGS OF THE TASK 2 WORKSHOP WASTE ISOLATION SAFETY ASSESSMENT PROGRAM. Bradley, D.J. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 1979. Contract EY-76-C-06-1830. 234p. Dep. NTIS, PC A11/MF A01.

The reports from the workshop on waste form release rate analysis are presented. The workshop started with overview presentations on the Office of Nuclear Waste Isolation (ONWI), the Waste Isolation Safety Assessment Program (WISAP), WISAP Task 2 (Waste Form Release Rate Analysis), and WISAP Task 4 (Sorption/Desorption Analysis). Technical presentations followed in these areas: leaching studies on spent fuels, leaching studies on high-level waste glass, waste form surface science experiments, radiation effects, and leach modeling. Separate abstracts were prepared for each.

487 (PNL-SA--7992, pp 1-10) OVERVIEW OF THE OFFICE OF NUCLEAR WASTE ISOLATION. Kircher, J.F. (Office of Nuclear Waste Isolation, Columbus, OH). 1979. Dep. NTIS, PC A11/MF A01.

Proceedings of the Task 2 workshop Waste Isolation Safety Assessment Program.

The role of the Office of Nuclear Waste Isolation (ONWI) is discussed. The 2 main goals are to identify and characterize geological sites for use as nuclear waste depositories and to manage the development and application of specific supporting data and base technologies. The specific means of carrying out these goals and an organizational chart are included.

488 (PNL-SA--7992, pp 11-19) WASTE ISOLATION SAFETY ASSESSMENT PROGRAM. Brandstetter, A. (Battelle Pacific Northwest Lab., Richland, WA). 1979. Dep. NTIS, PC A11/MF A01.

Proceedings of the Task 2 workshop Waste Isolation Safety Assessment Program.

Associated with commercial nuclear power production in the United States is the generation of potentially hazardous radioactive wastes. The Department of Energy (DOE), through the National Waste Terminal Storage (NWTS) Program, is seeking to develop nuclear waste isolation systems in geologic formations that will preclude contact with the biosphere of waste radionuclides in concentrations which are sufficient to cause deleterious impact on humans or their environments. Comprehensive analyses of specific isolation systems are needed to assess the expectations of meeting that objective. The Waste Isolation Safety Assessment Program (WISAP) has been established at the Pacific Northwest Laboratory (operated by Battelle Memorial Institute) for developing the capability of making those analyses. The objectives of the Waste Isolation Safety Assessment Program (WISAP) are to: (1) develop the capabilities needed to assess the post-closure safety of geologic repositories, (2) obtain scientifically defensible generic and site-specific data necessary for safety assessments, (3) provide, as needed, studies to further support these data and analyses, (4) demonstrate the assessment capabilities by performing analyses of reference sites, (5) apply the assessment methodology to assist the National Waste Terminal Storage Program in site selection, and (6) perform repository site analyses responsive to the time schedule and to the level of sophistication required to meet the licensing needs of the National Waste Terminal Storage Program. Release scenario analysis, Task 1, is discussed.

489 (PNL-SA--7992, pp 21-31) WASTE ISOLATION SAFETY ASSESSMENT PROGRAM. TASK 2: WASTE FORM RELEASE RATE ANALYSIS. Bradley, D.J. (Battelle Pacific Northwest Lab., Richland, WA). 1979. Dep. NTIS, PC A11/MF A01.

Proceedings of the Task 2 workshop Waste Isolation Safety Assessment Program.

The purpose of this task is to measure and understand radionuclide release rates for waste forms anticipated for geologic isolation. At the present time, these waste forms include spent fuel, High-Level waste (HLW) glass, and for TRU, concrete, bitumen, urea-formaldehyde, and polymers. Radionuclide release processes are of major importance to breach consequence analyses because slower leach rates delay the transport of radionuclides from a breached repository to the biosphere. These processes are also responsible for determining the source term for migration away from the repository.

490 (PNL-SA--7992, pp 33-43) WISAP TASK 4 (SORPTION-DESORPTION ANALYSIS). Serne, R.J. (Battelle Pacific Northwest Lab., Richland, WA). 1979. Dep. NTIS, PC A11/MF A01.

Proceedings of the Task 2 workshop Waste Isolation Safety Assessment Program.

The overall objective of Task 4 is to provide the WISAP Task 3, release consequence analysis modelers the necessary information on long-term migration potential of leached nuclear wastes. The data being generated on nuclide sorption-desorption and precipitation-dissolution is obtained by both theoretical and experimental studies. At present only rock-nucilde interactions in the far-field (away from the waste induced heat and radiation fields) are under the preview of WISAP Task 4. The transport rate of nuclides is quantified by measurement of a paremeter called Kd, the distribution coefficient. Kd is defind as the ratio of the amount of nuclide adsorbed on the solid media (rock) divided by the amount of nuclide remaining in the contacting solution (groundwater). This distribution, for each nuclide, can be dependent on the rock type, groundwater composition, nuclide concentration, nuclide valence state, and rock-nuclide contact time and surface area contacted. Presently the thrust of the Task 4 program is generic in nature. Until more direction and actual samples are available, work is being performed to gather data on nuclide-rock interactions for a wide range of rocks and groundwater types.

491 (PNL-SA--7992, pp 45-66) NON-FLOW RELEASE STUDIES. Bradley, D. (Battelle Pacific Northwest Lab., Richland, WA). 1979. Dep. NTIS, PC A11/MF A01.

Proceedings of the Task 2 workshop Waste Isolation Safety Assessment Program.

Associated with commercial nuclear power production in the United States is the generation of potentially hazardous radioactive wastes. The Department of Energy (DOE), through the National Waste Terminal Storage (NWTS) Program, is seeking to develop nuclear waste isolation systems in geologic formations that will preclude contact with the biosphere of waste radionuclides in concentrations which are sufficient to cause deleterious impact on humans or their environments. Comprehensive analyses of specific isolation systems are needed to assess the expectations of meeting that objective. The Waste Isolation Safety Assessment Program (WISAP) has been established at the Pacific Northwest Laboratory (operated by Battelle Memorial Institute) for developing the capability of making those analyses. The subtask with this title encompasses all the tests that are not run above 100°C, which falls into the Autoclave release subtask. Currently we are conducting IAEA and static leach tests in parallel on both spent fuel and cold and actinide doped waste glasses.

492 (PNL-SA--7992, pp 67-71) AUTOCLAVE RELEASE STUDIES. Westsik, J.H. Jr. (Battelle Pacific Northwest Lab., Richland, WA). 1979. Dep. NTIS, PC A11/MF A01.

Proceedings of the Task 2 workshop Waste Isolation Safety Assessment Program.

Associated with commercial nuclear power production in the United States is the generation of potentially hazardous radioactive wastes. The Department of Energy (DOE), through the National Waste Terminal Storage (NWTS) Program, is seeking to develop nuclear waste isolation systems in geologic formations that will preclude contact with the biosphere of waste radionuclides in concentrations which are sufficient to cause deleterious impact on humans or their environments. Comprehensive analyses of specific isolation systems are needed to assess the expectations of meeting that objective. The Waste Isolation Safety Assessment Program (WISAP) has been established at the Pacific Northwest Laboratory (operated by Battelle Memorial Institute) for developing the capability of making those analyses. Among the analyses required for isolation system evaluation is the detailed assessment of the post-closure performance of nuclear waste repositories in geologic formations. The objective of this subtask is to examine the behavior of a high-level waste glass and spent fuel in aqueous solutions at high temperatures and pressures. By studying the waste water interactions we will (1) generate leach rate data, (2) gain some understanding of the kinetics and mechanisms of the interactions, (3) simulate long time, low temperature leaching, and (4) provide input to the leach model subtask.

493 (PNL-SA--7992, pp 73-81) RELEASE MODEL DEVELOPMENT. Turcotte, R.P. (Battelle Pacific Northwest Lab., Richland, WA). 1979. Dep. NTIS, PC A11/MF A01.

Proceedings of the Task 2 workshop Waste Isolation Safety Assessment Program.

Associated with commercial nuclear power production in the United States is the generation of potentially hazardous radioactive wastes. The Department of Energy (DOE), through the National Waste Terminal Storage (NWTS) Program, is seeking to develop nuclear waste isolation systems in geologic formations that will preclude contact with the biosphere of waste radionuclides in concentrations which are sufficient to cause deleterious impact on humans or their environments. Comprehensive analyses of specific isolation systems are needed to assess the expectations of meeting that objective. The Waste Isolation Safety Assessment Program (WISAP) has been established at the Pacific Northwest Laboratory (operated by Battelle Memorial Institute) for developing the capability of making those analyses. Among the analyses required for isolation system evaluation is the detailed assessment of the post-closure performance of nuclear waste repositories in geologic formations. The objective of this subtask is to develop a release model. Although leaching studies of nuclear waste glasses and other solid forms have been part of every development effort, we are not yet at a point where a defensible leach model can be presented. A major result of our work during the last year has been a clearer recognition that leaching models proposed previously, based on studies of simple glasses, are simply not applicable to typical nuclear waste glasses.

494 (PNL-SA--7992, pp 93-111) LEACHING CHARACTERISTICS OF ACTINIDES FROM SIMULATED REACTOR WASTE GLASS. Weed, H.C. (Lawrence Livermore Lab., CA); Coles, D.G.; Bradley, D.J.; Mensing, R.W.; Schweiger, J.S.; Rego, J.H. 1979. Dep. NTIS, PC A11/MF A01.

Proceedings of the Task 2 workshop Waste Isolation Safety Assessment Program.

This investigation is conducted for the Office of Nuclear Waste Isolation through Task 2 of the PNL Waste Isolation Safety Assessment Program (WISAP). One of the important goals of this program is to be able to calculate the rate of radionuclide release from candidate waste forms for geologic repository disposal. The present study has the following objectives in support of that goal: (1) to provide radionuclide source term information for use in release consequence analysis and migration rate calculations - this includes the effect of leaching, solution composition, flow rate, temperature, and time on the leach rate of simulated high level reactor waste glass; (2) to compare results obtained from the dynamic one-pass leaching method at LLL with those from the IAEA method at PNL. This progress report includes results to 120 days. Additional reports will be issued as the results are analyzed statistically, and a final report will be issued with PNL after the end of the experiments.

495 (PNL-SA--7992, pp 113-146) LEACHING PROPERTIES OF SOLIDIFIED TRU WASTE FORMS (WBS ELEMENT 0206). Colombo, P.; Neilson, R.M. Jr. (Brookhaven National Lab., Upton, NY). 1979. Dep. NTIS, PC A11/MF A01.

Proceedings of the Task 2 workshop Waste Isolation Safety Assessment Program.

Approximately 12 x 10^6 ft^3 of defense TRU wastes are in storage or burial at various DOE installations, principally at the Idaho National Engineering Laboratory (INEL), Hanford, and the Savannah River Plant. (Continued waste generation rates of 2 x 10^5 ft^3/yr are estimated.) These wastes will ultimately be moved and placed into a geologic repository for disposal. While there is some question as to whether existing TRU wastes will be disposed in their current form or first undergo some additiona volume reduction process, it is evident that volume reduction will be required for future TRU wastes. There are a number of volume reduction options available for combustible TRU wastes, the most viable option currently being some form of incineration. While incineration does result in a large volume reduction for combustible solid wastes, the ash that results is highly dispersible. As such, the immobilization of incinerator ash by solidification producing a solid monolithic waste form is desirable. Safety analysis of solidified TRU contaminated incinerator ash waste forms requires an estimate of the ability of these waste forms to retain activity in the disposal environment. The primary mechanism for the potential release of this activity is assumed to be by leaching. The experimental program conducted at Brookhaven National Laboratory will determine the leaching properties of TRU contaminated incinerator ash waste forms using hydraulic cement, urea-formaldehyde, bitumen, and vinyl ester-styrene as solidification agents. The data obtained will indicate the relative activity retention abilities of these waste forms and provide the basis for long-term release projections from full-scale waste forms.

496 (PNL-SA--7992, pp 159-166) LEACHING U-NP-PU-LOADED SIMULATED WASTE-GLASS. Diamond, H.; Friedman, A. (Argonne National Lab., IL). 1979. Dep. NTIS, PC A11/MF A01.
Proceedings of the Task 2 workshop Waste Isolation Safety Assessment Program.
Static leaching measurements were made in an oxygen-free environment. Three solutions, water, 0.3 M $NaHCO_3$ and brine were used; three intervals 2d, 8d, and 32d, and 23°C and 75°C temperatures were examined. The preliminary results show a varied response to temperature and some evidence that the leached material is not bound in siliceous fragments.

497 (PNL-SA--7992, pp 167-179) SPENT FUEL LEACH TESTS. Katayama, B. (Battelle Pacific Northwest Lab., Richland, WA). 1979. Dep. NTIS, PC A11/MF A01.
Proceedings of the Task 2 workshop Waste Isolation Safety Assessment Program.
This presentation is divided into two parts, pre-WISAP and WISAP. The pre-WISAP leach tests were started before WISAP sponsorship and do not give data directly applicable to the spent fuel release modeling studies being done in WISAP Task 2. However, the general leaching trends have suggested the general approach to some mechanistic studies. The WISAP portion of this presentation was started under WISAP sponsorship and is designed to fulfill the requirement of obtaining radionuclide release rates from spent fuel and understanding the radionuclide release process under simulated geologic storage conditions.

498 (PNL-SA--7992, pp 181-204) FUEL AND WASTE IMMOBILIZATION IN CANADA RELATED TO WISAP TASK 2. Strathdee, G.G. (Battelle Seattle Research Center, WA); Cameron, D.J.; Sargent, F.P.; Vandergraaf, T.T.; Johnson, L.H.; Shoesmith, D.W.; Taylor, P.; McIntyre, N.S.; Tait, J.C. 1979. Dep. NTIS, PC A11/MF A01.
Proceedings of the Task 2 workshop Waste Isolation Safety Assessment Program.
Durable glass samples which are leached for prolonged periods in standard tests behave in unexpected ways, particularly with respect to deposition of multi-valent cations at the glass alteration layer-solution interface. The standard leach tests normally used to provide comparisons of glass product durabilities may yield leach rate values which are conservative and high with respect to those within an actual repository environment. Depth profile studies indicate that addition of certain cations to waste glasses (i.e., those found to concentrate at the aqueous-glass interface and which promote mineralization of the alteration layer) may lead to in-situ generation of an additional barrier to radionuclide loss.

499 (PNL-SA--7992, pp 205-229) WIPP WASTE CHARACTERIZATION AND EXPERIMENTAL PROGRAMS. Molecke, M.A. (Sandia Labs., Albuquerque, NM). 1979. Dep. NTIS, PC A11/MF A01.
Proceedings of the Task 2 workshop Waste Isolation Safety Assessment Program.
The Waste Isolation Pilot Plant (WIPP) is intended to serve as a nuclear waste repository for those defense transuranic wastes, both contact-handled and remote-handled, which the Department of Energy (DOE) may designate as requiring deep geologic disposal. The WIPP will also provide a separate underground facility in which in-situ experiments with various waste forms, including high-level waste (HLW) and spent fuel, may be conducted. The DOE has also recommended (but not determined) that the WIPP be used to demonstrate the disposal of a limited number (no more than 1000) of spent fuel assemblies from commercial nuclear reactors. All the wastes, both TRU and spent fuel, placed into the WIPP for intended disposal would be retrievable for the periods required to demonstrate the validity of the disposal concept. These periods are not expected to exceed 10 years for TRU waste and 25 years for spent fuel. Wastes used in the experimental program will be removed at the conclusion of the experiments.

500 (PNL-SA--7992, pp 147-157) X-RAY PHOTOEMISSION SPECTROSCOPY (XPS) AND EXTENDED X-RAY ABSORPTION FINE STRUCTURE (EXAFS) STUDIES OF SILICATE BASED GLASSES. Karim, D.; Lam, D.J. (Argonne National Lab., IL). 1979. Dep. NTIS, PC A11/MF A01.
Proceedings of the Task 2 workshop Waste Isolation Safety Assessment Program.
The application of the x-ray photoemission spectroscopy (XPS) technique to study the electronic structure and bonding of heavy metal oxides in alkali- and alkali-earth-silicate glasses had been demonstrated. The bonding characteristics of the iron oxide and uranium oxide in sodium silicate glasses were deduced from the changes in the oxygen 1s levels and the heavy metal core levels. It is reasonable to expect that the effect of leaching on the heavy metal ions can be monitored using the appropriate core levels of these ions. To study the effect of leaching on the glass forming network, the valence band structure of the bridging and nonbridging oxygens in sodium silicate glasses were investigated. The measurement of extended x-ray absorption fine-structure (EXAFS) is a relatively new analytical technique for obtaining short range (<5 A) structural information around atoms of a selected species in both solid and fluid systems. Experiments have recently begun to establish the feasibility of using EXAFS to study the bonding of actinides in silicate glasses. Because of the ability of EXAFS to yield specific structural data even in complex multicomponent systems, it could prove to be an invaluable tool in understanding glass structure.

501 (PNL-SA--7992, pp 83-91) GAMMA-IRRADIATION EFFECTS ON LEACHING CHARACTERISTICS OF WASTE CONTAINMENT MATERIALS. McVay, G.L. (Battelle Pacific Northwest Lab., Richland, WA). 1979. Dep. NTIS, PC A11/MF A01.
Proceedings of the Task 2 workshop Waste Isolation Safety Assessment Program.
Materials which contain typical radioactive waste are subjected to gamma-irradiation from the waste itself. Therefore, it is important to understand the effect of this irradiation on the leaching characteristics of waste containing materials in the event of an

exposure to water during the time span when significant gamma-irradiation is occurring. With this goal in mind, an experiment has been initiated to characterize the leaching behavior of 7668 glass, supercalcine ceramic, and UO_2 during exposure to gamma-irradiation.

502 (PNL-SA--8011) RADIOLOGICAL ASPECTS OF SEA BED DUMPING IN THE DEEP OCEANS. Templeton, W.L. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 1979. Contract EY-76-C-06-1830. 19p. (CONF-791050--2). Dep. NTIS, PC A02/MF A01.

From 3. Nuclear Energy Agency seminar on marine radioecology; Tokyo, Japan (1 Oct 1979).

In order to control coastal discharges or ocean dumping of any kind of material, it is necessary to determine a release rate. This can only come from a knowledge of the composition and chemical form of the source materials, the distribution and bioavailability of these materials in the ocean ecosystem, the degree and rates of bioaccumulation and the actual or potential use of the ocean resources. With this information release rates within acceptable limits for man and the ecosystem can then be determined. Today, probably the only situations which apply this approach are the controlled disposal of radioactive wastes. In this paper a recent radiological assessment of the dumping of packaged radioactive wastes on the seabed is discussed and some environmental aspects of the United States Department of Energy program are described examining the feasibility of the emplacement of contained radioactive wastes within the deep ocean sediments.

503 (PNL-SA--8182) IRRADIATION EFFECTS ON BOROSILICATE WASTE GLASSES. Roberts, F.P. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jun 1980. Contract AC06-76RL01830. 16p. (CONF-800609--16). NTIS, PC A02/MF A01.

From 10. international symposium on effects of radiation on materials; Savannah, GA, USA (3 Jun 1980).

The effects of alpha decay on five borosilicate glasses containing simulated nuclear high-level waste oxides were studied. Irradiations carried out at room temperature were achieved by incorporating 1 to 8 wt % $^{244}Cm_2O_3$ in the glasses. Density changes and stored-energy build-up saturated at doses less than 2 x 10^{24} alpha decays/kg. Damage manifested by stored energy was completely annealed at 633°K. Positive and negative density changes were observed which never exceeded 1%. Irradiation had very little effect on mechanical strength or on chemical durability as measured by aqueous leach rates. Also, no effects were observed on the microstructure for vitreous waste glasses, although radiation-induced microcracking could be achieved on specimens that had been devitrified prior to irradiation.

504 (PNL-SA--8257, pp 4-27) HYDROGEOLOGIC FRAMEWORK OF THE PASCO BASIN. Crosby, J.W. III. Feb 1981. NTIS, PC A08/MF A01.

From WISAP release scenario analysis workshop; Seattle, WA, USA (13 Sep 1979).

Tholeiitic flood basalts of the Columbia River Basalt Group comprise the major rock sequence of the Pasco Basin. These basalts are known to be at least 10,665 ft thick at one place, but the total thickness is unknown. The basalts are commonly overlain by silts and sands of the Pliocene Ringold Formation. Glaciofluvial deposits referred to as the Pasco Gravels and Touchet Silts appear frequently at the surface. Other materials exposed at the surface include the Palouse loess and recent accumulations of dune sands, alluvium, and colluvium. The Columbia River Group has been subdivided into five formations and numerous members based upon petrography, magnetic polarity, and chemical composition. Adequate knowledge of the stratigraphy is a necessary prerequisite to understanding the ground-water characteristics. Unconfined ground waters occur in the glaciofluvial materials and the upper part of the Ringold Formation. Confined ground waters occur in the lower Ringold deposits and in the Columbia River Basalt Group. Disposal of liquid radioactive wastes in surficial materials is reflected in the configuration of the water table and movements of waters in the unconfined system. Ground waters in the confined basaltic environment move largely through the interflow contact zones and certain interbedded sediments. Hydraulic measurements in the basalts have led to contradictory findings about the relative distribution of hydraulic conductivity. For many years vertical movement of water in the basalts was thought to be severely restricted in most places. Some recent studies suggest that vertical conductivity may be much more important than was previously believed.

505 (PNL-SA--8257) ASSESSMENT OF EFFECTIVENESS OF GEOLOGIC ISOLATION SYSTEMS. WISAP RELEASE SCENARIO ANALYSIS WORKSHOP, SEPTEMBER 13-15, 1979. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Feb 1981. Contract AC06-76RL01830. 158p. (CONF-7909206--). NTIS, PC A08/MF A01.

From WISAP release scenario analysis workshop; Seattle, WA, USA (13 Sep 1979).

Individual papers were indexed for inclusion in the data base.

506 (PNL-SA--8257, pp 28-54) EXTENT OF PREVIOUS GLACIATIONS. Bull, C. Feb 1981. NTIS, PC A08/MF A01.

From WISAP release scenario analysis workshop; Seattle, WA, USA (13 Sep 1979).

Several features associated with glaciers are discussed in relation to the possible effects glaciation might have on a radioactive waste repository. The extent of past glaciatious is estimated. The effect of a glacier on the hydrological parameters of an area, the changes in climate associated with glaciers, and the erosion likely to accompany glaciation are discussed. (ACR)

507 (PNL-SA--8257, pp 55-89) EXPECTED ENVIRONMENTS AND CLIMATES IN THE NORTHWESTERN USA IN THE NEXT MILLION YEARS (THE WISAP CLIMATE SUBMODEL). Kukla, G.J. Feb 1981. NTIS, PC A08/MF A01.

From WISAP release scenario analysis workshop; Seattle, WA, USA (13 Sep 1979).

Probabilities of future climate-related stresses in northwestern USA in the coming million years were estimated as a contribution to a conceptual simulation model for release-scenario analysis of a hypothetical site in the Columbia Plateau. The estimates are based upon the reconstructed history of past climates and environments. The assumption is made that the shifts in climate experienced in the past will be repeated in the future, principally unaffected by man's technology. We focused on the extremes and on the time interval of the first 100,000 years. We expect that the potentially most serious climate-related stresses will consist of the effects of future alpine glaciers and of the failures of ice-dammed lakes on the ground-water regime and on the geologic structure of the repository's confinement. Probability of ice advances will increase significantly some 3000 to 5000 years from now. Major catastrophic floods caused by the failed ice jams are only marginally probable over the next 50 millennia but are highly probable thereafter. Annual air surface temperatures in the Columbia Plateau will drop in the future by up to approx. 10 to 15°C and rise at other times by up to approx. 5°C. Annual precipitation is estimated to change by a factor of 0.5 to about 3.0.

508 (PNL-SA--8257, pp 90-92) VOLCANIC HAZARD STUDIES. Crowe, B. Feb 1981. NTIS, PC A08/MF A01.

From WISAP release scenario analysis workshop; Seattle, WA, USA (13 Sep 1979).

Volcanic hazards studies for the WISAP program are concerned with evaluation of risk due to future volcanism, with respect to long-term isolation of radioactive waste through deep geologic storage. Three major areas of research have been examined: (1) regional distribution of Quaternary Volcanism with respect to repository siting; (2) the consequences or disruption effects due to penetration of a repository by volcanism; and (3) probability calculations of the likelihood of disruption of a repository by volcanism.

509 (PNL-SA--8257, pp 121-129) GEOCHEMISTRY AND THE RELEASE SCENARIO ANALYSIS TASK. Deutsch, W.J. Feb 1981. NTIS, PC A08/MF A01.

From WISAP release scenario analysis workshop; Seattle, WA, USA (13 Sep 1979).

The release scenario analysis of a waste repository must include possible changes in the geochemistry of the system because of the importance of the geochemistry in both the dissolution rate of the waste and in the interaction between radionuclide-laden ground water and host rock. The system will be studied by determining the limits of the effect that various natural perturbations will have on the geochemical parameters determined to be of importance.

510 (PNL-SA--8491) INCENTIVES FOR PARTITIONING, REVISITED. Cloninger, M.O. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 24 Mar 1980. Contract AC06-76RL01830. 30p. (CONF-800444--2). NTIS, PC A03/MF A01.

From 2. technical meeting on the nuclear transmutation of actinides; Ispra, Italy (21 Apr 1980).

The incentives for separating and eliminating various elements from radioactive waste prior to final geologic disposal were investigated. Exposure pathways to humans were defined, and potential radiation doses to an individual living within the region of influence of the underground storage site were calculated. The assumed radionuclide source was 1/5 of the accumulated high-level waste from the US nuclear power economy through the year 2000. The repository containing the waste was assumed to be located in a reference salt site geology. The study required numerous assumptions concerning the transport of radioactivity from the geologic storage site to man. The assumptions used maximized the estimated potential radiation doses, particularly in the case of the intrusion water well scenario, where hydrologic flow field dispersion affects were ignored. Thus, incentives for removing elements from the waste tended to be maximized. Incentives were also maximized by assuming that elements removed from the waste could be eliminated from the earth without risk. The results of the study indicate that for reasonable disposal conditions, incentives for partitioning any elements from the waste in order to minimize the risk to humans are marginal at best.

511 (PNL-SA--8571(Vol.1)) WASTE ISOLATION SAFETY ASSESSMENT PROGRAM. TASK 4. THIRD CONTRACTOR INFORMATION MEETING. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jun 1980. Contract AC06-76RL01830. 558p. (CONF-7910160--(Vol.1)). NTIS, PC A24/MF A01.

From 3. contractor information meeting on waste isolation safety assessment program task 4; Seattle, WA, USA (14 Oct 1979).

The Contractor Information Meeting (October 14 to 17, 1979) was part of the FY-1979 effort of Task 4 of the Waste Isolation Safety Assessment Program (WISAP): Sorption/Desorption Analysis. The objectives of this task are to: evaluate sorption/desorption measurement methods and develop a standardized measurement procedure; produce a generic data bank of nuclide-geologic interactions using a wide variety of geologic media and groundwaters; perform statistical analysis and synthesis of these data; perform validation studies to compare short-term laboratory studies to long-term in situ behavior; develop a fundamental understanding of sorption/desorption processes; produce x-ray and gamma-emitting isotopes suitable for the study of actinides at tracer concentrations; disseminate resulting information to the international technical community; and provide input data support for repository safety assessment. Conference participants included those subcontracted to WISAP Task 4, representatives and independent subcontractors to the Office of Nuclear Waste Isolation, representatives from other waste disposal programs, and experts in the area of waste/geologic media interaction. Since the meeting, WISAP has been divided into two programs: Assessment of Effectiveness of Geologic Isolation Systems (AEGIS) (modeling efforts) and Waste/Rock Interactions Technology (WRIT) (experimental work). The WRIT program encompasses the work conducted under Task 4. This report contains the information presented at the Task 4, Third Contractor Information Meeting. Technical Reports from the subcontractors, as well as Pacific Northwest Laboratory (PNL), are provided along with transcripts of the question-and-answer sessions. The agenda and abstracts of the presentations are also included. Appendix A is a list of the participants. Appendix B gives an overview of the WRIT program and details the WRIT work breakdown structure for 1980.

512 (PNL-SA--8604) ENVIRONMENTAL EFFECTS OF REACTOR WASTE DISPOSAL ALTERNATIVES. Unruh, C.M. (Pacific Northwest Lab., Richland, WA (USA)). May 1980. Contract AC06-76RL01830. 37p. (CONF-8006214--1). NTIS, PC A03/MF A01. Order Number DE82005911.

From Summer Faculty Institute Center for Educational Affairs conference; Argonne, IL, USA (16 Jun 1980).

This present document, Environmental Impact Statement on Management of Commercially Generated Radioactive Waste, describes ten alternative methods for disposal of nuclear wastes and evaluates their anticipated environmental impacts. The ten alternatives are: (1) geologic disposal using conventional mining techniques; (2) chemical resynthesis; (3) very deep hole concept; (4) rock melting concept; (5) island disposal; (6) sub-seabed geologic disposal; (7) ice sheet disposal; (8) reverse-well disposal; (9) partitioning and transmutation; and (10) space disposal. In evaluating the various technical strategies, issues and environmental impacts have been analyzed as best understood currently. Based on the analysis presented here, and in the light of the greater depth of knowledge on geologic disposal, DOE proposes that: (1) the disposal of radioactive wastes in geologic formations can likely be developed and applied with minimal environmental consequences; and (2) therefore the program emphasis should be on the establishment of mined repositories as the operative disposal technology.

513 (PNL-SA--8672) EVALUATION OF CERAMIC AND POLYMERIC MATERIALS FOR USE IN ENGINEERED BARRIER SYSTEMS. Fullam, H.T.; Skiens, W.E. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 1980. Contract AC06-76RL01830. 14p. (CONF-801124--44). NTIS, PC A02/MF A01.

From 3. annual meeting of the Materials Research Society; Boston, MA, USA (17 Nov 1980).

Ceramic materials evaluated in the screening studies were Al_2O_3 (99.8%), mullite, vitreous silica, $BaTiO_3$, $CaTiO_3$, $CaZrO_3$, $CaTiSiO_5$, TiO_2, $ZrSiO_4$, basalt, Pyroceram 9617, and Marcor code 9658 machinable glass ceramic. One grade of graphite (Toyotanso IG-11) was also evaluated. Demineralized water, a synthetic Hanford groundwater, and a synthetic NaCl brine solution were used in the screening tests. Demineralized water was used in all five of the leach tests, but the other solutions were only used in the static leach tests at 150 and 250°C. Based on the results obtained, graphite appears to be the most leach resistant of the materials tested with the two grades of alumina being the best of the ceramic materials. Titanium dioxide and ZrO_2 are the most leach resistant of the remaining materials. Candidate materials from all three general classes of polymers (thermoplastics, thermosets, and elastomers) were considered in the selection of materials. Selected groups of polymers were tested in the flowing autoclave at 150, 200, and 250°C with some polymers being further tested at the next higher temperature. Next, selected samples were exposed to gamma radiation. These samples were then submitted for tensile and elongation measurements. Selected samples which appeared promising from both autoclave and radiation testing were further evaluated by impact tests. The materials that appeared most promising after autoclave testing were the EPDM rubbers, polyphenylene sulfide, poly(ethylene-tetrafluoroethylene) copolymer, and polyfurfuryl alcohol. The radiation dose had little effect on polyfurfuryl alcohol and polyphenylene sulfide samples; very significant decreases in elongation were observed for the fluorocarbon copolymer and the EPDM rubbers. While the polyphenylene sulfide and polyfurfuryl alcohol showed little change in impact strength, poly(ethylene-tetrafluoroethylene) decreased in impact strength.

514 (PNL-SA--8712) STANDARD LEACH TESTS FOR NUCLEAR WASTE MATERIALS. Strachan, D.M.; Barnes, B.O.; Turcotte, R.P. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 1980. Contract AC06-76RL01830. 9p. (CONF-801124--46). NTIS, PC A02/MF A01.

From 3. annual meeting of the Materials Research Society; Boston, MA, USA (17 Nov 1980).

Five leach tests were conducted to study time-dependent leaching of waste forms (glass). The first four tests include temperature as a variable and the use of three standard leachants. Three of the tests are static and two are dynamic (flow). This paper discusses the waste-form leach tests and presents some representative data. 4 figures. (DLC)

515 (PNL-SA--8744) PROBABLE LEACHING MECHANISMS FOR UO_2 AND SPENT FUEL. Wang, R.; Katayama, Y.B. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 1980. Contract AC06-76RL01830. 8p. (CONF-801124--42). NTIS, PC A02/MF A01.

From 3. annual meeting of the Materials Research Society; Boston, MA, USA (17 Nov 1980).

The oxidation and dissolution mechanisms for UO_2 and spent fuel will be quite similar based on this preliminary work with electrochemical leaching of UO_2 and spent fuel. In solutions containing oxygen or other oxidizing species, the UO_2 surface will be rapidly oxidized and dissolved following the transformation of uranium from U(IV) to U(VI). The hydrolysis of dissolved uranyl ions forms solid UO_3 hydrates or related complex compounds deposited onto the UO_2 surface, or other surfaces, as thin or thick coatings. Depending on the pH, temperature, and time, the various kinds of porosity and the mechanical properties of the hydrate coatings will control the dissolution rate. The effects of radiation, in terms of generation of H_2O_2, will enhance the dissolution kinetics. Electrochemical methods may be useful for determining the surface conditions, dissolution rate, and accelerated dissolution behavior for NO_2 and spent fuel. Electrochemial methods can rapidly generate much information in terms of dissolution rate and surface film properties - such as thickness, porosity, and oxidation state - in-situ during the leaching process.

516 (PNL-SA--8840) DUMPING OF LOW-LEVEL RADIOACTIVE WASTE IN THE DEEP OCEAN. Templeton, W.L. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 1980. Contract AC06-76RL01830. 19p. (CONF-801063--4). NTIS, PC A02/MF A01.

From International symposium on the impacts of radionuclide releases into the marine environment; Vienna, Austria (6 Oct 1980).

Two international agreements relate to the dumping of packaged radioactive waste into the oceans - the Convention on the Prevention of Marine Pollution by Dumping Wastes and Other Matter of 1972 (London Convention) and the Multilateral Consultation and Surveillance Mechanism for Sea Dumping of Radioactive Waste of 1977 under the Organization for Economic Co-operation and Development (OECD). The International Atomic Energy Agency was given the responsibility to define high-level radioactive wastes which are unsuitable for dumping in the oceans and to make recommendations for the dumping of other radioactive wastes. A revised Definition and Recommendations was submitted and accepted by the London Convention. This paper reviews the technical basis for the Definition and describes how it has been applied to the radiological assessment of the only operational dumping site in the North East Atlantic.

517 (PNL-SA--8854) RADON CONTROL BY MULTILAYER EARTH BARRIERS. 1. MODELING OF MOISTURE AND DENSITY EFFECTS ON RADON DIFFUSION FROM URANIUM MILL TAILINGS. Nelson, R.W.; Gee, G.W.; Oster, C.A. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 1980. Contract AC06-76RL01830. 10p. (CONF-801155--5). NTIS, PC A02/MF A01.

From 3. annual symposium on uranium mill tailings management; Ft Collins, CO, USA (24 Nov 1980).

A new approach is proposed and used to analyze radon emissions from uranium mill tailings. The approach has been developed based upon the analysis of partially saturated water flow effects on diffusion of radon through the cover and tailings materials. Such an analysis provides the moisture distribution with depth and time in the profile, hence, the gas filled pore space available for radon diffusion is known as a function of both depth and time. Through using the diffusion coefficient for radon gas in air, appropriately reduced to account for the porous material at the particular gas filled pore space, the bulk diffusion coefficient, which in the past could only be obtained experimentally, is provided directly. Solution for the radon concentration in the engineered tailings-cover profile is then possible so the radon emission rate at the ground surface can be calculated directly. Results for evaluation of tailings and multilayered cover materials are presented using the new approach. Some of the engineered covers analyzed by the method were installed in a field trial at Grand Junction, Colorado, last summer. Initial measurements were made on the installed cover and generally good agreement is found between calculated and field measured emission rates. Continuing measurements are being made for longer term comparison with calculated results and to verify the effectiveness of thinner covers with use of engineered moisture retention for radon emissions control.

518 (PNL-SA--8861) IMPROVING LONG TERM RESISTANCE OF URANIUM MILL TAILINGS COVERS TO HYDROLOGIC STRESS USING RIPRAP: A PRELIMINARY ASSESSMENT. Walters, W.H.; Skaggs, R.L. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 1980. Contract AC06-76RL01830. 7p. (CONF-801155--1). NTIS, PC A02/MF A01.

From 3. annual symposium on uranium mill tailings management; Ft Collins, CO, USA (24 Nov 1980).

The long term effectiveness of riprap to protect uranium mill tailings repositories against erosion by flowing water is currently being evaluated by Pacific Northwest Laboratory for the US Nuclear Regulatory Commission. The investigation is part of a comprehensive study to determine technically justified and cost effective alternatives for armoring of radon suppression covers. As a protective cover, riprap appears to be one of the most reliable methods because of its proven durability to resist channel bed scour around bridge piers and erosion of river banks. Damage to radon suppression covers by water could occur under similar circumstances where the erosion would be due to overland flow or a large flood event. The principal mechanisms of soil erosion by water are discussed with respect to earth surfaces of mounted or flat geometries. Since long time periods on the order of one thousand years may be involved, any riprap design procedure should consider allowances for partial failures of the cover. Design procedures are summarized and both advantages and disadvantages are discussed. Problems that may occur over long term storage of mill tailings are evaluated in order to put the use of riprap in proper perspective.

519 (PNL-SA--8927) ASSESSMENT OF EFFECTIVENESS OF GEOLOGIC ISOLATION SYSTEMS. RADIONUCLIDE-TRANSPORT SENSITIVITY STUDIES. Cole, C.R. (Pacific Northwest Lab., Richland, WA (USA)). Sep 1980. Contract AC06-76RL01830. 12p. (CONF-801209--10). NTIS, PC A02/MF A01. Order Number DE82005872.

From 2. annual National Waste Terminal Storage (NWTS) information meeting; Columbus, OH, USA (9 Dec 1980).

This paper will summarize some of the sensitivity study results for the Paradox and the INFCE studies. Sensitivity studies are performed by determining the effect that a change in a single parameter has on system response. Many of the sensitivity studies performed for the Paradox were predictable with results varying in a linear manner with the parameter being changed. For the Paradox study (a two-aquifer system connected via a fracture through the bedded salt), sensitivity studies were performed to determine what effect fracture resistance has on predicted flow through the repository and the resultant water velocity in the transporting aquifer. Results indicated that interaquifer transfer could nearly double from the base case but was ultimately limited by the upper and lower aquifer characteristics. The important result, however, was that water velocity in the contaminant transporting aquifer was affected very little. While the increased flow through the repository could potentially dissolve the waste over a shorter time frame, one of the main isolation parameter (travel time) was affected very little. The above sensitivity results for variations in typical hydrologic parameters, while interesting, are a priori predictable. The more important sensitivity study results relate to travel time, variation in travel time, and the combined effects of travel time variation and contaminant retardation factors as well as the insight these studies have provided in helping us better understand the complex transport process for radionuclides and radionuclide chains.

520 (PNL-SA--8939) EVALUATION OF METALLIC MATERIALS FOR USE IN ENGINEERING BARRIER SYSTEMS. Pitman, S.G.; Griggs, B.; Elmore, R.P. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 1980. Contract AC06-76RL01830. 8p. (CONF-801124--41). NTIS, PC A02/MF A01.

From 3. annual meeting of the Materials Research Society; Boston, MA, USA (17 Nov 1980).

Conclusions of this work are as follows: Inconel, Incoloy, Hastelloy C-276, and titanium alloys all had excellent corrosion resistance in all postulated repository environments tested. Further work will be required to evaluate the pertinent enviro-mechanical properties of these materials; the mechanical properties of grade 2 titanium are better than those of grade 12 titanium, except the tensile and yield strengths. These properties include fatigue-crack-growth rate, environmental fatigue-crack-growth rate, fracture toughness, impact toughness, and dynamic fracture toughness; there is no evidence in the current data to indicate that the simulated repository environment is aggressive to grade 2 or grade 12 titanium. This includes data from corrosion-fatigue, crevice corrosion, wedge-loaded cracked specimens, and residual-stress specimens.

521 (PNL-SA--6972) SHORT DESCRIPTION OF THE AEGIS APPROACH. Dove, F.H. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Oct 1980. Contract AC06-76RL01830. 18p. (CONF-8010130--5). NTIS, PC A02/MF A01.

From Waste-Rock Interactions Technology annual information meeting; Seattle, WA, USA (13 Oct 1980).

The evaluation of the long-term

effectiveness of isolating nuclear wastes in geologic formation involves a number of distinct steps. Initially, the specific nature of the engineered components in the repository and of the existing geologic and hydrologic systems surrounding the repository must be adequately understood. Since natural geologic processes and future human activities may alter these systems over the long time frames necessary for isolation, an evaluation must also be made to determine if there are plausible scenarios for breaching the repository. If such a scenario is identified, then the transport of radionuclides from the repository to the evnrionment must be estimated. A final step can be added to estimate the effects of the radionuclides in the environment. The Assessment of Effectiveness of Geologic Isolation Systems (AEGIS) currently has the methods for performing these evaluation steps. The existing AEGIS approach is to obtain the necessary information which characterizes the geologic isolation system from the programs designing the engineered components and the programs making field measurements of the geology and hydrology. AEGIS then develops a conceptual description of the geologic systems and uses computer models to simulate the existing ground-water pathways. AEGIS also uses a team of consulting experts, with the assistance of a computer model of the geologic processes, to develop and evaluate plausible release scenarios. Then other AEGIS computer models are used to simulate the transport of radionuclides to the surface and the resultant radiation doses to individuals and populations. The purpose of this report is to describe this approach.

522 (PNL-SA--8975) NATURAL ANALOGUES FOR CRYSTALLINE RADIOACTIVE WASTE FORMS. PART II. NON-ACTINIDE PHASES. Haaker, R.F.; Ewing, R.C. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Oct 1980. Contract AC06-76RL01830. 9p. (CONF-801124--26). NTIS, PC A02/MF A01.

From 3. annual meeting of the Materials Research Society; Boston, MA, USA (17 Nov 1980).

The purpose of this paper is to summarize relevant geologic literature on mineral analogues to important non-actinide phases in synroc and supercalcine. Solid solution chemistry, occurrence, and alteration effects are emphasized. Stable and unstable mineral associations are identified. The report also summarizes the behavior of mineral analogues to the actinide phases in synroc and supercalcine.

523 (PNL-SA--9054) TIME AND TEMPERATURE DEPENDENCE OF THE LEACHING OF A SIMULATED HIGH-LEVEL WASTE GLASS. Westsik, J.H. Jr.; Peters, R.D. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 1980. Contract AC06-76RL01830. 20p. (CONF-801124--47). NTIS, PC A02/MF A01.

From 3. annual meeting of the Materials Research Society; Boston, MA, USA (17 Nov 1980).

This study investigated how time and temperature affect the leaching of 76-68 waste glass. Two temperature-dependent regions were identified in the results. Above about 250°C, hydrothermal alteration and releases of B, Na, and Mo occur faster than would be expected from the results of tests at lower temperatures. At 250°C and below, leaching appears to proceed through three steps. Characteristically, in the first step, the pH of the leachate changes with time; in the second, the pH is constant and normalized releases based on B, Mo, Na, and Si are described by the empirical equation: Release (g glass/m²) = kt/sup 0.67(range 0.58-0.75)/, where k follows an Arrhenius temperature dependence (apparent activation energy = 5.3 x 10⁴ J/mol °K); and in the third step, so far observed only at 250°C, normalized releases show a weaker time dependence than they do in the second step.

524 (PNL-SA--9077) RADIONUCLIDE TRANSPORT UNDER CHANGING SOIL SALINITY CONDITIONS. Jones, T.L.; Gee, G.W.; Wierenga, P.J. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Dec 1980. Contract AC06-76RL01830. 14p. (CONF-801217--4). NTIS, PC A02/MF A01.

From Symposium on modeling and low-level waste management; Denver, CO, USA (1 Dec 1980).

This paper presents experimental measurements and computer simulations of strontium-85 movement through laboratory soil columns. There are two primary objectives of the work. The first is to examine the enhanced leachability of strontium-85 when the soil salinity is raised. The second is to compare the ability of two solute transport models to describe this phenomena. The experiment consisted of injecting a pulse of what will be termed a low salt solution spiked with strontium-85 to a soil column. The column was then leached with over eighteen pore volumes of the low salt solution resulting in no leaching of the strontium-85 from the column. The column was then leached with a high salt solution causing the rapid release of the strontium-85 within five to six pore volumes. The results demonstrate that strontium-85 is subject to greater mobility when the soil salinity is increased. This confirms the necessity of evaluating the soil salinity very carefully in modeling efforts. The results also indicate that batch Kd values, combined with column Kd values and parameters obtained by curve fitting procedures do a fairly good job in describing the snowplow effect. The comparison of the two models reveal insignificant differences when describing the movement of tritium and strontium-85 under high salinity conditions. There was a significant improvement, however, when the mobile-immobile water equations were used to describe the combined low salt-high salt experiment.

525 (PNL-SA--9092) EFFECTIVE SUMMARY EVALUATORS FOR DEEP NUCLEAR WASTE REPOSITORIES: GEOHYDROLOGIC RESPONSE FUNCTION. Nelson, R.W.; Dove, F.H. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Mar 1981. Contract AC06-76RL01830. 28p. (CONF-810372--1). NTIS, PC A03/MF A01.

From Symposium on uncertainties associated with the regulation of the geologic disposal of high-level radioactive waste; Gatlinburg, TN, USA (9 Mar 1981).

Useful insight has been gained over the past four years as hydrologic system modeling has been applied to evaluate hypothetical, waste-repository sites in various geologic media. The Geohydrologic Response Functions, described in this paper, are shown to: blend extensive results of technical analysis into simple summary relationships, and to potentially help the public and decision makers to evaluate the magnitude of any loss in repository integrity.

526 (PNL-SA--9094) DEALING WITH REGIONAL HYDROLOGIC DATA-BASE LIMITATIONS. CASE EXAMPLE: THE COLUMBIA RIVER BASALTS. Schalla, R.; Leonhart, L.S. (Battelle Pacific Northwest Labs., Richland, WA (USA); Rockwell International Corp., Richland, WA (USA).

Rockwell Hanford Operations). 1981. Contract AC06-76RL01830. 15p. (CONF-810372--12). NTIS, PC A02/MF A01. Order Number DE81030086.

From Symposium on uncertainties associated with the regulation of the geologic disposal of high-level radioactive waste; Gatlinburg, TN, USA (9 Mar 1981).

Limitations are encountered in assembling hydrologic data for a broad geographic region, such as the Columbia Plateau in the northwestern US, into a conceptual model of the hydrologic system. These limitations may become resonant in subsequent numerical simulations of hydrologic system behavior. Included among such data limitations are irregular spatial distributions of data, decreases in information with increasing depth from the land surface, uncertainties about the reliability of reported hydrologic data, disparities in time-dependent parameters, and lack of field verification of data. The preparation of a regional hydrologic system description, therefore, first involves a comprehensive data evaluation, wherein the data are classified and ranked in terms of their utility to the study. The results of this evaluation are essential in planning future data acquisition activities, as well as in selecting and developing models. In turn, iterative use of modeling, data refinement, and data acquisition is considered to be highly effective. The case example of preparing a hydrologic system description for the Columbia Plateau, as required for repository siting, illustrates methods of determining the accuracy of certain data, compensating for data limitations, evaluating the need for acquiring additional data, and refining data through iterative techniques. Emphasis is placed on professional subjectivity, which has proven to be essential in data base evaluation and refinement.

527 (PNL-SA--9101) THERMAL PHASE STABILITY OF SOME SIMULATED DEFENSE WASTE GLASSES. May, R.P. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Apr 1981. Contract AC06-76RL01830. 33p. (CONF-810528--11). NTIS, PC A03/MF A01. Order Number DE81028012.

From 83. symposium on nucleation and crystallization in glasses; Washington, DC, USA (3 May 1981).

Three simulated defense waste glass compositions developed by Savannah River Laboratories were studied to determine viscosity and compositional effects on the comparative thermal phase stabilities of these glasses. The glass compositions are similar except that the 411 glasses are high in lithium and low in sodium compared to the 211 glass, and the T glasses are high in iron and low in aluminum compared to the C glass. Specimens of these glasses were heat treated using isothermal anneals as short as 10 min and up to 15 days over the temperature range of 450°C to 1100°C. Additionally, a specimen of each glass was cooled at a constant cooling rate of 7°C/hour from an 1100°C melt down to 500°C where it was removed from the furnace. The following were observed. The slow cooling rate of 7°C/hour is possible as a canister centerline cooling rate for large canisters. Accordingly, it is important to note that a short range diffusion mechanism like cooperative growth phenomena can result in extensive devitrification at lower temperatures and higher yields than a long-range diffusion mechanism can; and can do it without the growth of large crystals that can fracture the glass. Refractory oxides like CeO_2 and $(Ni, Mn, Fe)_2O_4$ form very rapidly at higher temperatures than silicates and significant yields can be obtained at sufficiently high temperatures that settling of these dense phases becomes a major microstructural feature during slow cooling of some glasses. These annealing studies further show that below 500°C there is but little devitrification occurring implying that glass canisters stored at 300°C may be kinetically stable despite not being thermodynamically so.

528 (PNL-SA--9102) CORRELATIONS BETWEEN PH AND GLASS LEACHING. Buckwalter, C.Q.; McVay, G.L. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Apr 1981. Contract AC06-76RL01830. 16p. (CONF-810528--4). NTIS, PC A02/MF A01.

From 83. symposium on nucleation and crystallization in glasses; Washington, DC, USA (3 May 1981).

The effect of pH on the dissolution of glass constituents from three simple silicate glass compositions and one waste glass during leaching experiments is illustrated. All glasses were leached in modified D.I. water according to the MCC-1 procedure. Leachate analysis was done by ICP and solid state analyses by SEM. The major conclusion is that the mechanism of dissolution is different when comparing a silicate and a borosilicate glass. The borosilicate glasses react more readily in an acid medium, whereas the silicate-based glasses are relatively unreactive. The addition of the rare earths in the waste glass enhance the dissolution of the borosilicate glass in the acid medium. Both borosilicate glasses exhibit greater chemical durability in the high base medium, whereas the silicate glass tends to dissolve.

529 (PNL-SA--9110) AEGIS METHODOLOGY AND A PERSPECTIVE FROM AEGIS METHODOLOGY DEMONSTRATIONS. Dove, F.H. (Pacific Northwest Lab., Richland, WA (USA)). Mar 1981. Contract AC06-76RL01830. 28p. (CONF-810372--16). NTIS, PC A03/MF A01. Order Number DE82005927.

From Symposium on uncertainties associated with the regulation of the geologic disposal of high-level radioactive waste; Gatlinburg, TN, USA (9 Mar 1981).

Objectives of AEGIS (Assessment of Effectiveness of Geologic Isolation Systems) are to develop the capabilities needed to assess the post-closure safety of waste isolation in geologic formation; demonstrate these capabilities on reference sites; apply the assessment methodology to assist the NWTS program in site selection, waste package and repository design; and perform repository site analyses for the licensing needs of NWTS. This paper summarizes the AEGIS methodology, the experience gained from methodology demonstrations, and provides an overview in the following areas: estimation of the response of a repository to perturbing geologic and hydrologic events; estimation of the transport of radionuclides from a repository to man; and assessment of uncertainties. (DLC)

530 (PNL-SA--9111) ASSESSMENT OF EFFECTIVENESS OF GEOLOGIC ISOLATION SYSTEMS: THE AEGIS GEOLOGIC SIMULATION MODEL. Foley, M.G.; Petrie, G.M. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Feb 1981. Contract AC06-76RL01830. 17p. (CONF-810372--11). NTIS, PC A02/MF A01. Order Number DE81028007.

From Symposium on uncertainties associated with the regulation of the geologic disposal of high-level radioactive waste; Gatlinburg, TN, USA (9 Mar 1981).

Portions of document are illegible.

Assessment of the post-closure performance of a nuclear waste repository has two basic components: the identification and analysis of potentially disruptive sequences and the pattern of geologic events and processes causing each sequence, and the identification and analysis of the environmental consequences of radionuclide transport and interactions subsequent to disruption of a repository. The AEGIS Scenario Analysis Task is charged with identifying and analyzing potenially disruptive sequences of geologic events and processes. The Geologic Simulation Model (GSM) was developed to evaluate the geologic/hydrologic system surrounding an underground repository, and describe the phenomena that alone, or in concert, could perturb the system and possibly cause a loss of repository integrity. The AEGIS approach is described in this report. It uses an integrated series of models for repository performance analysis: the GSM for a low-resolution, long-term, comprehensive evaluation of the geologic/hydrologic system, followed by more detailed hydrogeologic, radionuclide transport, and dose models to more accurately assess the consequences of disruptive sequences selected from the GSM analyses. This approach is felt to be more cost-effective than an integrated one because the GSM can be used to estimate the likelihoods of different potentially disruptive future evolutionary developments within the geologic/hydrologic system. The more costly consequence models can then be focused on a few disruptive sequences chosen for their representativeness and effective probabilities.

531 (PNL-SA--9114) PRELIMINARY RESULTS OF MONTE CARLO SIMULATIONS FOR THE PASCO BASIN, WASHINGTON, USING THE AEGIS GEOLOGIC SIMULATION MODEL. Petrie, G.M.; Foley, M.G. (Pacific Northwest Lab., Richland, WA (USA)). Mar 1981. Contract AC06-76RL01830. 18p. (CONF-810372--17). NTIS, PC A02/MF A01. Order Number DE82005928.

From Symposium on uncertainties associated with the regulation of the geologic disposal of high-level radioactive waste; Gatlinburg, TN, USA (9 Mar 1981).

(AEGIS stands for Assessment of Effectiveness of Geologic Isolation Systems.) Some typical examples of statistical output from the model are considered. Contingency tables and correlation matrixes are used to show relations between variables. Cumulative distribution functions and density curves are used to characterize significant parameters. It is suggested that if a critical controlling point in a system can be identified, it may be necessary only to reduce the uncertainty at that point rather than to expend more effort on the whole system, and that undue pessimism with regard to system performance may cause an obvious solution to a problem to be overlooked. The modeling effort has identified several benefits from being close to a large ice sheet (glacier): lowered head gradients, reduced ground-water recharge area, increased buffer between waste and biosphere, and reduced uncertainty in man-caused effects. (DLC)

532 (PNL-SA--9645) PERMEABILITY, SWELLING, AND RADIONUCLIDE-RETARDATION PROPERTIES OF CANDIDATE BACKFILL MATERIALS. Westsik, J.H. Jr.; Bray, L.A.; Hodges, F.N.; Wheelwright, E.J. (Pacific Northwest Lab., Richland, WA (USA)). 1981. Contract AC06-76RL01830. 9p. (CONF-811122--44). NTIS, PC A02/MF A01. Order Number DE82005921.

From Annual meeting of the Materials Research Society; Boston, MA, USA (16 Nov 1981).

A backfill placed between a nuclear waste canister and the host geology of a nuclear waste repository can impede the migration of water through the waste package and retard the movement of radionuclides into the geologic formation. Hydraulic conductivities and swelling pressures are being determined as functions of the density of the compacted backfill, temperature, radiation dose, hydraulic head and the chemical composition of the permeating fluid. Bentonite clays and bentonite/sand mixtures have received initial emphasis. Sodium bentonite and calcium bentonite samples compacted to a dry density of 2.1 g/cm^3 had hydraulic conductivities in the range of 10^{-12} to 10^{-13} cm/s. In addition, batch distribution ratios (R/sub d/) for Sr, Cs, Am, Np, I, U and Tc have been measured for a number of candidate backfill materials. Both initial permeability and sorption studies have used a synthetic basaltic ground water.

533 (PNL-SA--9653) SOLUBILITY EFFECTS IN WASTE GLASS-DEMINERALIZED WATER SYSTEMS. Fullam, H.T. (Pacific Northwest Lab., Richland, WA (USA)). Nov 1981. Contract AC06-76RL01830. 9p. (CONF-811122--33). NTIS, PC A02/MF A01. Order Number DE82004003.

From Annual meeting of the Materials Research Society; Boston, MA, USA (16 Nov 1981).

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

534 (PNL-SA--9761) COMPARATIVE EVALUATION OF LINER MATERIALS FOR INACTIVE URANIUM-MILL-TAILINGS PILES. Buelt, J.L.; Barnes, S.M. (Pacific Northwest Lab., Richland, WA (USA)). 1981. Contract AC06-76RL01830. 16p. (CONF-811049--7). NTIS, PC A02/MF A01. Order Number DE82003861.

From 4. symposium on uranium mill tailings management; Fort Collins, CO, USA (26 Oct 1981).

Under the funding of the Department of Energy's Uranium Mill Tailings Remedial Action (UMTRA) Program, Pacific Northwest Laboratory (PNL) has completed the initial accelerated testing phase of eight candidate liner materials. The tests were designed to comparatively evaluate the long term effectiveness of liner materials as a radionuclide and hazardous chemical leachate barrier. The eight materials tested were selected from a technical review of published literature and industrial specialists. Conditions were then identified that would accelerate the aging processes expected in a uranium tailings environment for 1000 years. High calcium leachates were forced through thin layers of clay liners to accelerate the ion exchange rate of sodium and calcium. Asphalt and synthetic materials were accelerated by exposure to elevate temperatures, high concentrations of oxygen, and increased strengths of aqueous oxidizing agents. By comparing the charges of permeability with time of exposure, the most acceptable materials were then identified. These materials are a catalytically airblown asphalt membrane and natural soil amended with sodium bentonite. Both materials showed an increased resistance to leachate penetration throughout the exposure period with final permeabilities less than 10^{-7} cm/s. In addition, the asphalt membrane and sodium bentonite are among the least expensive materials to install at a disposal site. Therefore based on their economic and technical merits, these two materials are being evaluated further in field tests at Grand Junction, Colorado.

535 (PNL-SA--9815) FIELD TESTING OF ASPHALT-EMULSION RADON-BARRIER SYSTEM. Hartley, J.N.; Freeman, H.D.; Baker, E.G.; Elmore, M.R.; Nelson, D.A.; Voss, C.F.; Koehmstedt, P.L. (Pacific Northwest Lab., Richland, WA (USA)). Sep 1981. Contract AC06-76RL01830. 31p. (CONF-811049--6). NTIS, PC A03/MF A01. Order Number DE82003830.

From 4. symposium on uranium mill tailings management; Fort Collins, CO, USA (26 Oct 1981).

Three years of laboratory and field testing have demonstrated that asphalt emulsion seals are effective radon diffusion barriers. Both laboratory and field tests in 1979, 1980 and 1981 have shown that an asphalt emulsion seal can reduce radon fluxes by greater than 99.9%. The effective diffusion coefficient for the various asphalt emulsion admix seals averages about 10^{-6} cm^2/s. The 1981 joint field test is a culmination of all the technology developed to date for asphalt emulsion radon barrier systems. Preliminary results of this field test and the results of the 1980 field test are presented. 18 figures, 6 tables.

536 (PNL-SA--9820) MULTIDIMENSIONAL ANALYSIS OF RADON DIFFUSION THROUGH URANIUM MILL TAILINGS AND COVER SYSTEMS. Mayer, D.W.; Nelson, R.W.; Gee, G.W. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 1981. Contract AC06-76RL01830. 13p. (CONF-811049--10). NTIS, PC A02/MF A01. Order Number DE82004007.

From 4. symposium on uranium mill tailings management; Fort Collins, CO, USA (26 Oct 1981).

A new capability has been developed for modeling radon diffusion through uranium mill tailings and covers. This capability allows the transient solution for radon concentrations and fluxes in one, two and three dimensions. The method utilizes an integrated-finite difference solution to the conservation of mass equation for radon gas. The effects of variable soil moisture content can also be taken into account. Results are presented from the analysis of both bare and covered tailings piles. Comparisons of the numerical results are made with an available one dimensional analytic solution. Two dimensional effects of a discontinuous cover system are also analyzed.

537 (PNL-SA--9895) SOIL-WATER IMPACTS FROM USING VEGETATION AND ROCK COVERS FOR SURFACE STABILIZATION OF URANIUM-MILL TAILINGS. Mayer, D.W.; Beedlow, P.A.; Cadwell, L.L. (Pacific Northwest Lab., Richland, WA (USA)). 1982. Contract AC06-76RL01830. 9p. (CONF-820303--12). NTIS, PC A02/MF A01. Order Number DE82008585.

From Waste management conference; Tucson, AZ, USA (8 Mar 1982).

This paper presents the results from an analysis of vegetated and rock covers and their effect on the moisture content in a covered uranium mill tailings system. Based on a one-dimensional analysis of moisture movement, the results indicate that care must be taken when selecting a surface stabilization system for a tailings pile. The moisture-content response of the tailings pile and cover system can be radically altered by different surface treatments. The two cases considered in this study indicate that (under climatic conditions occurring at Grand Junction, Colorado) the evapotranspiration from a vegetated cover can result in a relatively stable moisture content. A rock cover, however, may increase the moisture content of the tailings pile by significantly reducing evaporation. In fact, moisture storage may increase to the point that drainage occurs. If drainage does occur, the potential for groundwater pollution is increased. These results suggest that vegetation, thinner rock covers, engineered drainage systems, and/or liner systems may be needed to reduce drainage and potential leaching of contaminants. Additional work is needed to improve the description of the surface boundary condition and provide a more accurate moisture sink term. This work should focus on better descriptions of plant growth and moisture extraction behavior as a function of climatological and soil conditions. Additional work is required to more accurately describe the diffusion of water vapor through rock covers, and to quantify the effects of wind.

538 (PNL-SA--9913) MANAGEMENT OF OCEAN DISPOSAL OF RADIOACTIVE WASTES: A BASIS FOR THE CONTROL OF OTHER POLLUTANTS. Templeton, W.L. (Pacific Northwest Lab., Richland, WA (USA)). Sep 1981. Contract AC06-76RL01830. 20p. (CONF-8110104--2). NTIS, PC A02/MF A01. Order Number DE82003956.

From 3. international ocean dumping symposium; Woods Hole, MA, USA (12 Oct 1981).

To manage, on a scientific basis, the quantities of all kinds of waste disposal to coastal waters and open oceans it is necessary to assess the environmental or assimilative capacity for these materials which will not result in an unacceptable biological impact upon the components of the ecosystem nor on man who uses its resources. One approach available is that which has been demonstrated for the management of the disposal of radioactive wastes to the oceans. Methodologies have been developed, both generic and site-specific, which allow the relationship between discharge or release rate and the radiation dose to be established. Guidelines and recommendations which govern acceptable radiation exposed to man have been developed by the International Commission on Radiological Protection (ICRP). These methodologies developed for the control of radioactive wastes can be applied directly for public health protection for non-radioactive wastes such as metals and organochlorine pesticides. ICRP recommendations on justification and optimization can be integrated into an overall management philosophy in order to quantify alternative waste disposal options.

539 (PNL-SA--9957) EFFECTS OF SURFACE AREA TO VOLUME RATIO AND SURFACE ROUGHNESS ON WASTE GLASS LEACHING. McVay, G.L.; Buckwalter, C.Q.; Pederson, L.R. (Pacific Northwest Lab., Richland, WA (USA)). 1981. Contract AC06-76RL01830. 25p. (CONF-811086--3). NTIS, PC A02/MF A01. Order Number DE82005229.

From American Ceramic Society conference; Newport Beach, CA, USA (25 Oct 1981).

This study assesses the effects of surface roughness and surface area to solution volume (SA/V) ratio changes on the leaching of PNL 76-68, a simulated nuclear waste glass. Monolith specimens were used to avoid the difficulties associated with powders. Inductively Coupled Argon Plasma Spectroscopy (ICP) was used to analyze leachate solutions. Solid samples were depth profiled using Secondary Ion Mass Spectroscopy (SIMS). Corrosion of PNL 76-68 glass increased with increased surface roughness at a constant solution volume, in agreement with earlier studies on simple glasses. However, the increased leaching occurred only for several days at 90°C, after which the rates of reaction of glasses given three widely different surface preparations were essentially identical. Surface roughness of a waste glass in a repository thus is not a major consideration in determining elemental release rates. In sharp contrast to the results of earlier studies on simple silicate glasses, leaching of PNL 76-68 glass strongly decreased with increases in the SA/V ratio. These findings imply that corrosion within cracks in a waste glass monolith exposed to water is much less important than previously suspected on the basis of simple glass measurements. Surfaces within small cracks in the waste glass are actually more durable than are the outer surfaces.

540 (PNL-SA--10019) EFFECT OF ORGANIC COMPLEXANTS ON THE MOBILITY OF LOW LEVEL WASTE RADIONUCLIDES IN SOILS. Swanson, J.L. (Pacific Northwest Lab., Richland, WA (USA)). Oct 1981. Contract AC06-76RL01830. 12p. (CONF-811130--16). NTIS, PC A02/MF A01. Order Number DE82005839.

From 3. annual DOE participants information meeting on low-level waste management; New Orleans, LA, USA (4 Nov 1981).

The effect of organic complexants, EDTA and DTPA, on the equilibrium sorption properties of europium, cesium strontium, cobalt and nickel on Hanford soils were investigated. All work was performed in the presence of air. Humic acid which is also a complexant occurs in some soils. The goal of the work was to obtain data that are at equilibrium so that the results would be meaningful in a thermodynamic sense. This goal was met with europium, cesium and strontium, but was not possible with cobalt and nickel where slow kinetics prevented equilibrium data from being obtained. (ATT)

541 (PNL-SA--10178) INTERNATIONAL ASPECTS OF THE MANAGEMENT OF LOW-LEVEL DUMPING OF RADIOACTIVE WASTES IN THE OCEANS. Templeton, W.L. (Pacific Northwest Lab., Richland, WA (USA)). Jan 1982. Contract AC06-76RL01830. 15p. (CONF-820303--14). NTIS, PC A02/MF A01. Order Number DE82008588.

From Waste management conference; Tucson, AZ, USA (8 Mar 1982).

Portions of document are illegible.

The following topics are discussed: international regulations governing radioactive waste disposal; radiological principles as applied to disposal to the environment; historical dumping practices; assessment of the North East Atlantic dump site; IAEA generic studies; and national and international implications. A recent analysis of international issues associated with ocean disposal of low-level radioactive wastes indicated a number of points which impact on US needs and policies and need resolution. The first is that the development of adequate international criteria and standards will assist the US in evaluating the option of using the oceans for the disposal of low-level radioactive wastes. Secondly, it is essential that international cooperation in research and radiological surveillance be expanded. Thirdly, the delays in the agreements on international mechanisms, criteria and standards, sometimes as a direct result of a lack of coordinated US policies makes the implementation of the intent of the London Dumping Convention and the NEA mechanism more difficult. Last of all in the unresolved question of how the US should apply the London Convention to the 200 mile exclusive economic zone. (ATT)

542 (PNL-SA--10347-Vol.2, pp 733-738) SOME LIKE IT HOT, SOME LIKE IT COLD. Yellott, J.I. (Arizona State Univ., Tempe). 1981. NTIS (US Sales Only).

From International energy storage conference; Seattle, WA, USA (19 Oct 1981).

The purpose of this paper is not to deal in any depth with the technical agenda of the conference, since most of the conferees know far more about seasonal energy storage and compressed air than I, despite the fact that I have devoted most of a 50 year career to the study of energy and its uses. Rather, the attempt is to consider the concept of production of or procurement of essential commodities, of which energy is only one, during periods of abundance and the preservation of those commodities for later use during periods of scarcity.

543 (RHO-BWI-C--61) WELL LOCATION AND LAND-USE MAPPING IN THE COLUMBIA PLATEAU AREA. Stephan, J.; Foote, H.; Coburn, V. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Oct 1979. Contract AC06-77RL01030. 159p. (PNL--3295). NTIS, PC A08/MF A01.

Irrigation wells in a 41,000-square mile area located in Washington and northern Oregon were the subject of this study. Approximately 30,000 square miles of the area were mapped within the boundary of the Columbia Plateau, which covers some 48,200 square miles in the states of Washington, Oregon, and Idaho. Advanced state-of-the-art computer analysis techniques for processing Landsat digital multispectral data were used for mapping the area into ten land-use classes. Specially designed computer programs were used for mapping the locations of 1476 irrigation wells located in 13 counties. Six thematic color-encoded maps were prepared which show additional land-use types and relative areal distribution. Three maps depict the location of irrigation wells.

544 (RHO-BWI-SA--43) LANDSAT DATA AS A BASIS FOR REGIONAL ENVIRONMENTAL ASSESSMENT WITHIN THE COLUMBIA PLATEAU. Leonhart, L.S.; Stephan, J.G. (Atomics International Div., Richland, WA (USA). Rockwell Hanford Operations; Battelle Pacific Northwest Labs., Richland, WA (USA)). May 1980. Contract AC06-77RL01030. 19p. (CONF-800577--1). NTIS, PC A02/MF A01.

From 6. Canadian symposium on remote sensing; Halifax, Canada (21 May 1980).

LANDSAT data are being used to assist in classifying land cover for the purpose of providing regional environmental assessments of a 105,000 km² area within the Columbia Plateau region of the northwestern United States. The environmental assessment is being performed in conjunction with siting a radioactive waste terminal-storage facility within the federally operated Hanford Site in south-central Washington. Initial assessments involved classification of four contiguous LANDSAT scenes recorded over the region during the summer of 1975 and about 300 1978 high-altitude

(U-2) photographs. The resulting mosaic was segregated into ten land-cover classes. The classified land-cover data were then machine-integrated with digital irrigation well-location data taken from the US Geological Survey's Ground-Water Site Inventory. The resulting groundwater multi-source data product was required by hydrologists to segregate potential artificial recharge areas from artificial groundwater discharge areas. Related studies have employed LANDSAT data and aerial imagery to identify linear structures and other geologic features which may have a significant bearing on the tectonic and/or hydrologic setting of the Columbia Plateau. Future plans call for attempts to correlate land-cover patterns with other physical and environmental parameters such as digital terrain and hydrometeorological data.

545 (RHO-C--39) COMPARISON OF THE MICROSTRUCTURE OF HANFORD TYPE II CONCRETE STRUCTURES AND TEST SPECIMENS. Daniel, J.L.; Buck, A.D. (Battelle Pacific Northwest Labs., Richland, WA (USA); Rockwell International Corp., Richland, WA (USA). Rockwell Hanford Operations; Corps of Engineers, Vicksburg, MS (USA). Waterways Exeriment Station). Apr 1980. Contract AC06-77RL01030. 96p. NTIS, PC A05/MF A01.

High-level radioactive waste from the current waste management operation at Hanford is stored in underground reinforced concrete tanks with a steel liner on the sides and bottom. The removal of pumpable liquid results in a moist salt-cake of sodium salts that is relatively immobile compared to its original liquid state. Additional immobilization and isolation techniques are being studied to further improve waste isolation and containment. Retrieval of the waste and permanent disposal will be deferred until a permanent disposal mode and location is selected. The integrity of the storage tanks is of critical importance during this interim period. Technical studies and laboratory tests have been performed to determine the effect of the stored waste's temperatures and chemistry on the reinforced concrete's strength and elastic properties. However, relatively little attention has been given to studying or predicting very long-term behavior based on short-term specimen examination. This report describes the results of the first phase of a laboratory program for which the ultimate goal is the ability to predict the probable condition of Hanford Type II concrete structures after 100 years of more of lifetime. The purpose of this phase was the evaluation of the microstructure of several existing test and field samples of Hanford concrete, and to correlate microstructural data with the physical test measurements on hand from other programs and tasks.

546 (UCRL--81147(Pt.2)(Rev.1)) LEACHING CHARACTERISTICS OF ACTINIDES FROM SIMULATED REACTOR WASTE. Weed, H.C.; Coles, D.G.; Bradley, D.J.; Mensing, R.W.; Schweiger, J.S.; Rego, J.H. (Lawrence Livermore National Lab., CA (USA)). 1979. Contract W-7405-ENG-48. 9p. (CONF-791112--69(Pt.2)(Rev.1)). NTIS, PC A02/MF A01.

From Materials Research Society annual meeting; Boston, MA, USA (26 Nov 1979).

Leach rates for ^{237}Np and ^{239}Pu are investigated with a single-pass leaching system. The factorial experimental design uses several combinations of solution composition and flow rate; and two temperatures, 25° and 75°C. The 25°C results are compared with those form a modified IAEA procedure. At 25°C, leach rates decrease with time. Agreement between results from the single-pass and modified IAEA methods is fair with WIPP brine leachant, good with $NaHCO_3$, and good with distilled H_2O. Leach rates are approximately independent of flow rates at room temperature, but increase with flow rates at high temperature. Rates for ^{237}Np increase with temperature, but those fro ^{239}Pu either decrease or do not change with temperature.

547 (Y/OWI/SUB--77/14268, pp 29-31) OVERVIEW OF WASTE ISOLTAION SAFETY ASSESSMENT PROGRAM AND DESCRIPTION OF SOURCE TERM CHARACTERIZATION TASK AT PNL. Bradley, D. (Battelle-Pacific Northwest Labs., Richland, WA). 20 Aug 1977.

From Conference on waste-rock interactions; University Park, PA, USA (6 Jul 1977).

A project is being conducted to develop and illustrate the methods and obtain the data necessary to assess the safety of long-term disposal of high-level radioactive waste in geologic formations. The methods and data will initially focus on generic geologic isolation systems but will ultimately be applied to the long-term safety assessment of specific candidate sites that are selected in the NWTS Program. The activities of waste isolation safety assessment (WISAP) are divided into six tasks: (1) Safety Assessment Concepts and Methods, (2) Disruptive Event Analysis, (3) Source Characterization, (4) Transport Modeling, (5) Transport Data and (6) Societal Acceptance. (JRD)

548 INFLUENCE OF AN AMERICIUM SOLID PHASE ON AMERICIUM CONCENTRATIONS IN SOLUTIONS. Rai, D.; Strickert, R.G.; Moore, D.A.; Serne, R.J. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Geochimica et Cosmochimica Acta ; 45: No. 11, 2257-2265(Nov 1981).

Americium-241 concentrations in solutions contacting contaminated sediments for up to 2 yr were measured as a function of pH. Steady-state concentrations were reached within a few days. The solubility-limited Am concentration was found to decrease approximately 10-fold with one unit increase in pH. The log equilibrium constant for the solubility of Amsub(soil) solid was found to be -4.12. The predictions based upon thermodynamic data suggest that Am*sub(aq.complex) is likely to be $Am(OH)_2^+$. Although the chemical formula of Amsub(soil) was not determined, it does not appear to be $Am(OH)_3(a)$. Published data on sorption coefficients of Am by different rocks, soils, and minerals were critically evaluated. Final Am solution concentrations calculated from the sorption coefficients of a variety of earth materials with several solutions agreed well with the concentrations predicted from the solubility of Amsub(soil) solid, indicating that the sorption coefficient data are controlled by Am precipitation.

549 (PNL-TR--410) CORROSION TESTING OF UNALLOYED TITANIUM IN SIMULATED DISPOSAL ENVIRONMENTS FOR REPROCESSED NUCLEAR WASTE. FINAL REPORT. Henrikson, S.; de Pourbaix, M. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 7 Apr 1981. Contract AC06-76RL01830. Translation of STUDSVIK-E1--79/83, 7 May 1979 42p. NTIS, PC A03/MF A01.

On the commission of the Nuclear Safety Project (KBS) corrosion tests have been carried out on unalloyed titanium, which is planned to be used as an outer corrosion-resistant canister for reprocessed nuclear waste. Tests were conducted for 300 days in a corrosive medium of modified Baltic water at 100 and

130°C with high (8 ppM) or low (< 10 ppB) oxygen content. The tests at low oxygen content and 100°C were continued for 600 days. Very low oxidation rates (0.01-0.1 µm/year) were obtained, corresponding to a lifetime of tens to hundreds of thousands of years for a 6-mm-thick titanium canister. In spite of the considerably more severe corrosion conditions (higher temperature, higher chloride and fluoride contents, as well as lower pH) compared with those estimated for the final disposal, no signs of localized corrosion were found, nor could any hydrogen pick-up be detected.

550 EFFECT OF GAMMA RADIATION ON GLASS LEACHING. McVay, G.L.; Pederson, L.R. (Battelle, Pac Northwest Lab, Richland, Wash, USA). Journal of the American Ceramic Society : 64: No. 3, 154-158(Mar 1981).

The present study considers the effect of gamma irradiation on the leaching stability of a simulated nuclear waste glass in aqueous solutions. Significant increases in the elemental release rates on the glass during irradiation, as reported here, demonstrate the importance of consideration of this parameter in predicting long-term stabilities in geologic waste repositories. 16 refs.

551 DEVELOPMENT OF STRUCTURAL ENGINEERED BARRIERS FOR THE LONG-TERM CONTAINMENT OF NUCLEAR WASTE. Westerman, R.E. (Battelle Memorial Inst., Richland, WA). pp 515-522 of Scientific basis for nuclear waste management. Volume 3. Moore, J.G. (ed.). New York, NY: Plenum Press (1981). Contract W-7405-ENG-26.

From 3. annual meeting of the Materials Research Society; Boston, MA, USA (17 Nov 1980).

A task directed toward the development of licensable engineered barrier systems for the long-term containment of high-level nuclear waste under conditions of deep geologic disposal has been underway at the Pacific Northwest Laboratory (PNL) since January 1979. Metallic, ceramic, and polymeric materials are being investigated by means of mechanical property and corrosion screening studies to determine the suitability of these materials for use as long-lived structural barriers, i.e., canister, overpack, and hole sleeve, in nuclear waste packages. At the present time the barrier program at PNL is primarily involved in conducting screening studies, though some in-depth investigations are underway, notably in the mechanical testing area. The actual status of the subtask work elements and the results obtained to date for the following are presented: mechanical property evaluation of metallic materials; corrosion evaluation of metallic materials; ceramic materials; and polymeric materials.

552 IN-SITU RADIATION MEASUREMENTS AND INSTRUMENTATION FOR MONITORING NUCLEAR WASTE STORAGE FACILITIES. Brodzinski, R.L. (Battelle Pacific Northwest Labs., Richland, WA). IEEE (Institute of Electrical and Electronics Engineers) Transactions on Nuclear Science; NS-27: No. 4, 1277-1279(Aug 1980).

A summary is presented of the reasons, requirements, and methodology for monitoring the integrity of nuclear waste storage facilities. Various types of nuclear wastes and the radiation emitted by them are defined. Descriptions of commonly used storage facilities are given. Instrumentation available or under development for making these measurements is described with particular emphasis on in-situ techniques.

553 (PNL-TR--408) BEHAVIOR OF RADIOACTIVE WASTES INCORPORATED IN CONCRETE AND STORED ON OPEN-AIR BED AT CAMEN. Beltiore, A.; Lo Moro, A.; Orsini, T.; Panciatici, G. Translated from Energia Nucleare (Milan) ; 27: No. 2, vp(Feb 1980). Contract AC06-76RL01830. 8p. NTIS, PC A02/MF A01.

A study was performed on the state of preservation of some 400 concrete containers used to condition radioactive wastes, stored in the temporary open-air bed at Centro Applicazioni Militari Energia Nucleare (CAMEN). The investigation, whose results are reported in the present note, showed the effectiveness of the adopted method of storing the containers in the open.

554 (PNL-TR--384) CEDRA INFORME: NATIONAL COOPERATIVE ASSOCIATION FOR THE STORAGE OF RADIOACTIVE WASTES. 31 Jan 1980. Contract EY-76-C-06-1830. Translation of CEDRA Informe, No. 6/1979 November 34p. Dep. NTIS, PC A03/MF A01.

Activities of the CEDRA, a Swiss organization, in the field of radioactive waste storage are reported. Besides various international cooperative projects, an underground laboratory will be constructed at Grimsel. International news briefs are included. The Swiss program for nuclear waste management is discussed. (DLC)

555 ASPHALT EMULSION SEALING OF URANIUM MILL TAILINGS. Hartley, J.N.; Koehmstedt, P.L.; Esterl, D.J. (Battelle Pacific Northwest Labs., Richland, WA). pp 681-688 of Scientific basis for nuclear waste management. Northrup, C.J.M. Jr. (ed.). New York, NY: Plenum Press (1980).

From Materials Research Society annual meeting; Boston, MA, USA (26 Nov 1979).

The use of asphalt emulsion to contain radon and radium in uranium tailings is being investigated at the Pacific Northwest Laboratory. Results of these studies indicate that a radon flux reduction of greater than 99% can be obtained using either a poured-on/sprayed-on seal (3.0 to 7.0 mm thick) or an admix seal (2.5 to 15.2 cm thick) containing about 18 wt % residual asphalt. A field test was carried out at the Grand Junction tailings pile in order to demonstrate the sealing process. A reduction in radon flux ranging from 4.5 to greater than 99% (76% average) was achieved using a 15.2 cm (6 in.) admix seal with a sprayed-on top coat. A hydrostatic stabilizer was used to apply the admix. This was followed by compaction to form the radon seal. Overburden was applied to provide a protective soil layer over the seal. Included in part of the overburden was a herbicide to prevent root penetration.

556 COMPLEX RELATIONSHIP BETWEEN GROUNDWATER VELOCITY AND CONCENTRATION OF RADIOACTIVE CONTAMINANTS. Kaszeta, F.E.; Bond, F.W. (Battelle Pacific Northwest Lab., Richland, WA). pp 739-745 of Scientific basis for nuclear waste management. Northrup, C.J.M. Jr. (ed.). New York, NY; Plenum Press (1980).

From Materials Research Society annual meeting; Boston, MA, USA (26 Nov 1979).

This paper uses the results from the Multi-component Mass Transport model to examine the complex interrelationship between groundwater velocity and contaminant dispersion, decay, and retardation with regard to their influence on the contaminant concentration distribution as it travels through the geosphere to the biosphere. The rate of transport of contaminants through the geosphere is governed by groundwater velocity, leach rate, and contaminant retardation. The dominant characteristics of the contaminant concentration distribution are inherited during leaching and modified during transport by dilution, dispersion and decay. For a hypothetical non-decaying, non-dispersing contaminant with no retardation properties, the shape of the source term distribution is governed by the groundwater velocity (dilution) and leach rate. This distribution remains unchanged throughout transport. Under actual conditions, however, dispersion, decay and retardation modify the concentration distribution during both leaching and transport. The amount of dispersion is determined by the distance traveled, but it does have a greater peak-reducing influence on spiked distributions than square-shaped distributions. Decay acts as an overall scaling factor on the concentration distribution. Retardation alters the contaminant travel time and therefore indirectly influences the amount of dilution, dispersion and decay. Simple relationships between individual parameters and groundwater velocity as they influence peak concentration do not exist. For those cases where the source term is not solubility-limited and flow past the waste is independent of regional hydrologic conditions, a threshold concentration occurs at a specific groundwater velocity where the effects of dilution balance those of dispersion and decay.

557 BIOBARRIERS USED IN SHALLOW-BURIAL GROUND STABILIZATION. Cline, J.F. (Battelle Northwest Lab., Richland, WA). Transactions of the American Nuclear Society ; 34: 121-122(1980). (CONF-800607--).
From American Nuclear Society annual meeting; Las Vegas, NV, USA (8 Jun 1980).
RADIOACTIVE WASTE DISPOSAL;UNDERGROUND DISPOSAL;LOW-LEVEL RADIOACTIVE WASTES;HAPO; GROUTING;ROCKS;RADIATION PROTECTION;MATERIALS TESTING

558 GEOHYDROLOGIC TECHNOLOGY DEVELOPMENTS FOR RETIRED RADIOACTIVE WASTE BURIAL SITE DECOMMISSIONING. Phillips, S.J. (Battelle Pacific Northwest Labs., Richland, WA (USA)). pp 69-82 of Proceedings of the specialist meeting on decommissioning requirements in the design of nuclear facilities, Paris, 17-19 March 1980. Paris, France; OECD (1980).
From NEA specialist meeting on decommissioning requirements in the design of nuclear facilities; Paris, France (17 Mar 1980).
Terminal disposition of radioactive wastes in shallow-land disposal sites requires waste containment in the partially saturated geohydrologic system until such time as radioisotopes in the buried waste reach innocuous levels as a result of decay. In order to assure waste containment, technologies must be developed to characterize and monitor retired, operational, and future burial sites. This paper discusses examples of geophysical, geochemical, and geohydrologic field and laboratory instrument systems and methods used to assess burial sites. From results and applications of these systems and methods, examples of recommendations for future design of shallow-land burial sites are presented.

559 (PNL-TR--378) STATEMENT BY DR. ERNST ALBRECHT ON THE PROVINCIAL GOVERNMENT'S POSITION REGARDING THE PLANNED NUCLEAR WASTE DISPOSAL FACILITY (NEZ) IN GORLEBEN. PART II. POSITION TAKEN BY THE FEDERAL GOVERNMENT REGARDING THE POLICY STATEMENT BY THE NIEDERSACHSEN PROVINCIAL GOVERNMENT ON THE PLANNED NUCLEAR WASTE DISPOSAL FACILITY IN GORLEBEN. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 6 Dec 1979. Contract EY-76-C-06-1830. Translation source information not available 15p. Dep. NTIS, PC A02/MF A01.
Questions regarding the safety and policy aspects of the proposed DWK waste disposal center in Gorleben are discussed. The position of the Niedersachsen provincial government is that the center should not be built; instead, the salt formations should be tested and interim fuel storage facilities provided. The response of the Federal government to the Niedersachsen statement is also listed.

560 METHODS USED TO ESTIMATE GEOCHEMICAL ASPECTS OF RADIONUCLIDE MIGRATION FROM WASTE REPOSITORIES. Serne, R.J.; Bradley, D.J. (Battelle Northwest Labs., Richland, WA). Transactions of the American Nuclear Society ; 32: 161-162(Jun 1979). (CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).
RADIONUCLIDE MIGRATION;RADIOACTIVE WASTE DISPOSAL;GEOLOGIC DEPOSITS;GROUND WATER; LEACHING;ADSORPTION;DESORPTION;RADIOISOTOPES; RADIOACTIVE WASTES

561 LEACHING OF FULLY RADIOACTIVE HIGH-LEVEL WASTE GLASS AND WASTE--GEOLOGIC ENVIRONMENT INTERACTION STUDIES. Bradley, D.J. (Battelle Pacific Northwest Labs., Richland, WA). pp 75-91 of Radioactive waste in geologic storage. Fried, S. (ed.). Washington, DC; American Chemical Society (1979).
From American Chemical Society meeting; Miami, FL, USA (10 Sep 1978).
The release rate was determined for 10 radioisotopes from fully radioactive waste glasses in deionized water for a period of 1.75 years. For cesium and strontium, good agreement exists between the leach rates for simulated and fully radioactive glass of the same composition. The release rate mechanism is dependent on time, sampling frequency, and type of element. For this study, only sampling intervals greater than one month had significant impact on the leach rate. Over the long testing period two different release mechanisms occurred. 8 figures, 3 tables.

562 HYDROTHERMAL GLASS REACTIONS IN SALT BRINE. Westsik, J.H. Jr.; Turcotte, R.P. (Battelle Pacific Northwest Labs., Richland, WA). pp 341-344 of Scientific basis for nuclear waste management. McCarthy, G.J. (ed.). New York, NY; Plenum Press (1979).
From Annual meeting of Materials Research Society; Boston, MA, USA (28 Nov 1978).
A simulated high-level waste glass was exposed to high-temperature salt brine and deionized water. The glass undergoes partial crystallization, depending on the conditions, and yields $NaFeSi_2O_6$ as the primary alteration product. At the extreme temperature of these tests, cesium, rubidium, strontium and molybdenum show moderate-to-high solubilities in salt brine. Solubility in deionized water is an order of magnitude lower than in salt brine, but the glass alteration rate is higher. In a similar salt brine test, a supercalcine showed similar cesium and strontium releases when compared to the glass.

563 INTERACTION OF WASTE RADIONUCLIDES WITH GEOMEDIA: PROGRAM APPROACH AND PROGRESS. Relyea, J.F.; Rai, D.; Serne, R.J. (Battelle Pacific Northwest Labs., Richland, WA). pp 379-394 of Scientific basis for nuclear waste management. McCarthy, G.J. (ed.). New York, NY: Plenum Press (1979).

From Annual meeting of Materials Research Society; Boston, MA, USA (28 Nov 1978).

The Waste Isolation Safety Assessment Program (WISAP) approach is discussed. The data for formation-dissolution of solid phases of the radionuclides and adsorption and exchange of radionuclides by the geomedia are needed for determining radionuclide-geomedia interactions. In order to evaluate the effect of formation and dissolution of minerals and precipitates on solution composition and concentration, the available thermodynamic data were summarized. From these data, solid phase and solution species diagrams were developed to predict the most stable solid and solution phases and the radionuclide concentrations at equilibrium with them in different environments. Computer codes are being developed or modified to handle the thermodynamic data to (i) avoid repetitive calculations by hand and (ii) allow easy modification or inclusion of new data. Solubility isotherms of different elements for a few selected minerals will also be generated. Kinetic effects are ignored in thermodynamic calculations. If the reaction kinetics can be ignored, these calculations should be valid for the required isolation period. Some of the problems that have been encountered in carrying out this problem are discussed. Data analysis and validation techniques are covered.

564 (PNL-TR--353) MODEL OF THE TRANSFER OF DISSOLVED RADIOELEMENTS IN DEEP GEOLOGICAL FORMATIONS. de Marsily, G.; Goblet, P.; Ledoux, E.; Barbreau, A. Dec 1978. Contract EY-76-C-06-1830. Translated from French Society for Radiological Protection, fourth seminar on dispersion in natural media (CONF-780392--1(Trans)). 87p. Dep. NTIS, PC A05/MF A01.

From 4. seminar on dispersion in natural media; Cadarache, France (13 Mar 1978).

The disposal of nuclear wastes in deep geological strata presents the problem of their possible return to the human environment through natural flows of groundwater. To estimate the risks which these migration mechanisms might represent, it is necessary to model the migration of radioelements dissolved in formations of very low permeability. Transport is governed by convection, which is the carrying along at the mean velocity of the fluid; molecular diffusion caused by concentration gradients; kinematic dispersion caused by the heterogeneity of the real velocity vector of the flow in a rock, relative to the mean convective velocity; and retention of radioelements by the solid and/or the immobile fluid fraction. These mechanisms are introduced schematically into a transport equation and several problems presented by this mathematical representation of the transport are examined. Some mechanisms which may have an indirect effect on the migration of radioelements by combining with the natural mechanisms are cited.

565 EFFECT OF SOIL AFFIXANTS ON GERMINATION, EMERGENCE, AND GROWTH OF CHEATGRASS AND RUSSIAN THISTLE. Cline, J.F. (Battelle Pacific Northwest Labs., Richland, WA (USA)); Holter, G. Health Physics; 35: No. 2, 409-411(Aug 1978).

Shallow burial has been used successfully for the disposal of low-level radioactive wastes at Hanford. Soil affixants are used to control soil erosion by wind and to prevent plant roots or burrowing animals from entering the buried material. Growth chamber experiments were conducted to determine whether the affixants have any adverse effect on seed germination or plant growth. Layers of latex emulsions and asphalt emulsion mixtures applied to soil surfaces did not reduce seed germination significantly. There was no visible evidence of plant toxicity induced by the treatments. Asphalt emulsion and asphalt-neoprene reduced loss of soil water, but water utilization by plants growing in latex treated pots was nearly the same as for controls. Results suggest that these treatments can be used as soil stabilizers, to control the rate of moisture flow from soil and to help control wind erosion.

566 LONG-TERM SAFETY ASSESSMENT OF GEOLOGIC ISOLATION REPOSITORIES. Burkholder, H.C.; Greenborg, J.; Stottlemyre, J.A.; Bradley, D.J.; Raymond, J.R.; Serne, R.J. (Battelle Northwest Labs., Richland, WA). Transactions of the American Nuclear Society ; 28: 93-84(Jun 1978).

From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).

See CONF-780622--.

Progress in development of long-term safety technology. RADIOACTIVE WASTE DISPOSAL;GEOLOGIC DEPOSITS;SAFETY ENGINEERING;EVALUATION; MATHEMATICAL MODELS;RADIONUCLIDE MIGRATION;DATA ACQUISITION;DATA ANALYSIS;UNDERGROUND STORAGE

567 UTILITY SPENT FUEL STORAGE EXPERIENCE. Johnson, A.B. Jr. (Battelle Pacific Northwest Lab., Richland, WA). pp 16p, Paper 10 of Spent fuel policy and its implications. Executive conference proceedings. La Grange Park, IL; American Nuclear Society (Apr 1978).

From Conference on spent fuel policy and its implications; Buford, GA, USA (2 Apr 1978).

Past experience with storage of spent fuel elements in pools are reviewed. Data are presented which demonstrate that stainless and Zircaloy-clad spent light water reactor fuel elements have been stored for periods of 12 to 18 years, respectively, without damage due to storage in pools. Techniques for storing elements damaged in the reactor or in transfer to storage are also described. Routine surveillance and exploratory fuel examinations during storage are recommended. It is recommended that past experiences would justify expansion of fuel storage facilities and extension of storage time for commercial water reactor fuel.

568 (BNWL-tr--272) EVALUATION OF THE SAFETY OF STORING RADIOACTIVE WASTES IN GEOLOGICAL FORMATIONS: A PRELIMINARY APPLICATION OF THE FAULT TREE ANALYSIS TO SALT FORMATIONS. Bertozzi, G.; D'Alessandro, M.; Girardi, F.; Vanossi, M. 24 Oct 1977. Contract EY-76-C-07-1830. Translation of French report (CONF-770565--2). 23p. Dep. NTIS, PC A02/MF A01.

From Workshop on risk analysis and geologic modelling; Ispra, Italy (23 May 1977).

Two imaginary formations were selected: salt bed and salt dome. Hypotheses on their stratigraphic, hydrologic, and geomorphologic conditions were made. Fault tree analysis was used on the various groups of phenomena which could cause the geological barrier to fail. The types of failures and their probabilities were evaluated on the basis of four time periods (10^3, 10^4, 10^5, 10^6 years). (DLC)

569 (BNWL-tr--271) GEOLOGY PROGRAM: CHRONOLOGICAL ASPECTS OF WASTE HANDLING STATUS REPORT, AUGUST 1977. Sep 1977. Contract EY-76-C-06-1830. Translated from a Swedish paper, KBS information 04 22p. Dep. NTIS, PC A02/MF A01.

The Geology program is intended to determine the bedrock and ground water conditions for a number of chosen sites for waste storage installations in Swedish primary rock formations. Objectives, research methods, some results migration, and schedules are given. The time elapsed between the removal of used fuel from the reactor and final waste deposition is discussed. The status of the various KBS (Nuclear Fuel Safety) projects (safety analysis, glass leeching, drilling, Stripa, etc.) is summarized. (DLC)

570 IN SITU SUBTERRANEAN GAMMA-RAY SPECTROSCOPY. Nielson, H.L.; Wogman, N.A.; Brodzinski, R.L. (Battelle Pacific Northwest Labs., Richland, Wash. (USA)). Nuclear Instruments and Methods; 143: No. 2, 385-399(1 Jun 1977).

The adaptation of Ge(Li) and intrinsic germanium diodes and small NaI(Tl) and plastic phosphor crystals to the in situ determination of subterranean gamma-ray emitting radionuclides is discussed. Techniques are described for the quantitative measurement of radionuclides in sediments through concentration variations over eight orders of magnitude. Methods are outlined for determining the source of entry of radionuclides into sediments, their direction of travel, their migration rate, and their distance from the point of measurement. Application of the technology to the in situ measurement of radionuclides leaked from underground nuclear fuel waste storage tanks is discussed.

571 LEACHING OF IRRADIATED LWR FUEL PELLETS IN DEIONIZED WATER, SEA BRINE, AND TYPICAL GROUNDWATER. Katayama, Y.B.; Mendel, J.E. (Battelle Northwest Labs., Richland, WA). Transactions of the American Nuclear Society ; 27: 447-448(1977).

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

See CONF-771109--.

FUEL PELLETS;LEACHING;SEAWATER;GROUND WATER; UNDERGROUND STORAGE;UNDERGROUND DISPOSAL; RADIOACTIVE WASTES

572 CHARACTERIZATION OF ACTINIDE-BEARING SEDIMENTS UNDERLYING LIQUID WASTE DISPOSAL FACILITIES AT HANFORD. Price, S.M. (Atlantic Richfield Hanford Co., Richland, Wash. (USA)); Ames, L.L. (Battelle Pacific Northwest Labs., Richland, Wash. (USA)). pp 191-210 of Transuranium nuclides in the environment. Vienna; IAEA (1976).

From Symposium on transuranium nuclides in the environment; San Francisco, Calif., USA (17 - 21 Nov 1975).

Past liquid waste disposal practices at the US Energy Research and Development Administration's Hanford Reservation have included the discharges of solutions containing low concentrations of actinides directly into the ground via structures collectively termed ''trenches''. Characterization of samples from two of these trenches, the 216-Z-9 and the 216-Z-1A(a), has been initiated to determine the present form and migration potential of plutonium stored in sediments which received high salt, acidic waste liquids. Analysis of samples acquired by drilling has revealed that the greatest measured concentration of plutonium, approximately 10^6 $\mu Ci^{239}Pu/l$ of sediment, occurs in both facilities just below the points of release of the waste liquids. This concentration decreases to approximately 10^3 $\mu Ci^{239}Pu/l$ of sediment within the first 2 m of the underlying sediment columns and to approximately 10 $\mu Ci^{239}Pu/l$ of sediment at the maximum depth sampled (9m). Examination of relatively undisturbed sediment cores illustrated two types of Pu occurrence responsible for this distribution. One of these types is composed of Pu particles (>70 wt% PuO_2) added to the disposal site in the same form. This ''particulate'' type was ''filtered out'' within the upper 1 m of the sediment column, accounting for the high concentration of Pu/l of sediment in this region. The second type of Pu (<0.5 wt% PuO_2) was originally disposed of as soluble Pu(IV). This ''nonparticulate'' type penetrated deeper within the sediment profile and was deposited in association with silicate hydrolysis of the sediment fragments.

573 ULTIMATE DISPOSAL: A PLAN FOR ACHIEVEMENT. Bartlett, J.W. (Battelle-Northwest, Richland, WA). pp 44-56 of Waste management '75. Post, R.G. (ed.). Tucson, AZ; University of Arizona (1975).

Four major topics relevant to R and D plans for disposal were: functions of planning, plans development procedures, R and D program procedures, and R and D plans content. Comments on these topics emphasize four major points: plans and their results support decisions on disposal methods; decisions will winnow options on the basis of comprehensive assessments; the R and D plan for disposal will be comprehensive and maintain options; time frame for the R and D program may be about 20 years. Prior and on-going work has provided a good foundation for this planning effort and the content of the plans. The R and D plans are expected to be developed this year and updated periodically.

Environmental Aspects

574 (BNWL--2100(Pt.5), pp 1-27) ENVIRONMENTAL CONTROL TECHNOLOGY. Jun 1977.

Pacific Northwest Laboratory annual report for 1976 to the ERDA Assistant Administrator for Environment and Safety. Part 5. Control technology, overview, safety, and policy analysis.

The objective of the overall Environmental Control Technology Program is to assure that the environmental control capability for each ERDA energy technology is complete, practical, cost effective, and available in a timely manner as the energy source is developed. Program activities are oriented to identifying control technology status and needs for emerging energy systems, then developing methods and equipment for meeting these needs. Progress is reported on studies in support of both nonnuclear and nuclear technologies, with programs in oil shale, coal, energy materials transport, and nuclear fuel cycle analysis. Results are reported from studies on the environmental control technology treatment of oil shale; the assessment of environmental control technologies for commercial coal gasification; transportation safety studies; transportation problems for 1976 to 2000; a safety and economic study of special trains; development of high-level waste shipping cask models; analysis of nuclear fuel cycles; toxic materials in the nuclear fuel cycle; and decontamination and decommissioning of retired contaminated ERDA facilities at Hanford.

575 (BNWL-SA--6311) MONITORING METHODS FOR PARTICULATE AND GASEOUS EFFLUENT FROM WASTE SOLIDIFICATION PROCESSES. Gordon, R.L. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Mar 1977. Contract EY-76-C-06-1830. 17p. (CONF-770656--30). Dep. NTIS, PC A02/MF A01.
From Annual meeting of the Institute of Nuclear Materials Management; Washington, DC, USA (29 Jun 1977).
The objective of this program is to develop reliable and accurate techniques for process monitoring and stack emission monitoring for future commercial waste solidification facilities.

576 (DOE/ET--0029(Vol.1)) ENVIRONMENTAL ASPECTS OF COMMERCIAL RADIOACTIVE WASTE MANAGEMENT. (Battelle Pacific Northwest Labs., Richland, WA (USA)). May 1979. Contract EY-76-C-06-1830. 587p. Dep. NTIS, PC A25/MF A01.
Environmental effects (including accidents) associated with facility construction, operation, decommissioning, and transportation in the management of commercially generated radioactive waste were analyzed for plants and systems assuming a light water power reactor scenario that produces about 10,000 GWe-yr through the year 2050. The following alternative fuel cycle modes or cases that generate post-fission wastes requiring management were analyzed: a once-through option, a fuel reprocessing option for uranium and plutonium recycle, and a fuel reprocessing option for uranium-only recycle. Volume 1 comprises five chapters: introduction; summary of findings; approach to assessment of environmental effects from radioactive waste management; environmental effects related to radioactive management in a once-through fuel cycle; and environmental effects of radioactive waste management associated with an LWR fuel reprocessing plant. (LK)

577 (DOE/ET--0029(Vol.2)) ENVIRONMENTAL ASPECTS OF COMMERCIAL RADIOACTIVE WASTE MANAGEMENT. (Battelle Pacific Northwest Labs., Richland, WA (USA)). May 1979. Contract EY-76-C-06-1830. 492p. Dep. NTIS, PC A21/MF A01.
Volume 2 contains chapters 6 through 10: environmental effects related to radioactive waste management associated with LWR fuel reprocessing - mixed-oxide fuel fabrication plant; environmental effects related to transporting radioactive wastes associated with LWR fuel reprocessing and fabrication; environmental effects related to radioactive waste management associated with LWR fuel reprocessing - retrievable waste storage facility; environmental effects related to geologic isolation of LWR fuel reprocessing wastes; and integrated systems for commercial radioactive waste management. (LK)

578 (DOE/ET--0029(Vol.3)) ENVIRONMENTAL ASPECTS OF COMMERCIAL RADIOACTIVE WASTE MANAGEMENT. (Battelle Pacific Northwest Labs., Richland, WA (USA)). May 1979. Contract EY-76-C-06-1830. 330p. Dep. NTIS, PC A15/MF A01.
Volume 3 contains eight appendices: a reference environment for assessing environmental impacts associated with construction and operation of waste treatment, interim storage and/or final disposition facilities; dose calculations and radiologically related health effects; socioeconomic impact assessments; release/dose factors and dose in 5-year intervals to regional and world wide population from reference integrated systems; resource availability; environmental monitoring; detailed dose results for radionuclide migration groundwater from a waste repository; and annual average dispersion factors for selected release points. (LK)

579 (PNL--2253) ECOLOGY OF THE 200 AREA PLATEAU WASTE MANAGEMENT ENVIRONS: A STATUS REPORT. Rogers, L.E.; Rickard, W.H. (eds.). (Battelle Pacific Northwest Labs., Richland, WA (USA)). Oct 1977. Contract EY-76-C-06-2130. 215p. Dep. NTIS, PC A10/MF A01.
Separate abstracts were prepared for nine sections of this report.

580 (PNL--2253, pp 1.1-1.18) INTRODUCTION. Uresk, V.A.; Rogers, L.E. Oct 1977.
Ecology of the 200 Area plateau waste management environs: a status report.
A general description of ecosystem functioning introduces the need for comprehensive ecological studies as part of the overall waste management operations on the 200 Area plateau. A history of waste management operations and location of 200 Area facilities is provided. An estimate of inventories of radioactive materials in the 200 Area cribs, ponds, and ditches at the end of September 1976 is included.

581 (PNL--2253, pp 2.1-2.5) ABIOTIC COMPONENTS. Rickard, W.H.; Emery, R.M. Oct 1977.
Ecology of the 200 Area plateau waste management environs: a status report.
The climatic regime of the 200 Area plateau consists of cold winters, hot summers and low precipitation amounts. Properties of surface soils and physiochemical properties of ponds and ditches are described.

582 (PNL--2253, pp 3.1-3.40) BIOTIC COMPONENTS. Uresk, D.W.; Fitzner, R.E.; Rogers, L.E.; Rickard, W.H. Oct 1977.
Ecology of the 200 Area plateau waste management environs: a status report.
Representative plant communities are described. The major community is dominated by sagebrush/cheatgrass-sandberg blue grass. Mammal, bird and insect species inhabiting the 200 Area plateau are representative of surrounding regions. Prairie falcons are the only species present possibly threatened with extinction. They do not nest on the plateau but probably forage over the area.

583 (PNL--2253, pp 4.1-4.10) ENERGY FLOW AND MINERAL CYCLING MECHANISMS. Rogers, L.E. Oct 1977.
Ecology of the 200 Area plateau waste management environs: a status report.
Analysis of energy flow patterns and mineral cycling mechanisms provides a first step in identifying major transport pathways away from waste management areas. A preliminary food web pattern is described using results from ongoing and completed food habit studies. Biota possessing the greatest potential for introducing radionuclides into food chains leading to man include deer, rabbits, hares, waterfowl, honeybees and upland game birds and are discussed separately.

584 (PNL--2253, pp 5.1-5.14) RADIONUCLIDE TRANSPORT. Cataldo, D.A.; Paine, D.; Cushing,

C.E.; Emery, R.M.; Vaughan, B.E. Oct 1977.
Ecology of the 200 Area plateau waste management environs: a status report.
The availability of radionuclides to biota is discussed especially with reference to specific elements in local soils. Two annual plant species have received concentrated study. These are cheatgrass and tumbleweed, both important inhabitants of waste burial sites. Little is known concerning the radionuclide dynamics of perennial grasses, forbs, or shrub species. The potential for radionuclide transport by jackrabbits, waterfowl, small mammals, and biota inhabiting pond systems is discussed. Concentration ratios are tabulated.

585 (PNL--2253, pp 6.1-6.13) IMPACT OF RADIOACTIVE WASTE MANAGEMENT OPERATIONS. Paine, D.; Rogers, L.E.; Uresk, D.W. Oct 1977.
Ecology of the 200 Area plateau waste management environs: a status report.
Impact assessment of radioactive waste management operations is considered separately for nonradiological impact on biota, impact on ecosystem structure and function and radiological impact on biota. Localized effects related to facility construction and maintenance activities probably occur but the large expanse of relatively undisturbed surrounding landscape minimizes any overall effects.

586 (PNL--2253, pp 7.1-7.15) MANAGEMENT MODELS. Sauer, R.; Schreckhise, R.G.; Uresk, D.W. Oct 1977.
Ecology of the 200 Area plateau waste management environs: a status report.
Management models are used here as a vehicle to organize the ecological research tasks required for the biotic transport program. Three kinds of models are described: ecosystem level simulation models; radionuclide transport models; and optimization models. The ecosystem simulation model is included here as a coupling agent between climatic variables and waste management practices with the transport and optimization models. Certainly the potential for radionuclide transport depends to a large degree upon the density, type and vigor of plant species inhabiting burial sites. These parameters, in turn, are related to past management practices, precipitation patterns, and successional stages, all predictable using current techniques.

587 (PNL--2253, pp 8.1-8.22) APPLICATIONS TO WASTE MANAGEMENT OPERATIONS. Paine, D.; Uresk, V.; Schreckhise, R.G. Oct 1977.
Ecology of the 200 Area plateau waste management environs: a status report.
Ecological studies of the 200 Area plateau waste management environs have provided preliminary answers to questions concerning the environmental health of associated biota, potential for radionuclide transport through the biotic system and risk to man. More importantly creation of this ecological data base provides visible evidence of environmental expertise so essential for maintenance of continued public confidence in waste management operations.

588 (PNL--2253, pp 9.1-9.17) FUTURE RESEARCH NEEDS. Rogers, L.E.; Cataldo, D.A. Oct 1977.
Ecology of the 200 Area plateau waste management environs: a status report.
The future research needs of the biotic transport program are identified in terms of six research categories. Each category is identified with a starting date, period of emphasis and completion date. Research tasks are identified for each category and scheduled in a milestone chart. Task interrelationships are identified in a flow diagram to facilitate program management.

589 (PNL--2296) DESCRIPTIONS OF REFERENCE LWR FACILITIES FOR ANALYSIS OF NUCLEAR FUEL CYCLES. Schneider, K.J.; Kabele, T.J. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Sep 1979. Contract EY-76-C-06-1830. 528p. Dep. NTIS, PC A28/MF A01.
To contribute to the Department of Energy's identification of needs for improved environmental controls in nuclear fuel cycles, a study was made of a light water reactor system. A reference LWR fuel cycle was defined, and each step in this cycle was characterized by facility description and mainline and effluent treatment process performance. The reference fuel cycle uses fresh uranium in light water reactors. Final treatment and ultimate disposition of waste from the fuel cycle steps were not included, and the waste is assumed to be disposed of by approved but currently undefined means. The characterization of the reference fuel cycle system is intended as basic information for further evaluation of alternative effluent control systems.

590 (PNL--2287) SURVEY OF LWR ENVIRONMENTAL CONTROL TECHNOLOGY PERFORMANCE AND COST. Heeb, C.M.; Aaberg, R.L.; Cole, B.M.; Engel, R.L.; Kennedy, W.E. Jr.; Lewallen, M.A. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Mar 1980. Contract EY-76-C-06-1830. 194p. Dep. NTIS, PC A09/MF A01.
This study attempts to establish a ranking for species that are routinely released to the environment for a projected nuclear power growth scenario. Unlike comparisons made to existing standards, which are subject to frequent revision, the ranking of releases can be used to form a more logical basis for identifying the areas where further development of control technology could be required. This report describes projections of releases for several fuel cycle scenarios, identifies areas where alternative control technologies may be implemented, and discusses the available alternative control technologies. The release factors were used in a computer code system called ENFORM, which calculates the annual release of any species from any part of the LWR nuclear fuel cycle given a projection of installed nuclear generation capacity. This survey of fuel cycle releases was performed for three reprocessing scenarios (stowaway, reprocessing without recycle of Pu and reprocessing with full recycle of U and Pu) for a 100-year period beginning in 1977. The radioactivity releases were ranked on the basis of a relative ranking factor. The relative ranking factor is based on the 100-year summation of the 50-year population dose commitment from an annual release of radioactive effluents. The nonradioactive releases were ranked on the basis of dilution factor. The twenty highest ranking radioactive releases were identified and each of these was analyzed in terms of the basis for calculating the release and a description of the currently employed control method. Alternative control technology is then discussed, along with the available capital and operating cost figures for alternative control methods.

591 (PNL--2652) WASTE ISOLATION SAFETY ASSESSMENT PROGRAM. A BRIEF DESCRIPTION OF

THE THREE-DIMENSIONAL FINITE ELEMENT GROUND-WATER FLOW MODEL ADAPTED FOR WASTE ISOLATION SAFETY ASSESSMENTS. Cole, C.R.; Gupta, S.K. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Aug 1979. Contract EY-76-C-06-1830. 57p. Dep. NTIS, PC A04/MF A01.

Four levels of hydrologic models have been categorized to handle varying complexities and degrees of available input parameters. The first level is for the simplest one-dimensional models having analytical solutions; the second level includes idealized analytic or hybrid analytic models for single aquifer systems with scanty input data; the third level deals with more complex single or quasi-multilayered systems; and the fourth level is for complex multilayered systems. The three-dimensional finite element ground-water model described in this report falls under the fourth level of hydrologic models. This model is capable of simulating single-layered systems having variable thickness or multilayered systems where not only thickness can be varied, but the number of layers can be changed to agree with the vertical geologic section. Supporting programs have been developed to plot grid values, contour maps and three-dimensional graphics of the input data used in simulation as well as the results obtained. At present, the model considers only confined aquifers. The capabilities of the model were demonstrated by using a test case consisting of the multilayered ground-water system beneath Long Island, New York.

592 (PNL--2774) CHARACTERIZATION OF THE HANFORD 300 AREA BURIAL GROUNDS. TASK IV. BIOLOGICAL TRANSPORT. Fitzner, R.E.; Gano, K.A.; Rickard, W.H.; Rogers, L.E. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Oct 1979. Contract EY-76-C-06-1830. 86p. Dep. NTIS, PC A05/MF A01.

The characteristics of radioactive waste burial sites at the 300 area burial grounds on the Department of Energy's Hanford Site, southeastern Washington were studied. The potential vectors of radionuclide transport studied were vegetation and animals. The overall results showed a low potential for uptake and transport of radionuclides from the 300 area sites. However, additional methods to control physical and biological mechanisms may contribute to the effectiveness of waste burial practices. From the results, the Biological Transport task recommended field studies which include reduction of soil erosion and addition of biobarriers to plants and animals. Vegetation plays a major role in reducing soil erosion, and thereby maintaining the backfill over the burial sites. Of the several species found on the 300 area sites, cheatgrass (Bromus tectorum) appears to be the most desirable as a cover. Besides retarding erosion, it has a shallow root system (does not easily penetrate buried material); it has a low affinity for radionuclide uptake; and its tissues are not easily blown away. Small mammals (specifically, mice) appear to have the most potential for radionuclide exposure and uptake. Small mammals were live-trapped within 10 x 10-meter trap grids. Each animal trapped was surgically implanted with a thermoluminescent dosimeter. When the animal was recaptured, the dosimeter was removed and read for exposure. Exposures were reported in milli-Roentgens. The most consistently trapped small mammals were the Great Basin pocket mouse (Perognathus parvus) and the deer mouse (Peromyscus maniculatus). Results from the dosimeter readings showed that some of those animals had higher than background exposures. Biobarriers to animals could be considered as a mechanism to reduce the potential for radionuclide transport.

593 (PNL--2850(Pt.2), pp 10.9-10.10) DECOMMISSIONING AND DECONTAMINATION (300 AREA BURIAL GROUND) STUDIES. Fitzner, R.E. Feb 1979.

Pacific Northwest Laboratory annual report for 1978 to the DOE Assistant Secretary for Environment. Part 2. Ecological Sciences.

Progress is reported in the decommissioning and decontamination of retired Hanford facilities and planning the future use of surrounding landscapes. Ecological studies were developed to characterize the 300 Area waste burial grounds and to determine the potential for biotic transport of buried contaminated wastes away from managed facilities.

594 (PNL--2850(Pt.5), pp 2.0-2.52) ENVIRONMENTAL CONTROL ENGINEERING. Feb 1979.

Pacific Northwest Laboratory annual report for 1978 to the DOE Assistant Secretary for Environment. Part 5. Environmental assessment, control, health and safety.

Progress is reported on the following projects: assessment of environmental technology for coal gas separation; energy material transport, 1977 to 2000; dry/wet cooling towers; liquefied natural gas safety and control; burning of oil spills; liquified petroleum (LPG) research assessment; treatment of oil shale; geothermal liquid waste disposal; compressed air energy storage; energy-conserving industrial waste treatment; analysis of nuclear fuel cycles; transportation safety study; decommissioning of retired facilities at Hanford; planning; characterization of Hanford 300 Area burial grounds; decontamination and decommissioning of Hanford facilities: technology; assistance for nationwide decommissioning planning for DOE nuclear facilities; and, asphalt emulsion sealing of uranium tailings. (JGB)

595 (PNL--2970) GETOUT: A COMPUTER PROGRAM FOR PREDICTING RADIONUCLIDE DECAY CHAIN TRANSPORT THROUGH GEOLOGIC MEDIA. DeMier, W.V.; Cloninger, M.O.; Burkholder, H.C.; Liddell, P.J. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Aug 1979. Contract EY-76-C-06-1830. 342p. Dep. NTIS, PC A15/MF A01.

A guide to the use of the GETOUT code is presented. The code models the migration of radionuclides from an underground source to a surface body of water. Output from the code comprises: (1) a nuclide discharge summary file in a format suitable for input to an existing biosphere model, and (2) printed reports showing discharge profiles for selected nuclides by four release modes. The guide provides a general description of the GETOUT system, program listings, sample output, and instructions, with examples for executing the code.

596 (PNL--3129) ENVIRONMENTAL CONTROL ASPECTS FOR FABRICATION, REPROCESSING AND WASTE DISPOSAL OF ALTERNATIVE LWR AND LMFBR FUELS. Nolan, A.M.; Lewallen, M.A.; McNair, G.W. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Nov 1979. Contract EY-76-C-06-1830. 94p. Dep. NTIS, PC A05/MF A01.

Environmental control aspects of alternative fuel cycles have been analyzed by evaluating fabrication, reprocessing, and waste disposal operations. Various indices have been used to assess potential environmental control requirements. For the fabrication and reprocessing operations, 50-year dose commitments were used. Waste disposal was evaluated by comparing projected nuclide concentrations in ground water at various time periods with maximum permissible concentrations (MPCs). Three different fabrication plants were analyzed: a fuel fabrication plant (FFP) to produce low-activity uranium and uranium-thorium fuel rods; a plutonium fuel refabrication plant (PFRFP) to produce plutonium-uranium and plutonium-thorium fuel rods; and a uranium fuel refabrication plant (UFRFP) to produce fuel rods containing the high-activity isotopes ^{232}U and ^{233}U. Each plant's dose commitments are discussed separately. Source terms for the analysis of effluents from the fuel reprocessing plant (FRP) were calculated using the fuel burnup codes LEOPARD, CINDER and ORIGEN. Effluent quantities are estimated for each fuel type. Bedded salt was chosen for the waste repository analysis. The repository site is modeled on the Waste Isolation Pilot Program site in New Mexico. Wastes assumed to be stored in the repository include high-level vitrified waste from the FRP, packaged fuel residue from the FRP, and transuranic (TRU) contaminated wastes from the FFP, PFRFP, and UFRFP. The potential environmental significance was determined by estimating the ground-water concentrations of the various nuclides over a time span of a million years. The MPC for each nuclide was used along with the estimated ground-water concentration to generate a biohazard index for the comparison among fuel compositions.

597 (PNL--3253) ENVIRONMENTAL CONTROL TECHNOLOGY FOR MINING, MILLING, AND REFINING THORIUM. Weakley, S.A.; Blahnik, D.E.; Young, J.K.; Bloomster, C.H. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Feb 1980. Contract AC06-76RL01830. 253p. NTIS, PC A12/MF A01.

The purpose of this report is to evaluate, in terms of cost and effectiveness, the various environmental control technologies that would be used to control the radioactive wastes generated in the mining, milling, and refining of thorium from domestic resources. The technologies, in order to be considered for study, had to reduce the radioactivity in the waste streams to meet Atomic Energy Commission (10 CFR 20) standards for natural thorium's maximum permissible concentration (MPC) in air and water. Further regulatory standards or licensing requirements, either federal, state, or local, were not examined. The availability and cost of producing thorium from domestic resources is addressed in a companion volume. The objectives of this study were: (1) to identify the major waste streams generated during the mining, milling, and refining of reactor-grade thorium oxide from domestic resources; and (2) to determine the cost and levels of control of existing and advanced environmental control technologies for these waste streams. Six potential domestic deposits of thorium oxide, in addition to stockpiled thorium sludges, are discussed in this report. A summary of the location and characteristics of the potential domestic thorium resources and the mining, milling, and refining processes that will be needed to produce reactor-grade thorium oxide is presented in Section 2. The wastes from existing and potential domestic thorium oxide mines, mills, and refineries are identified in Section 3. Section 3 also presents the state-of-the-art technology and the costs associated with controlling the wastes from the mines, mills, and refineries. In Section 4, the available environmental control technologies for mines, mills, and refineries are assessed. Section 5 presents the cost and effectiveness estimates for the various environmental control technologies applicable to the mine, mill, and refinery for each domestic resource.

598 (PNL--3300(Pt.5), pp 43-49) ANALYSIS OF NUCLEAR FUEL CYCLES. Feb 1980. Dep. NTIS, PC A07/MF A01.

Pacific Northwest Laboratory annual report for 1979 to the DOE Assistant Secretary for Environment. Part 5. Environmental assessment, control, health, and safety.

The operation of nuclear fuel cycle facilities will introduce noxious materials, both radiological and chemical, into the environment through routine discharges of both liquid and airborne effluents. The environmental control implications of continuing to develop existing nuclear fuel cycles and implementing new fuel cycles must be systematically determined so technologies that control or eliminate the discharge of noxious materials to the environment can be quickly developed and demonstrated. The objective of this program is to identify areas in developing nuclear fuel cycles where (1) inadequate consideration is being given to environmental controls, (2) inconsistencies and conflicts exist in environmental policy, and (3) environmental control improvements can be justified on a cost/risk/benefit basis to ensure that funds are not expended for control in instances where neither the potential effects nor public concerns warrant such expenditures.

599 (PNL--3302) RELATIONSHIPS OF DISPERSIVE MASS TRANSPORT AND STOCHASTIC CONVECTIVE FLOW THROUGH HYDROLOGIC SYSTEMS. Simmons, C.S. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jan 1981. Contract AC06-76RL01830. 149p. NTIS, PC A07/MF A01.

Uncertainty in water flow velocity appears to be a major factor in determining the magnitude of contaminant dispersion expected in a ground water system. This report discusses some concepts and mathematical methods relating dispersive contaminant transport to stochastic aspects of ground water flow. The theory developed should not be construed as absolutely rigorous mathematics, but is presented with the intention of clarifying the physical concepts.

600 (PNL--3304) MONITORING AND PHYSICAL CHARACTERIZATION OF UNSATURATED ZONE TRANSPORT. LABORATORY ANALYSIS. Gee, G.W.; Campbell, A.C. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Mar 1980. Contract EY-76-C-06-1830. 67p. Dep. NTIS, PC A04/MF A01.

Batch Kd screening tests on Hanford sediments indicated that anionic radionuclide mobility is dependent upon specie and concentration. Technetium, as TcO_4^-, is not sorbed to any extent and will move as rapidly as water. Iodine, as I^-, will also move at high concentrations (>5 ppB) but is retarded at very low concentrations (<1 ppB). Neutral species such as tritium and complexed nuclides such as cobalt-60 (EDTA) are only slightly retarded while uncomplexed cobalt-60 is strongly bound to the sediments under the conditions studied. Strontium sorption is controlled by ion exchange-like reactions and sorption is predictable from a knowledge of soil and solution characteristics. Cesium is strongly bound to the Hanford sediments under typical conditions and appears to be sorbed independent of salt concentration for the synthetic ground waters tested. The sorption of relatively mobile nuclides such as tritium, cobalt-EDTA, and technetium could be measured with greater precision using the unsaturated column tests than with batch methods. Little or no difference in sorption was observed when the flow rates were changed by more than an order of magnitude (2 to 50 cm/day). Changes in saturation percentage of the soil, from 56 to 31%, also had only a minor effect on nuclide sorption. For layered sediments, the effective sorption of a layered sequence is controlled by the sorption characteristics of the most sorbing layer.

601 (PNL--3363) TOPICAL REPORT ON RELEASE SCENARIO ANALYSIS OF LONG-TERM MANAGEMENT OF HIGH-LEVEL DEFENSE WASTE AT THE HANFORD SITE. Wallace, R.W.; Landstrom, D.K.; Blair, S.C.; Howes, B.W.; Robkin, M.A.; Benson, G.L.; Reisenauer, A.E.; Walters, W.H.; Zimmerman, M.G. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Nov 1980. Contract AC06-76RL01830. 223p. NTIS, PC A10/MF A01.

Potential release scenarios for the defense high-level waste (HLW) on the Hanford Site are presented. Presented in this report are the three components necessary for evaluating the various alternatives under consideration for long-term management of Hanford defense HLW: identification of scenarios and events which might directly or indirectly disrupt radionuclide containment barriers; geotransport calculations of waste migration through the site media; and consequence (dose) analyses based on groundwater and air pathways calculations. The scenarios described in this report provide the necessary parameters for radionuclide transport and consequence analysis. Scenarios are categorized as either bounding or nonbounding. Bounding scenarios consider worst case or what if situations where an actual and significant release of waste material to the environment would happen if the scenario were to occur. Bounding scenarios include both near-term and long-term scenarios. Near-term scenarios are events which occur at 100 years from 1990. Long term scenarios are potential events considered to occur at 1000 and 10,000 years from 1990. Nonbounding scenarios consider events which result in insignificant releases or no release at all to the environment. Three release mechanisms are described in this report: (1) direct exposure of waste to the biosphere by a defined sequence of events (scenario) such as human intrusion by drilling; (2) radionuclides contacting an unconfined aquifer through downward percolation of groundwater or a rising water table; and (3) cataclysmic or explosive release of radionuclides by such mechanisms as meteorite impact, fire and explosion, criticality, or seismic events. Scenarios in this report present ways in which these release mechanisms could occur at a waste management facility. The scenarios are applied to the two in-tank waste management alternatives: in-situ disposal and continued present action.

602 (PNL--3414) POTENTIAL INFLUENCE OF ORGANIC COMPOUNDS ON THE TRANSPORT OF RADIONUCLIDES FROM A GEOLOGIC REPOSITORY. ASSESSMENT OF EFFECTIVENESS OF GEOLOGIC ISOLATION SYSTEMS. Silviera, D.J. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Mar 1981. Contract AC06-76RL01830. 42p. NTIS, PC A04/MF A01.

This study identifies organic compounds that may be present in a repository and outlines plausible interactions and mechanisms that may influence the forms and chemical behavior of these compounds. A review of the literature indicates that large quantities of organic radioactive wastes are generated by the nuclear industry and if placed in a repository could increase or decrease the leach rate and sorption characteristics of waste radionuclides. The association of radionuclides with organic matter can render the nuclides soluble or insoluble depending on the particular nuclide and such parameters as the pH, Eh, and temperature of the hydrogeologic system as well as the properties of the organic compounds themselves. 44 references.

603 (PNL--3473) WASTE/ROCK INTERACTIONS TECHNOLOGY PROGRAM. STATUS REPORT ON LWR SPENT-FUEL LEACH TESTS. Katayama, Y.B.; Bradley, D.J.; Harvey, C.O. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Nov 1980. Contract AC06-76RL01830. 51p. NTIS, PC A04/MF A01.

Spent fuels with burnups of 3000, 28,000 and 54,500 MWd/MTU have been leach tested at 25°C. Three leach-test procedures (Paige, IAEA and static) were used. IAEA and static tests were conducted in five different solutions: deionized water, sodium bicarbonate, sodium chloride, calcium chloride and Waste Isolation Pilot Plant B brine solutions. Elemental leach data are reported based on the release of $^{90}Sr+^{90}Y$, ^{106}Ru, ^{137}Cs, ^{144}Ce, ^{154}Eu, $^{239+240}Pu$, ^{125}Sb, ^{244}Cm, ^{129}I, ^{99}Tc, and total uranium. This is the first report on ^{129}I and ^{99}Tc from spent fuel. Termination of the Paige test showed that the plateout (radionuclide adsorption) on the test apparatus had negligible effect on the leach rate of cesium and plutonium, but a major (up to a factor of 50 times) effect on the curium leach rate. Three-hundred additional days of leach testing by the IAEA procedure, from 467 to 769 d, showed a continuation of the leaching trends observed during the first 467 d. Results from the first two static leach test series, 2 and 8 d, gave the ^{129}I and ^{99}Tc release numbers.

604 (PNL--3518) GEOCHEMICAL MODELING OF THE NUCLEAR-WASTE REPOSITORY SYSTEM. A STATUS REPORT. Deutsch, W.J. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Dec 1980. Contract AC06-76RL01830. 31p. NTIS, PC A03/MF A01.

The primary objective of the geochemical modeling task is to develop an understanding of the waste-repository geochemical system and provide a valuable tool for estimating future states of that system. There currently exists a variety of computer codes which can be used in geochemical modeling studies. Some available codes contain the framework for simulating a natural chemical system and estimating, within limits, the response of that system to environmental changes. By data-base enhancement and code development, this modeling technique can be even more usefully applied to a nuclear-waste repository. In particular, thermodynamic data on elements not presently in the data base but identified as being of particular hazard in the waste-repository system, need to be incorporated into the code to estimate the near-field as well as the far-field reactions during a hypothetical breach. A reaction-path-simulation code, which estimates the products of specific rock/water reactions, has been tested using basalt and ground water. Results show that the mass-transfer capabilities of the code will be useful in chemical-evolution studies and scenario analyses. The purpose of this report is to explain the status of geochemical modeling as it currently applies to the chemical system of a hypothetical nuclear-waste repository in basalt and to present the plan proposed for further developmet and application.

605 (PNL--3528) STATUS REPORT ON STRS:

SORPTION INFORMATION RETRIEVAL SYSTEM. Hostetler, D.D.; Serne, R.J.; Baldwin, A.J.; Petrie, G.M. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Nov 1980. Contract AC06-76RL01830. 77p. NTIS, PC A05/MF A01.

Two major uses were identified for the Sorption Information Retrieval System: (1) to aid geochemists in the elucidation of sorption mechanisms; and (2) to aid safety assessment modelers in selection of Kds for any given scenario. Other benefits such as providing an auditable vehicle for the Kd selection were also discussed.

606 (PNL--3574) GEOCHEMICAL MODELING: A REVIEW. Jenne, E.A. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jun 1981. Contract AC06-76RL01830. 58p. NTIS, PC A04/MF A01. Order Number DE81028969.

Two general families of geochemical models presently exist. The ion speciation-solubility group of geochemical models contain submodels to first calculate a distribution of aqueous species and to secondly test the hypothesis that the water is near equilibrium with particular solid phases. These models may or may not calculate the adsorption of dissolved constituents and simulate the dissolution and precipitation (mass transfer) of solid phases. Another family of geochemical models, the reaction path models, simulates the stepwise precipitation of solid phases as a result of reacting specified amounts of water and rock. Reaction path models first perform an aqueous speciation of the dissolved constituents of the water, test solubility hypotheses, then perform the reaction path modeling. Certain improvements in the present versions of these models would enhance their value and usefulness to applications in nuclear-waste isolation, etc. Mass-transfer calculations of limited extent are certainly within the capabilities of state-of-the-art models. However, the reaction path models require an expansion of their thermodynamic data bases and systematic validation before they are generally accepted.

607 (PNL--3614) SOLUBILITY EFFECTS IN WASTE-GLASS/DEMINERALIZED-WATER SYSTEMS. Fullam, H.T. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jun 1981. Contract AC06-76RL01830. 66p. NTIS, PC A04/MF A01. Order Number DE81026562.

Aqueous systems involving demineralized water and four glass compositions (including standins for actinides and fission products) at temperatures of up to 150°C were studied. Two methods were used to measure the solubility of glass components in demineralized water. One method involved approaching equilibrium from subsaturation, while the second method involved approaching equilibrium from supersaturation. The aqueous solutions were analyzed by induction-coupled plasma spectrometry (ICP). Uranium was determined using a Scintrex U-A3 uranium analyzer and zinc and cesium were determined by atomic absorption. The system that results when a waste glass is contacted with demineralized water is a complex one. The two methods used to determine the solubility limits gave very different results, with the supersaturation method yielding much higher solution concentrations than the subsaturation method for most of the elements present in the waste glasses. The results show that it is impossible to assign solubility limits to the various glass components without thoroughly describing the glass-water systems. This includes not only defining the glass type and solution temperature, but also the glass surface area-to-water volume ratio (S/V) of the system and the complete thermal history of the system. 21 figures, 22 tables. (DLC)

608 (PNL--3630) ANALYTICAL SOLUTIONS OF MOISTURE FLOW EQUATIONS AND THEIR NUMERICAL EVALUATION. Gibbs, A.G. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Apr 1981. Contract AC06-76RL01830. 43p. NTIS, PC A03/MF A01.

The role of analytical solutions of idealized moisture flow problems is discussed. Some different formulations of the moisture flow problem are reviewed. A number of different analytical solutions are summarized, including the case of idealized coupled moisture and heat flow. The evaluation of special functions which commonly arise in analytical solutions is discussed, including some pitfalls in the evaluation of expressions involving combinations of special functions. Finally, perturbation theory methods are summarized which can be used to obtain good approximate analytical solutions to problems which are too complicated to solve exactly, but which are close to an analytically solvable problem.

609 (PNL--3700(Pt.2), pp 75-78) RADIOECOLOGY OF NUCLEAR FUEL CYCLES. Schreckhise, R.G.; Cadwell, L.L.; Emery, R.M. Feb 1981. NTIS, PC A09/MF A01.

Pacific Northwest Laboratory annual report for 1980 to the DOE Assistant Secretary for Environment. Part 2. Ecological sciences.

This study provides information to help assess the environmental impacts and certain potential human hazards associated with nuclear fuel cycles. A data base is being developed to define and quantify biological transport routes which will permit credible predictions and assessment of routine and potential large-scale releases of radionuclides and other toxic materials. Information obtained from existing storage and disposal sites will provide a meaningful radioecological perspective with which to improve the effectiveness of waste management practices. This paper focuses on terrestrial and aquatic radioecology of waste management areas and biotic transport parameters.

610 (PNL--3700(Pt.5), pp 39-43) ANALYSIS OF NUCLEAR FUEL CYCLES. Feb 1981. NTIS, PC A05/MF A01.

Pacific Northwest Laboratory annual report for 1980 to the DOE Assistant Secretary for Environment. Part 5. Environmental assessment, control, health and safety.

The environmental control implications of continuing to develop existing nuclear fuel cycles and implementing new fuel cycles are systematically determined so technologies that control or eliminate the discharge of noxious materials can be developed and demonstrated. The objective of this program is to identify areas in developing nuclear fuel cycles where inadequate consideration is being given to environmental controls and areas where environmental control improvements can be justified on a cost/risk/benefit basis. Included are overviews of Gas-Cooled Fast Breeder Reactor, Light Water Reactor Improvements, and Liquid Metal Fast Breeder Reactors, and an overview of Thorium/Uranium Environmental Control Technology. (DLS)

611 (PNL--3789) ASSESSMENT OF EFFECTIVENESS OF GEOLOGIC ISOLATION SYSTEMS: GEOSTATISTICAL MODELING OF PORE VELOCITY. Devary, J.L.; Doctor, P.G. (Battelle Pacific Northwest

Labs., Richland, WA (USA)). Jun 1981. Contract AC06-76RL01830. 45p. NTIS, PC A03/MF A01. Order Number DE81023669.

A significant part of evaluating a geologic formation as a nuclear waste repository involves the modeling of contaminant transport in the surrounding media in the event the repository is breached. The commonly used contaminant transport models are deterministic. However, the spatial variability of hydrologic field parameters introduces uncertainties into contaminant transport predictions. This paper discusses the application of geostatistical techniques to the modeling of spatially varying hydrologic field parameters required as input to contaminant transport analyses. Kriging estimation techniques were applied to Hanford Reservation field data to calculate hydraulic conductivity and the ground-water potential gradients. These quantities were statistically combined to estimate the groundwater pore velocity and to characterize the pore velocity estimation error. Combining geostatistical modeling techniques with product error propagation techniques results in an effective stochastic characterization of groundwater pore velocity, a hydrologic parameter required for contaminant transport analyses.

612 (PNL--3817) USE OF GEOHYDROLOGIC RESPONSE FUNCTIONS IN THE ASSESSMENT OF DEEP NUCLEAR WASTE REPOSITORIES: ASSESSMENT OF EFFECTIVENESS OF GEOLOGIC ISOLATION SYSTEMS. Nelson, R.W. (Battelle Pacific Northwest Labs., Richland, WA (USA)). May 1981. Contract AC06-76RL01830. 53p. NTIS, PC A04/MF A01.

Geohydrologic Response Functions (GRFs) interrelate the three vital factors needed in the repository decision-making process: the quantity, arrival time, and location of contamination reaching the biosphere. GRFs further focus attention upon two related and additive parameters: the initial delay time and delay spread time. After the principal site selection, the GRFs may be applied to obtain more detailed performance evaluations concerning specific nuclides or waste components, specific results for various types of accidental release, and effects of a variety of contaminant source terms or leach models. The response functions may be applied to consider contaminant reductions and delays through material sorption as well as through a variety of other interactions and effects.

613 (PNL--3877) SIMULATION OF WATER FLOW AND RETENTION IN EARTHEN-COVER MATERIALS OVERLYING URANIUM MILL TAILINGS. Simmons, C.S.; Gee, G.W. (Pacific Northwest Lab., Richland, WA (USA)). Sep 1981. Contract AC06-76RL01830. 102p. (UMT--0203). NTIS, PC A06/MF A01. Order Number DE82000239.

The water retention characteristics of a multilayer earthen cover for uranium mill tailings were simulated under arid weather conditions common to Grand Junction, Colorado. The multilayer system described in this report consists of a layer of wet clay/gravel (radon barrier), which is separated from a surface covering of fill soil by a washed rock material used as a capillary barrier. The capillary barrier is designed to prevent the upward migration of water and salt from the tailings to the soil surface and subsequent loss of water from the wet clay. The flow model, UNSATV, described in this report uses hydraulic properties of the layered materials and historical climatic data for two years (1976 and 1979) to simulate long-term hydrologic response of the multilayer system. Application of this model to simulate the processes of infiltration, evaporation and drainage is described in detail. Simulations over a trial period of one relatively wet and two dry years indicated that the clay-gravel layer remained near saturation, and hence, that the layer was an effective radon barrier. Estimates show that the clay-gravel layer would not dry out (i.e., revert to drying dominated by isothermal vapor-flow conditions) for at least 20 years, provided that the modeled dry-climate period continues.

614 (PNL--3927) EFFECT OF ORGANIC COMPLEXANTS ON THE MOBILITY OF LOW-LEVEL WASTE RADIONUCLIDES IN SOILS: STATUS REPORT. Swanson, J.L. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Sep 1981. Contract AC06-76RL01830. 33p. NTIS, PC A03/MF A01. Order Number DE81030409.

The effects of some organic complexants on the sorption by soil of some elements typical of those present in low-level wastes are being evaluated and procedures are being developed to efficiently obtain valid sorption data. Data have been obtained with Hanford soil and the elements europium, nickel, cobalt, cesium, and strontium. Complexants studied to date include ethylenediaminetetraacetic acid (EDTA), diethylenetriaminepentaacetic acid (DTPA), and humic acid. The sorption of cesium and strontium has been found to be affected very little by EDTA or DTPA, as expected. However, these complexants have been found to greatly reduce the sorption of europium, nickel, or cobalt by the soil. The Eu/EDTA system was found to be well behaved, and the effect of the complexant on the sorption by soil was quantified. With nickel and cobalt, however, kinetic problems have been encountered, and the effects of complexants on the sorption by soil have not yet been quantified. The problems encountered in the nickel and cobalt systems have further emphasized the need for the development of methods to assure the general validity of the data obtained in the experiments. Series of experimental results that looked very good within themselves were found by additional procedures to be artifacts of the conditions used. One procedure that has been of great value in identifying invalid data is that of contacting the equilibrated solution from a soil contact with a second batch of soil. Unless the sorption coefficient in the second contact is equivalent to that in the first contact, the data are likely invalid. Efforts are continuing to develop procedures that will allow generally valid data to be obtained in an efficient manner and to employ such procedures to obtain data in a variety of systems pertinent to low-level waste disposal.

615 (PNL--3971) ACTINIDE LEACHING FROM WASTE GLASS: AIR-EQUILIBRATED VERSUS DEAERATED CONDITIONS. Peters, R.D.; Diamond, H. (Pacific Northwest Lab., Richland, WA (USA); Argonne National Lab., IL (USA)). Oct 1981. Contract AC06-76RL01830. 42p. NTIS, PC A03/MF A01. Order Number DE82003098.

Leach tests were conducted in aerated and deaerated solutions using glass containing ^{239}Pu, ^{237}Np and ^{238}U, at temperatures of 25 and 75°C and in deionized water, 0.03M $NaHCO_3$ and WIPP 3 salt brine for periods up to 341 days. Neptunium leaching was decreased by factors of 10 to 100 (depending on leach time) in the deaerated solutions at 75°C. Plutonium leaching decreased by factors of 3 to 5 due to deaeration, but only in the deionized water leachate at 25°C. Uranium leaching in salt brine and deionized water at 25°C was decreased by factors of 2 to 5 in deaerated solutions. Time and temperature dependencies were also observed for the leaching of the actinides during the course of this work. After the first leach interval (2 days), the time dependent release curve for Pu was essentially flat or decreasing under all conditions, and maximum Pu solution concentration (at 25°C), as implied by release in aerated leachate, agrees with independent solubility data. The low ^{239}Pu releases observed in leach solutions are consistent with accumulation of ^{239}Pu on the leached glass surface. The amounts of uranium and neptunium leached increased with time under most conditions. For Pu leaching, temperature has a small effect in deaerated leachates and negative effect in aerated leachates. Neptunium leaching generally increase with temperature under aerated conditions, but not in proportion to increases of matrix element leaching. In deaerated leachates, Np leaching decreases with temperature. Uranium leaching increases with temperature under aerated and deaerated conditions but not in proportion to matrix element increases. 4 figures, 6 tables.

616 (PNL--3989) RADON DIFFUSION THROUGH MULTILAYER EARTHEN COVERS: MODELS AND SIMULATIONS. Meyer, D.W.; Oster, C.A.; Nelson, R.W.; Gee, G.W. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Sep 1981. Contract AC06-76RL01830. 67p. NTIS, PC A04/MF A01. Order Number DE82001420.

A capability to model and analyze the fundamental interactions that influence the diffusion of radon gas through uranium mill tailings and cover systems has been investigated. The purpose of this study is to develop the theoretical basis for modeling radon diffusion and to develop an understanding of the fundamental interactions that influence radon diffusion. This study develops the theoretical basis for modeling radon diffusion in one, two and three dimensions. The theory has been incorporated into three computer models that are used to analyze several tailings and cover configurations. This report contains a discussion of the theoretical basis for modeling radon diffusion, a discussion of the computer models used to analyze uranium mill tailings and multilayered cover systems, and presents the results that have been obtained.

617 (PNL--4100-Pt.2, pp 91-95) RADIOECOLOGY OF NUCLEAR FUEL CYCLES. Cadwell, L.L. Feb 1982. NTIS, PC A08/MF A01.

Annual Report for 1981 to the DOE Office of the Assistant Secretary for Environmental Protection, Safety, and Emergency Preparedness. Part 2. Ecological Sciences.

This study provides information to help assess the environmental impacts and certain potential human hazards associated with nuclear fuel cycles. A data base is being developed to define and quantify biological transport routes, which will permit credible predictions and assessment of routine and potential large-scale releases of radionuclides and other toxic materials. These data, used in assessment models, will increase the accuracy of estimating radiation doses to man and other life forms. Results will provide information to determine if waste management procedures on the Hanford site have caused ecological perturbations, and, if so, to determine the source, nature and magnitude of such disturbances.

618 (PNL--4178) BEHAVIOR OF REDUCED ^{99}TC AND ^{99}TC ORGANIC COMPLEXES ON HANFORD SOIL. Franz, J.A.; Martin, L.Y.; Wiggins, D.J. (Battelle Columbus Labs., OH (USA)). Feb 1982. Contract AC06-76RL01830. 28p. NTIS, PC A03/MF A01. Order Number DE82008986.

Both synthetically and naturally derived organic complexing agents in soil have been found in other studies to accelerate the rate of migration of specific radionuclides. In an effort to aid in the development of comprehensive plans for the disposal of low-level waste, the effect of organic complexing agents on the transport properties of ^{99}Tc, a long-lived constituent of radioactive waste, was examined. The effect of ethylenediaminetetraacetic acid (EDTA), diethylenetriaminepentaacetic acid (DTPA), and sodium citrate on the mobility of reduced Tc species was examined from two aspects: first, by techniques of desorption from soil in which reduced, sorbed Tc was exposed to solutions of organic ligands; and second, by exposure of synthetic Tc-organic complexes to soils. Only a slight desorption of reduced Tc from Hanford soil by EDTA was found to occur in 10-day desorption tests. However, when synthetic EDTA and DTPA complexes of reduced Tc were exposed to soil under similar conditions, only a slight amount of Tc was sorbed by the soil. This comparison clearly shows that at least one type of test did not reach equilibrium in the 10-day time period allowed. Although the effects of these complexants at equilibrium cannot be quantified from these data, it is obvious that they can have large effects on the mobility of reduced Tc. Desorption tests of reduced Tc from Hanford soil by citrate solutions showed definite indications of desorption only at a relatively high (0.01 M) citrate concentrations. Results with synthetic Tc(IV) and Tc(V) citrate complexes indicate the Tc(V) citrate complex to be strongly sorbed, while the Tc(IV) citrate complex was sorbed only slightly.

619 (PNL-SA--6957, pp 189-193) THEORETICAL AND EXPERIMENTAL EVALUATION OF WASTE TRANSPORT IN SELECTED ROCKS. Apps, J.A.; Benson, L.V.; Lucas, J.; Mathur, A.K.; Tsao, L. (Univ. of California, Berkeley). 1977.

From Contractor information meeting; Seattle, WA, USA (20 Sep 1977).

During fiscal year 1977, the following subtasks were performed: 1. Thermodynamic data were tabulated for those aqueous complexes and solid phases of plutonium, neptunium, americium, and curium likely to form in the natural environment. 2. Eh-pH diagrams were computed and drafted for plutonium, neptunium, americium and curium at 25°C and one atmosphere. 3. The literature on distribution coefficients of plutonium, neptunium, americium, and curium was reviewed. 4. Preliminary considerations were determined for an experimental method of measuring radionuclide transport in water-saturated rocks. 5. The transport mechanisms of radionuclides in water-saturated rocks were reviewed. 6. A computer simulation was attempted of mass transfer involving actinides in water-saturated rocks. Progress in each task is reported. Subtasks 1, 2, 3, and 4 are complete. Progress made in subtask 5 is an initial theoretical survey to define the conditions needed to characterize the transport of radionuclides in rocks.

620 (PNL-SA--6957, pp 309-342) TRANSPORT PROPERTIES OF NUCLEAR WASTES IN GEOLOGIC MEDIA. Seitz, M.G.; Rickert, P.; Fried, S.; Friedman, A.M.; Steindler, M. (Argonne National Lab., IL). 1977.

From Contractor information meeting; Seattle, WA, USA (20 Sep 1977).

Laboratory experiments were performed with

Cs, Pu, Np, and Am to examine the migratory characteristics of long-lived radionuclides that could be mobilized by groundwaters infiltrating a nuclear waste repository and the surrounding geologic body. In column infiltration experiments, the positions of peak concentrations of Cs in chalk or shale columns, Pu and Am in limestone, sandstone, or tuff and neptunium in a limestone column did not move when the columns were infiltrated with water. However, fractions of all of the nuclides were seen downstream from the peaks, indicating a large dispersion in the relative migration rates of the trace elements in the lithic materials studied. Static absorption experiments showed that plutonium and americium are strongly absorbed from solution by common rocks and that their migration relative to groundwater flow is thereby retarded. Reaction rates of these dissolved elements with rocks were found to vary considerably in different rock-element systems. Following a sorption step in batch experiments with granulated basalt and Am bearing water, Pu and Am were desorbed from rock and repartitioned between rock and solution to an extent comparable to their distribution during absorption. In contrast, when tablets of various rocks were allowed to dry between absorption and desorption tests, Pu and Am were not generally desorbed from the tablets.In batch experiments with Pu and Am-bearing water and granulated basalt of several different particle sizes, the partitioning of Am and Pu did not correlate with the calculated area of the fracture surfaces nor did the partitioning remain constant (as did the measured surface area). Partitioning is concluded to be a bulk phenomenon with complete penetration of 30 to 40 mesh and smaller particles. 9 tables, 4 figs.

621 (PNL-SA--7142) APPLICATION OF BIOLOGICAL BARRIERS IN MAINTAINING THE INTEGRITY OF RADIOACTIVITY IN SHALLOW BURIAL GROUNDS. Cline, J.F. (Pacific Northwest Lab., Richland, WA (USA)). May 1979. Contract AC06-76RL01830. 28p. (CONF-790728--8). NTIS, PC A03/MF A01. Order Number DE82007090.

From 24. annual meeting of the Health Physics Society; Philadelphia, PA, USA (7 Jul 1979).

Stabilization of a shallow burial site requires some means of keeping buried radioactive wastes in place and preventing the movement of radioactive elements into the biosphere by various vectors present in the soil covering the burial site. By placing a barrier between the surface of the soil and the buried wastes, it would be possible to isolate the wastes from the biosphere and eliminate the movement of radioactive elements into the environment. An effective biobarrier would make it possible to grow plants over the buried wastes regardless of rooting habits; the plants would stabilize the surface soil, prevent wind erosion, and transpire soil water back into the air, thus preventing it from percolating downward through the buried wastes. This report summarizes the finding of a study undertaken to determine the effectiveness of natural cobblestones as a long-term biobarrier. In the initial field study, we investigated whether a thick layer of cobblestones would prevent plant roots and burrowing animals from reaching contaminated materials and transferring radionuclides to the soil surface. In a subsequent greenhouse study, three modifications of the cobblestone barrier were tested, including the addition of another layer of stones, one of asphalt, and one of a root toxin. These data show that cobblestone can be effective as a barrier to burrowing animals and insects, but not totally effective as a barrier to plant roots. Because of variable weather patterns at Hanford, five to six year studies are recommended for further studies on the effectiveness of different materials as biobarriers to radioactive substances. Stone size appeared to affect the plants' rate of root growth since root growth slowed in the air spaces between stones. Root toxin was 100% effective as a means of keeping roots out of the buried waste; this method could be used as a barrier modification where no plant cover is needed.

622 (PNL-SA--7245) PRELIMINARY RESULTS ON COMPARISON OF ADSORPTION-DESORPTION METHODS AND STATISTICAL TECHNIQUES TO GENERATE KD PREDICTOR EQUATIONS. Serne, R.J.; Rai, D.; Relyea, J.F. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 1979. Contract EY-76-C-06-1830. 12p. Dep. NTIS, PC A02/MF A01.

The radioactive waste isolation safety assessment program (WISAP) is being performed for the United States Department of Energy. A large bank of data have been collected under this program. A part of the program is devoted to the evaluation of radionuclide-geomedia interactions in order to determine the behavior of different elements in the environment. This paper introduces and briefly summarizes data collected thus far on methods of determining adsorption coefficients, thermodynamic predictions of stable solid and solution species, Kd data, and studies for determining adsorption mechanisms.

623 (PNL-SA--7352(Vol.1), pp 331-403) CHROMATOGRAPHIC K/SUB D/ VALUES OF RADIONUCLIDES AND THE EFFECT OF REDOX POTENTIALS ON SORPTION OF RADIONUCLIDES BY GEOLOGIC MEDIA. Francis, C.W.; Reeves, M. III; Bondietti, E.A.; Smith, E.A. (Oak Ridge National Lab., TN). 1978. Dep. NTIS, PC A22/MF A01.

From 2. contractor information meeting; Seattle, WA, USA (1 Oct 1978).

Distribution and selectivity coefficients for ^{85}Sr and ^{63}Ni were determined in the presence of various Ca concentrations in channel chromatographs packed with Conasauga shale. These coefficients were obtained using a coupled-equations model consisting of (1) moisture transport in a partially saturated media, (2) mass transport of the calcium and radionuclide species contained in the influent, and (3) ion exchange between the ionic species in the solution phase and those sorbed on the solid matrix. A normalization procedure was developed so that profile data from one column or channel could be compared to data from other channels. Dispersivity measurements in the channel chromatographs were found to be the largest source of error in determining K/sub d/ values. Values for ^{85}Sr ranged from 2.4 to 3.6 and for ^{63}Ni from 5.3 to 8.6, and compared closely to those determined by a batch equilibrium method, 2.4 for ^{85}Sr and 6.5 for ^{63}Ni. Redox measurements of igneous rock suspensions in anoxic stirred cells predicted that the species of TcO_4^- and NpO_2^+ would not be the stable forms of these elements. Adding T/sub c/O_4- and NpO_2^+ to similar anoxic suspensions revealed that the species are reduced to less soluble states. Thus, current risk assessments, which consider the oxidized forms (TcO_4^- and NpO_2^+) as potentially migrating from high-level radioactive waste repositories, may be overestimating their potential hazard to the public.

624 (PNL-SA--7468) WORKSHOP ON TRANSPORT

MODELING FOR NUCLEAR WASTE REPOSITORIES. Raymond, J.R. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jul 1977. Contract EY-76-C-06-1830. 18p. Dep. NTIS, PC A02/MF A01.

The Transport Modeling task includes developing methods to describe the transport and impact of radiocontaminants accidentally released from waste repositories; demonstrating the use of these methods; and making the methods and techniques available to the repository managers for application at specific sites. The workshop on Transfer Modeling for the Waste Isolation Safety Assessment Program was held at the Battelle Seattle Research Center on 25 and 26 July 1977. The objectives of the workshop were to: 1) discuss needs and criteria for the models, 2) obtain critiques of proposed model systems from experts in the field, and 3) obtain ideas for making models more useful to the safety assessment program. The proceedings of the workshop were presented.

625 (PNL-SA--7875) NUMERICAL ANALYSIS OF RADIONUCLIDE MOVEMENT FROM A HYPOTHETICAL NUCLEAR WASTE REPOSITORY. Raymond, J.R.; Segol, G. (Battelle Pacific Northwest Labs., Richland, WA (USA); Bechtel National, Inc., San Francisco, CA (USA)). 1979. Contract EY-76-C-06-1830. 27p. Dep. NTIS, PC A03/MF A01.

A numerical analysis of the consequences of radionuclide release from a hypothetical nuclear waste repository located in bedded salt and of their subsequent transport by ground water has been performed. The study was based on available geologic and hydrologic information for the reference location. The initial run was based on a best estimate of geologic and hydrologic parameters for this reference location. Its results indicate that those isotopes which reach the natural discharge point remain below MPC levels for uncontrolled water use. The initial run was followed by a parametric study to evaluate the sensitivity of the model to the values of input parameters. This analysis shows that the model is sensitive to the assumed values of hydraulic conductivity, dispersivity and retardation factor, as expected. But this analysis also emphasizes that there exist complex interactions between such parameters which are less intuitively evident. The influence of waste leach rates is being investigated.

626 (PNL-SA--7945) ENVIRONMENTAL IMPACT STATEMENT ON MANAGEMENT OF COMMERCIALLY GENERATED RADIOACTIVE WASTES. Shupe, M.W.; Kreiter, M.R. (Department of Energy, Richland, WA (USA). Richland Operations Office; Pacific Northwest Lab., Richland, WA (USA)). 1979. Contract AC06-76RL01830. 19p. (CONF-791108--28). NTIS, PC A02/MF A01. Order Number DE82010668.

From 72. AIChE meeting; San Francisco, CA, USA (25 Nov 1979).

This report describes the generic environmental impact statement on the management of generated high-level and transuranic radioactive wastes. The contents of the statement are summarized. The alternatives considered include: geologic disposal; chemical resynthesis; very deep hole disposal; rock melting concept; island disposal; subseabed disposal; icesheet disposal; reverse well disposal; transmutation treatment; and space disposal concepts. The types and quantities of wastes considered are from 3 different fuel cycles for the LWR reactor: once through; uranium-only recycle; and uranium and platinum recycle. (DMC)

627 (PNL-SA--8571(Vol.2)) WASTE ISOLATION SAFETY ASSESSMENT PROGRAM. TASK 4. THIRD CONTRACTOR INFORMATION MEETING . (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jun 1980. Contract AC06-76RL01830. 392p. (CONF-7910160--(Vol.2)). NTIS, PC A17/MF A01.

From 3. contractor information meeting on waste isolation safety assessment program task 4; Seattle, WA, USA (14 Oct 1979).

The study subject of this meeting was the adsorption and desorption of radionuclides on geologic media under repository conditions. This volume contans eight papers. Separate abstracts were prepared for all eight papers. (DLC)

628 (PNL-SA--8571(Vol.2), pp 1-79) SYSTEMATIC STUDY OF NUCLIDE ADSORPTION ON SELECTED GEOLOGIC MEDIA. ANNUAL PROGRESS REPORT, OCTOBER 1, 1978-SEPTEMBER 17, 1979. Meyer, R.E. (comp.). (Oak Ridge National Lab., TN). Jun 1980. NTIS, PC A17/MF A01.

From 3. contractor information meeting on waste isolation safety assessment program task 4; Seattle, WA, USA (14 Oct 1979).

This report summarizes the past year's work concerning the adsorption of nuclides of interest to nuclear waste on clays, hydrous oxides, and other geologic media as a function of the pertinent variables of salt concentration, pH, and loading. Most of the adsorbent-adsorbate systems were investigated in the pH range 5 to 10 and the salt concentration range from 0.01 to 5 M NaCl in order to cover the most probable compositions of groundwaters in the region of waste repositories located in bedded salt formations or salt domes. A batch method was generally used to obtain the data, but for some special conditions, the axial filter was used. For adsorption on clay minerals at pH values of about 5 where hydrolysis reactions are not likely to occur, a strong salt dependence was observed, and the data could be fairly well described by ideal ion exchange equations. As the pH was increased, for some of the systems, e.g. for Eu(III) adsorption on montmorillonite, distribution coefficients increased significantly with pH and the salt dependence decreased or disappeared entirely. For adsorption of Sr(II) and Eu(III) on hydrous oxides, usually a form of Al_2O_3, significant increases in distribution coefficients with pH were observed and the salt dependance was very small, but often increasing somewhat as the pH was increased. Distribution coefficients for adsorption of I^- on a large variety of minerals and rocks were almost always very small, generally almost zero, and only for a few adsorbents under special conditions were distribution coefficients as high as 10 liters/kilogram observed. These results demonstrate that a large number of experiments are required for a complete and systematic investigation of a single nuclide-adsorbent system but also show that the dependence of the distribution coefficient on the variables show regularities which may be used to interpolate and extrapolate the data to nontested conditions.

629 (PNL-SA--8571(Vol.2), pp 235-260) WASTE/ROCK INTERACTION TECHNOLOGY. SOLUBILITY OF PLUTONIUM COMPOUNDS AND THEIR BEHAVIOR IN SOILS. Rai, D.; Serne, R.J.; Moore, D.A. (Pacific Northwest Labs., Richland, WA). 1 Feb 1980. NTIS, PC A17/MF A01.

From 3. contractor information meeting on waste isolation safety assessment program task 4; Seattle, WA, USA (14 Oct 1979).

The solubilities of $^{239}PuO_2(c)$ and $^{239}Pu(OH)_4(am)$ under natural environmental conditions were determined. These data were then used to predict the (1) nature of the solid phases present in contaminated soils, and (2) total concentration of Pu that can be expected in soil solutions when these Pu solids are present. Based upon solubility measurements, an estimated value of the log K/sup c/ for the dissolution of $^{239}PuO_2(c)$ [$PuO_2(c)$ reversible PuO_2 + e^-] was found to be -14.8. The estimated value of the log K/sup c/ for the dissolution of $^{239}Pu(OH)_4(am)$ [$Pu(OH)_4(am)$ reversible PuO_2 + 2 H_2O + e^-] was found to be -12.8. Comparison of Pu concentration in equilibrium solutions of contaminated Hanford soils with the $PuO_2(c)$ and $Pu(OH)_4(am)$ solubility lines suggested that $Pu(OH)_4(am)$ was absent from all the samples and that two of the samples contained $PuO_2(c)$. The presence of $PuO_2(c)$ was also confirmed by X-ray diffraction of Pu particles isolated from one of the samples.

630 (PNL-SA--8571(Vol.2), pp 213-234) WASTE/ROCK INTERACTIONS TECHNOLOGY. INTERACTIONS OF PLUTONYL (VI) WITH SOIL MINERALS. Rai, D.; Serne, R.J.; Moore, D.A. (Pacific Northwest Labs., Richland, WA). Feb 1980. NTIS, PC A17/MF A01.

From 3. contractor information meeting on waste isolation safety assessment program task 4; Seattle, WA, USA (14 Oct 1979).

Plutonium-239 adsorption by differnt minerals from aerated 0.005 M $CaCl_2$ solutions containing 7.3 x 10^{-9} M Pu(VI) was studied. At pH 4, adsorption by the selected minerals differed significantly, with adsorption increasing in the following order: gibbsite < kaolinite < montmorillonite < nontronite approx. = hematite < vermiculite approx. = biotite. At pH 7, the adsorption on these minerals did not differ significantly and exceeded 93% in all cases. Several mechanisms that contributed to Pu adsorption were identified: ion exchange, specific adsorption, reduction of Pu(VI), and Pu(VI) hydrolysis. Vermiculite, biotite, and hematite, containing structural Fe(II), were found to reduce Pu(VI) even in the well aerated suspensions. Although the initial ^{239}Pu concentration added to the mineral suspensions was 22 times the MPC of ^{239}Pu in unregulated water, the concentration in equilibrated solutions at pH 7 were near or lower than the MPC. This was also true for vermiculite, biotite, and hematite at pH 4. A very small percentage of the adsorbed Pu was released when the minerals were contacted with 1 M NaCl. These results indicate that Pu in general and Pu(VI) specifically would be relatively immobile in geomedia that do not contain significant quantities of complexing ligands.

631 (PNL-SA--8571(Vol.2), pp 81-133) SORPTION-DESORPTION OF LONG-LIVED RADIONUCLIDE SPECIES ON GEOLOGIC MEDIA. ANNUAL REPORT. Francis, C.W.; Bondietti, E.A. (Oak Ridge National Lab., TN). 1 Oct 1979. NTIS, PC A17/MF A01.

From 3. contractor information meeting on waste isolation safety assessment program task 4; Seattle, WA, USA (14 Oct 1979).

Influence of time, temperature, pH, Eh, Fe^{+2}, and Tc concentrations on sorption-desorption of Tc on basalt, granite, and shale was investigated. Loss of TcO^-_4 from solution appears not to be due to sorption, but rather due to the reduction to a lower oxidation state, probably TcO_2 which is insoluble. Loss of TcO^-_4 increased with time. Increasing the temperature enhanced the initial reduction rate, but the rate exponentially decayed at high temperatures (72 vs 22°C). Basalt reduced TcO^-_4 faster and in greater quantities than the granite or shale samples. 10^8 more Tc-99 was retained on anoxic basalt columns than Tc-95m; however, the Sorption Concentration Ratios (0.64 and 3.54 ml/g) were not appreciably different. The additions of Fe^{+2} prior to TcO^-_4 additions did not affect TcO^-_4 reduction; but, additions of Fe^{+2} to Tc-rock suspensions reduced TcO^-_4. The reduced form of Tc is slowly oxidized to the TcO^-_4 under oxic conditions. In Eh-pH monitored stirred suspensions, the single most important variable influencing the loss of TcO^-_4 was Eh of the suspension. Losses of TcO^-_4 from basalt suspensions occurred at measured Eh's 0.240 volts higher than theoretically required for the formation of the water-insoluble TcO_2. Pertechnetate (μM) was observed to be stable in the presence of 100 μM Fe^{+2}, pH 5, and Eh approx. +0.400 volts; even at Fe^{+2}/TcO^-_4 ratios > 10^8, the pertechnetate was observed to be stable at pH's between 4.5 and 5.5. As the pH was increased to 6.5 and greater the loss of TcO^-_4 from solution was directly proportional to the loss of Fe^{+2}. TcO^-_4 was reduced by Fe^{+2} as the pH was increased, and Tc^{+4} was isomorphically substituted into the lattice of $Fe(OH)_3$ because of the similarity in ionic radii (0.65 A) of Tc^{+4} and Fe^{+3}.

632 (PNL-SA--8571(Vol.2), pp 135-158) LABORATORY STUDIES OF PU-237 SORPTION ON SELECTED MINERALS UNDER ANOXIC CONDITIONS. Relyea, J.F.; Serne, R.J.; Fulton, R.W.; Washburne, C.D.; Martin, W.J. (Pacific Northwest Lab., Richland, WA). 1 Oct 1979. NTIS, PC A17/MF A01.

From 3. contractor information meeting on waste isolation safety assessment program task 4; Seattle, WA, USA (14 Oct 1979).

Corrections have been made for measured Kd values for plutonium under anoxic conditions after 30 days of shaking. Values for Kd(Pu) were highest for the phyllosilicate minerals illite, montmorillonite, vermiculite and biotite. Corrected Kd/sub a/(Pu) values were generally higher under anoxic than ambient atmospheric conditions in both 5.13 N NaCl and 0.03 N $CaCl_2$. Lower values for Kd/sub a/(Pu) were observed in 0.03 N $NaHCO_3$ under anoxic than under ambient conditions. There were no statistically significant differences in Kd/sub a/(Pu) for the two oxygen conditions for the 0.03 N NaCl solution. Thermodynamic data suggest that Pu sorption would be greater under anoxic conditions as shown in two of the four solutions used. There is no explanation for the apparent lack of agreement of the other two solutions. In general Pu adsorption by the minerals was high with the lowest mean value for corrected data being Kd/sub a/(Pu) = 92 for quartz under anoxic conditions. Of the 96 permutations (12 minerals x 4 solutions x 2 oxygen conditions) in only 14 instances did the Kd/sub a/(Pu) fall below 1000 ml/g and in only 5 instances did Kd/sub a/(Pu) fall below 500 ml/g.

633 (PNL-SA--8571(Vol.2), pp 261-315) KINETICS AND REVERSIBILITY OF RADIONUCLIDE SORPTION REACTIONS WITH ROCKS. PROGRESS REPORT FOR FISCAL YEAR 1979. Barney, G.S.; Brown, G.E. (Rockwell International, Richland, WA). Sep 1979. NTIS, PC A17/MF A01.

From 3. contractor information meeting on waste isolation safety assessment program task 4; Seattle, WA, USA (14 Oct 1979).

Sorption-desorption reactions of cesium, strontium, neptunium, americium, and plutonium on basalt, granite, and argillite were observed for 218 days. Equilibrium in batch experiments was not reached for most radionuclides even after this long time. Reactions of the crushed rock with ground waters (dissolution, hydrolysis, precipitation, etc.) also did not reach equilibrium after 150 days. The dissolution of basalt is accompanied by the formation of colloidal particles which contain Si, Fe, Ca, and Al. These colloids sorb Cs, Sr, Am, and Pu during equilibration. Some of the colloids pass through 0.3-μm flters, are not retained even on 0.01-μm filters and, therefore, cause calculated K/sub d/ values to be too low. Samples of crushed basalt, granite, and argillite were artificially weathered by continuous leaching with distilled water for 6 months both in air and in an oxygen-free stream of nitrogen gas. The weathered rock was then characterized for surface area, surface structure, cation exchange capacity, and composition of weathered surface on the rock. Comparisons were made of radionuclide sorption (after 14 days) on fresh rock, rock weathered in air, and rock weathered in N_2. Sorption on rocks weathered in N_2 generally is less than on rock weathered in air. This is possibly due to the lack of an $Fe(OH)_3$ coating on the rock weathered in N_2. The $Fe(OH)_3$ is known to scavenge cations and silica from solution. Sorption of Cs, Si, Am, and Pu is strongly affected by weathering basalt and argillite. However, the cation exchange capacity is changed very little, suggesting that ion exchange plays a minor role in sorption of these radionuclides.

634 (PNL-SA--8571(Vol.2), pp 159-171) EFFECTS OF HYDROGEN PEROXIDE PRETREATMENT OF CLAY MINERALS ON THE ADSORPTION OF SR-85 AND TC-95M UNDER ANOXIC CONDITIONS. Relyea, J.F.; Washburne, C.D. (Pacific Northwest Lab., Richland, WA). Jul 1979. NTIS, PC A17/MF A01.

From 3. contractor information meeting on waste isolation safety assessment program task 4; Seattle, WA, USA (14 Oct 1979).

Treatment of three clay minerals with hydrogen peroxide affects the observed adsorption behavior of technetium relative to untreated clay under anoxic conditions. A possible adsorption mechanism of Tc is the reduction of TcO^-_4 to a more positively charged or better adsorbed species. Oxidation of the clay by H_2O_2 would hinder the reduction of TcO^-_4 by buffering the clay-water system at a higher Eh value, although a difference in measured Eh value may go undetected. Sorption of strontium by the clays under the same conditions is not affected by a pretreatment with H_2O_2. The behavior of strontium follows that expected from ion exchange theory. 13 tables.

635 (PNL-SA--8571(Vol.2), pp 173-211) SORPTION BEHAVIOR OF ^{85}SR, ^{137}CS AND /SUP 95M/ TC UNDER NORMAL ATMOSPHERIC AND REDUCED OXYGEN LEVELS. Relyea, J.F.; Washburne, C.D.; Fulton, R.W. (Pacific Northwest Lab., Richland, WA). Jul 1979. NTIS, PC A17/MF A01.

From 3. contractor information meeting on waste isolation safety assessment program task 4; Seattle, WA, USA (14 Oct 1979).

Identical mineral samples and simulated groundwaters were used in a study of the sorption of technetium, strontium and cesium under anoxic conditions. Results from these experiments show that the anoxic sorption of strontium and cesium is generally unchanged or lower than sorption under ambient atmospheric conditions. The sorption of both Cs and Sr is inhibited by the presence of high NaCl concentrations (brines), and Sr sorption is reduced by the presence of Ca^{2+} in solution. Values for Kd(Cs) and Kd(Sr) often change with time, as both the mineral properties and solution chemical composition change due to weathering of the minerals. In these instances equilibrium has not been attained; differences between ambient and anoxic Kd values found at short contact times may disappear at equilibrium. Both Kd values and their variability were increased for technetium under anoxic conditions. Technetium adsorption is greatest on the hard rock minerals (tektosilicates and inosilicates) which may be influenced by small chunks of metal from the apparatus used to grind the minerals to power form. 21 tables.

636 (PNL-SA--8733) EFFECT OF ORGANIC LIGANDS ON THE SOIL BEHAVIOR OF TECHNETIUM-99. Martin, L.Y.; Franz, J.A. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jun 1980. Contract AC06-76RL01830. 10p. (CONF-801107--68). NTIS, PC A02/MF A01.

From ANS international conference; Washington, DC, USA (17 Nov 1980).

Results of studies on the effects of organics on ^{99}Tc mobility are reported. The effects of organics (EDTA,DTPA and citrate) on the sorption/migration of Tc is examined from two aspects, first by desorption techniques where reduced, sorbed Tc is exposed to organic ligands; and second, by exposure of synthetic Tc-organic complexes to soils. A calcareous, sandy, loam Hanford soil (pH approx. 8.2) was used. Very little desorption of the reduced Tc has occurred both in the 10 day study (95-87% remains sorbed for 10^{-6} to 10^{-9} M levels even at cit/Tc - 10,000= and even less desorption is observed in the extended 45 day study (95-90% remains sorbed for 10^{-6} to 10^{-9} M levels). Similar results were also observed when a stronger chelating agent such as EDTA was used. Approximately 95% of the reduced Tc remains sorbed for the 10^{-5} to 10^{-6} M levels and approx. 70 remains sorbed at 10^{-7} M.

637 (PNL-SA--8935) DEVELOPMENT OF A SOURCE TERM FOR RADIONUCLIDE RELEASE FROM SPENT FUEL IN A BASALT REPOSITORY. Kuhn, W.L. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 1980. Contract AC06-76RL01830. 21p. (CONF-801206--11). NTIS, PC A02/MF A01.

From ORNL conference on the leachability of radioactive solids; Gatlinburg, TN, USA (9 Dec 1980).

An analysis of groundwater transport of radionuclides from a nuclear waste repository requires a source term, i.e., a prediction of the rate of release of radionuclides from the repository. A source term has been constructed to support a demonstration of analytical methodology developed for use in repository licensing. For this the case of release from spent fuel in a basalt repository sheared by a fault was considered. The resulting source term illustrates the role of leaching behavior without referring to a specific detailed waste package design. The technical approach used to construct the source term is discussed to illustrate how information on nuclear waste leaching behavior might be used in support of repository licensing activites. Assumptions made in lieu of a complete understanding of leaching behavior are discussed.

638 (PNL-SA--8992) IN SITU SUBTERRANEAN DETERMINATION OF ACTINIDES BY HIGH-RESOLUTION GAMMA-RAY SPECTROMETRY. Brodzinski, R.L. (Pacific Northwest Lab., Richland, WA (USA)). Apr 1981. Contract AC06-76RL01830. 14p. (CONF-810409--6; IAEA-SM--252/13). NTIS, PC A02/MF A01. Order Number DE82005903.

From IAEA international symposium on methods of low-level counting and spectrometry; Berlin, F.R. Germany (6 Apr 1981).

A system utilizing high resolution germanium diode gamma-ray spectroscopy for the simple, safe, and economical in situ determination of actinides is described. Six isotopes, ^{235}U, ^{238}U, ^{237}Np, ^{239}Pu, ^{241}Pu, and ^{241}Am, can be simultaneously measured at the 10 nCi g^{-1} level in less than 7 minutes. Collimators provide for measurement of horizontal strata as thin as 1 cm or solid angles as small as 0.1 steradians. Information obtainable with the system is discussed and compared to that obtainable with neutron activation/detection systems.

639 (PNL-SA--9503) USE OF CONTROLLED RELEASE HERBICIDES IN WASTE BURIAL SITES. Burton, F.G.; Cataldo, D.A.; Cline, J.F.; Skiens, W.E. (Pacific Northwest Lab., Richland, WA (USA)). Jul 1981. Contract AC06-76RL01830. 17p. (CONF-810765--2). NTIS, PC A02/MF A01. Order Number DE82006167.

From 8. international symposium on controlled release of bioactive materials; Fort Lauderdale, FL, USA (26 Jul 1981).

Controlled-release formulations of herbicides have been applied to the soil in the manner traditional for herbicides: on the surface or mixed into the top few inches of soil. The controlled-release formulation allows another option that we propose to use: to place herbicides, contained in controlled-release formulations, in a layer at least a foot below the surface of the soil, in order to prevent root penetration below that level. Ideally, the herbicide will prevent root tip cell division but will not translocate within the plant, thus assuring that the plant will survive, preserving the ground cover. Trifluralin is one of the herbicides which does not translocate and was chosen for use in this study. A number of applications for this technology are possible; particularly in waste management. In the present studies, we used two different forms of polymeric carrier/delivery (PCD) systems to investigate the controlled release of herbicides. In the initial study, a sheet was made of homogeneous mixtures of an individual polymer and trifluralin. We made several of these sheets, using a different polymer each time (with trifluralin) to compare release rates from the various polymers. We also fabricated cylindrical pellets in two sizes from mixtures of Profax/sup a/ PS-1600 polypropylene and trifluralin, formulated to determine the interaction of PCD systems with soil. Also developed is a trifluralin-releasing device with a theoretical effective lifetime approaching 100 years. The system was designed specifically to protect the asphalt layer or clay/aggregate barriers on uranium mill tailings piles. PCD devices composed of pellets could also be implanted over burial sites for radioactive and/or toxic materials, preventing translocation of those materials to plant shoots, and thence into the biosphere.

640 (UCID--18240) LEACHING CHARACTERISTICS OF ACTINIDES FROM SIMULATED REACTOR WASTE GLASS. Weed, H.C.; Coles, D.G.; Bradley, D.J.; Mensing, R.W.; Schweiger, J.S.; Rego, J.H. (Lawrence Livermore National Lab., CA (USA)). Aug 1979. Contract W-7405-ENG-48. 14p. Dep. NTIS, PC A02/MF A01.

Even without statistical analysis, some general trends can be seen in the results: leach rate increases with flow rate at high temperature, but is approximately independent of it at room temperature; agreement between the results from the one-pass method and those from the IAEA method are fair in the case of WIPP brine solution, and good in the case of the others; and the ^{237}Np leach rate increases with temperature, but the ^{239}Pu leach rate either decreases with temperature or does not change.

641 LOOSE ROCK AS BIOBARRIERS IN SHALLOW LAND BURIAL. Cline, J.F.; Gano, K.A.; Rogers, L.E. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Health Physics; 39: No. 3, 497-504(Sep 1980).

A layer of loose rock placed between buried waste and topsoil successfully prevented plant roots, burrowing mammals and ants from translocating waste materials into the biosphere in an arid climate. Air spaces between the rocks had to be maintained with stones of between 3.8 and 7.6 cm diameter. Small rocks and asphalt emulsion prevented the topsoil from sifting between the rocks that may make a pathway for plant roots to the waste material.

642 RADIOLOGICAL CHARACTERIZATION OF UNDOCUMENTED NUCLEAR SITES. McMurray, B.J. (Battelle Northwest Labs., Richland, WA). Transactions of the American Nuclear Society ; 34: 110(1980). (CONF-800607--).

From American Nuclear Society annual meeting; Las Vegas, NV, USA (8 Jun 1980).

NUCLEAR FACILITIES;ABANDONED SITES; EVALUATION;RADIOMETRIC SURVEYS;SAMPLING; PLANNING;DECOMMISSIONING;DECONTAMINATION

643 (BNWL-tr--311) MEASUREMENT OF THE DISTRIBUTION OF ^{129}I IN AND ITS DISCHARGE FROM THE KARLSRUHE REPROCESSING PLANT. Berg, R.; Schuttelkopf, H. Mar 1978. Contract EY-76-C-06-1830. Translated from pp 1-10 of a symposium on radioactive effluents from nuclear fuel reprocessing plants, Karlsruhe, November 22-24, 1977 (CONF-771139--2). 18p. Dep. NTIS, PC A02/MF A01.

From Radioactive effluents from nuclear fuel reprocessing plants seminar; Karlsruhe, F.R. Germany (22 Nov 1977).

Since 1975, measurements of ^{129}I have been carried out in almost all solutions and in the exhaust air of the Karlsruhe Reprocessing Plant (WAK). Samples from the air are collected by solid filters containing AC 6120. The solutions are treated radiochemically according to their composition. The separated ^{129}I was measured either by x-ray spectrometry or by neutron activation. During dissolution, more than 99% of the calculated ^{129}I inventory is released into the offgas. The separation efficiency of offgas scrubbers was measured. ^{129}I contents of the clean-air mixture from the dissolver offgas and of the vessel offgas and of the offgas from the high-level waste solutions have been measured since the start of 1975. Residual iodine in the feed solution was analyzed. After concentration of the high-level waste solution, the iodine in the concentrate and in the offgas of the concentrate vessel was measured once again. The ^{129}I discharged with low-level liquid waste has been monitored for almost two years. The solutions resulting from distillative purification of medium-level and low-level liquid wastes, as well as the liquid wastes from the final storage pools, have also been monitored in terms of their ^{129}I content for almost two years. From the results acquired, the complete iodine balance of the Karlsruhe Reprocessing Plant is established.

644 AIRBORNE PARTICULATE CONCENTRATIONS AND FLUXES AT AN ACTIVE URANIUM MILL TAILINGS SITE. Sehmel, G.A. (Battelle Pacific Northwest Labs.,

Richland, WA (USA)). pp 65-84 of Management, stabilisation and environmental impact of uranium mill tailings. Paris, France; OECD Nuclear Energy Agency (1978).
From Management, stabilization, and environmental impact of uranium mill trailings; Albuquerque, NM, USA (24 Jul 1978).
Direct measurements of airborne particulate concentrations and fluxes of transported mill tailing materials were measured at an active mill tailings site. Experimental measurement equipment consisted of meteorological instrumentation to automatically activate total particulate air samplers as a function of wind speed increments and direction, as well as particle cascade impactors to measure airborne respirable concentrations as a function of particle size. In addition, an inertial impaction device measured nonrespirable fluxes of airborne particles. Caclulated results are presented in terms of the airborne solid concentration in g/m^3, the horizontal airborne mass flux in g/(m^2-day) for total collected nonrespirable particles and the radionuclide concentrations in dpm/g as a function of particle diameter for respirable and nonrespirable particles.

645 (BNWL-tr--273) MATHEMATICAL MODELLING OF RADIONUCLIDE MIGRATION IN A HOMOGENEOUS CLAYEY FORMATION (APPLICATION TO THE CASE OF THE BOOM CLAY AT MOL). Put, M.; Heremans, R. Oct 1977. Contract EY-76-C-06-1830. Translation of Belgian report (CONF-770565--3). 18p. Dep. NTIS, PC A02/MF A01.
From Workshop on risk analysis and geologic modelling; Ispra, Italy (23 May 1977).
In the vicinity of CEN/SCK at Mol, the Tertiary clayey formation lies between aquiferous sands at depths of 160 to 270 m. Some experimental measurements are given. A mathematical model was developed and applied to ^{239}Pu. It is concluded that, for the Mol site, the Pu would move at the most 7.22 m in 100,000 years or 18 m in 250,000 years. (DLC)

646 EFFLUENTS FROM THE LWR STOWAWAY FUEL CYCLE. Kabele, T.J.; Heeb, C.M.; Schneider, K.J.; Kennedy, W.E.; Lewallen, M.A. (Battelle Northwest Labs., Richland, WA). Transactions of the American Nuclear Society ; 27: 127-128(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
FUEL CYCLE;RADIOACTIVE EFFLUENTS;PWR TYPE REACTORS;BWR TYPE REACTORS;WATER COOLED REACTORS;ENVIRONMENTAL IMPACTS;CHEMICAL EFFLUENTS;RADIOACTIVE WASTE STORAGE;HIGH-LEVEL RADIOACTIVE WASTES

647 ENFORM FUEL CYCLE EFFLUENTS MODEL. Heeb, C.M.; Lewallen, M.A.; Purcell, W.L. (Battelle Northwest Labs., Richland, WA). Transactions of the American Nuclear Society ; 27: 128-129(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
FUEL CYCLE;RADIOACTIVE EFFLUENTS;CHEMICAL EFFLUENTS;BWR TYPE REACTORS;PWR TYPE REACTORS;WATER COOLED REACTORS;MATHEMATICAL MODELS;ENVIRONMENTAL IMPACTS

648 ENVIRONMENTAL ASSESSMENT METHODOLOGY FOR THE NUCLEAR FUEL CYCLE. Brenchley, D.L.; ; Soldat, J.K.; McNeese, J.A.; Watson, E.C. (Battelle Northwest Labs., Richland, WA). Transactions of the American Nuclear Society ; 27: 129-130(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
FUEL CYCLE;RADIOACTIVE EFFLUENTS;CHEMICAL EFFLUENTS;PWR TYPE REACTORS;BWR TYPE REACTORS;MATHEMATICAL MODELS;ENVIRONMENTAL IMPACTS

Health and Safety

649 (BNWL--2269) SUPPLEMENT TO SAFETY ANALYSIS REPORT FOR THE 325 RADIOCHEMISTRY BUILDING. Bryan, G.H.; Wittenbrock, N.G. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 24 Jun 1977. Contract EY-76-C-06-1830. 80p. Dep. NTIS, PC A05/MF A01.
The Waste Preparation Task (Task 6) of the Commercial Nuclear Waste Vitrification Project (CNWVP) includes the design, installation and operation of small scale (1 MTU LWR fuel/month) equipment to prepare high-level liquid waste (HLLW) from commercial LWR spent fuel. The HLLW will be used in Task 5, the radioactive demonstration of waste vitrification. The operational aspects of this task include the procurement of irradiated LWR fuel; the processing operations to prepare HLLW; and other necessary operations associated with the disposition of the uranium waste and other fuel residues, including the plutonium associated with the fuel. The processing operations in the 325-A Building will require handling in excess of 45 percent of a Minimum Critical Mass (MCM), use of an organic solvent to extract uranium and plutonium away from the fission products, and ion exchange resin for plutonium purification. An analysis of the environmental consequences and probability of conceivable accidental conditions that may result from this operation indicates no undue consequences to the environment. The calculated maximum environmental consequences of postulated accidents would be low radiotoxic doses of 0.3 rem whole body 50-yr dose commitment to a maximum individual and 20 man-rem whole body 50-yr dose commitment to the surrounding population. The individual radiological exposures are much below the permissible exposure guidance levels in 10CFR100 of 25 rem whole body and 300 rem thyroid doses for nuclear reactor accidents of very low probability of occurrence. The calculated relative dose risk to individuals from accidental conditions (1.6 x 10^{-6} rem/yr) is very low compared to regulatory guidence for routine releases from nuclear reactors (5.0 x 10^{-3} rem/yr). Therefore, the design and operational plans for the CNWVP are judged not to represent an undue environmental risk from accidental conditions.

650 (BNWL-SA--6121) HEALTH PHYSICS ASPECTS OF WASTE MANAGEMENT. Unruh, C.M. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jul 1977. Contract EY-76-C-06-1830. 12p. (CONF-770720--14). Dep. NTIS, PC A02/MF A01.
From 22. annual meeting of the Health Physics Society; Atlanta, GA, USA (3 Jul 1977).
Radioactive waste is encountered at every step of the LWR fuel cycle, and health physics is needed in the management of these wastes. Environmental surveillance will be needed. Health physics analysis supports the program to put radioactive waste in deep geologic isolation. The role of the health physicist in public interaction and communications is discussed. (DLC)

651 (PB--296947, pp 270-279) USE OF A RISK LIMIT AS AN ENVIRONMENTAL SAFETY STANDARD FOR

RADIOACTIVE WASTE DISPOSAL SITES. Desrosiers, A.E. (Battelle Pacific Northwest Labs., Richland, WA); Njoku, E. May 1979. NTIS, PC A99/MF A01.

From 12. Health Physics Society midyear topical symposium; Williamsburg, VA, USA (12 Feb 1979).

this paper demonstrates how the recommendations of ICRP Publication 26 may be applied to setting environmental radiation standards for radioactive waste disposal sites. Traditionally, such standards prescribe the allowable radiation dose to the maximally exposed offsite individual. Dose limits are usually established for the total body and for individual organs. In this paper, the risk factors recommended by ICRP for individual organs and the doses to those organs are combined to calculate the total risk per unit of ingested radioactivity. The allowable ingestion of radioactivity is then calculated from ICRP's individual risk limit. When these data are compared to normally derived ingestion limits, significant differences appear whenever a relatively insensitive organ receives the majority of the dose. Maximum allowable concentrations of radionuclides in water are derived from the ingestion limits and the concept of an effective water consumption rate. Using risk assessments in environmental standards for waste disposal sites would allow (1) a rapid and conservative (cautious) assessment of the potential health impact of a waste disposal facility, and (2) a simpler evaluation of the impact of ingesting several radionuclides even if each radionuclide affects different human organs.

652 (PNL--2851, pp II.1-II.19) PACIFIC NORTHWEST GEOMORPHOLOGY AND HYDROLOGY: RATES AND PROBABILITIES OF SELECTED PROCESSES AND EVENTS. Tubbs, C.W. (Univ. of Kansas, Lawrence). Nov 1979. Dep. NTIS, PC A09/MF A01.

Summary of FY-1978 consultation input for Scenario Methodology Development.

This report presents results of one of the geomorphological and hydrological studies that have been conducted for the release scenario analysis of the Waste Isolation Safety Assessment Program (WISAP). Three general topics are considered: (1) determination of rates of denudation, (2) estimation of the probability of flooding due to each of several causes, and (3) evaluation of other surface processes that should be considered in the release scenario analysis. The third general topic was ultimately narrowed to the possible effects of landsliding. Rates of erosion are expressed as centimeters per 100 years, except that the original units are retained in figures taken from other sources. Probabilities are also expressed per 100 years.

653 (PNL--2851, pp III.1-III.21) ANALYSIS OF THE SEISMIC HAZARD TO AN UNDERGROUND WASTE REPOSITORY. Wight, L.H. (Tera Corp., Berkeley, CA). Nov 1979. Dep. NTIS, PC A09/MF A01.

Summary of FY-1978 consultation input for Scenario Methodology Development.

Conclusions are: The consequence associated with intense vibratory shaking of a well-designed repository is essentially negligible. The specification of an appropriate seismic vibratory design criteria could best be accomplished with a Bayesian seismic hazard assessment, using geologic slip rates as input. The consequence associated with fault displacement is very site specific and dependent on the host geologic media and its permeability changes in response to fault displacement. The probability of faulting through a repository in its million year design life is rather high, principally because of a high probability of primary or secondary faulting on undetected faults. The faulting probability can be minimized by deploying sophisticated site certification programs. High resolution microseismic surveillance seems to be most appropriate. The author's judgement is that the repository simulation program can neglect consequences associated with shaking of the repository, but that the probability of significant fault displacement through the repository during its design life should be conservatively taken as one.

654 (PNL--2851, pp IV.1-IV.13) GLACIOLOGICAL PARAMETERS OF DISRUPTIVE EVENT ANALYSIS. Bull, C. (Ohio State Univ., Columbus). Nov 1979. Dep. NTIS, PC A09/MF A01.

Summary of FY-1978 consultation input for Scenario Methodology Development.

The following disruptive events caused by ice sheets are considered: continental glaciation, erosion, loading and subsidence, deep ground water recharge, flood erosion, isostatic rebound rates, melting, and periodicity of ice ages. (DLC)

655 (PNL--2851, pp V.1-V.8) GEOLOGIC FACTORS IN THE ISOLATION OF NUCLEAR WASTE: EVALUATION OF LONG-TERM GEOMORPHIC PROCESSES AND EVENTS. Mara, S.J. Nov 1979. Dep. NTIS, PC A09/MF A01.

Summary of FY-1978 consultation input for Scenario Methodology Development.

In this report the rate, duration, and magnitude of changes from geomorphic processes and events in the Southwest and the Gulf Coast over the next million years are projected. The projections were made by reviewing the pertinent literature; evaluating the geomorphic history of each region, especially that during the Quaternary Period; identifying the geomorphic processes and events likely to be significant in the two regions of interest; and estimating the average and worst-case conditions expected over the next million years.

656 (PNL--2851, pp VI.1-VI.15) LONG-TERM METEORITE HAZARDS TO BURIED NUCLEAR WASTE. REPORT 2. Hartmann, W.K. (Planetary Sciences Inst., Santa Clara, CA). Nov 1979. Dep. NTIS, PC A09/MF A01.

Summary of FY-1978 consultation input for Scenario Methodology Development.

Part 1 presents new data on the relation between crater size and total impact energy, with equation (1) expressing the relation. Part 2 derives an equation which gives the rate of accumulation of area covered by craters larger than diameter D. A graphical relation between D and the depth of disturbance is given. This section concludes that the probability of a single site 600 m deep being disturbed in a million years is of the order 2.5×10^{-8}. Part 4 points out that meteorite impacts are also sources of seismic disturbance and should be factored into the seismic model for the hazard study. Another equation gives a methodology for including meteorite impacts in the seismic model. Part 5 and an equation give a methodology for dealing with repositories with extended surface area. Part 6 gives examples of applications.

657 (PNL--2851, pp X.1-X.3) HYDROLOGIC EFFECTS OF NATURAL DISRUPTIVE EVENTS ON NUCLEAR

REPOSITORIES. Davis, S.N. (Univ. of Arizona, Tucson). Nov 1979. Dep. NTIS, PC A09/MF A01.
Summary of FY-1978 consultation input for Scenario Methodology Development.
This report describes some possible hydrogeologic effects of disruptive events which may affect repositories for nuclear waste. The report concentrates on the effects of natural events which are judged to be most probable.

658 (PNL--2851, pp XIII.1-XIII.8) PROBABILITY OF EXPECTED CLIMATE STRESSES IN NORTH AMERICA IN THE NEXT ONE MY. Kukla, G. (Lamont-Doherty Geological Observatory, Palisades, NY). Nov 1979. Dep. NTIS, PC A09/MF A01.
Summary of FY-1978 consultation input for Scenario Methodology Development.
Climates one million years ahead were predicted upon the assumption that the natural climate variability during the past My will continue. Response of environment and climate in the Basin and Range province of the western USA to global fluctuations was reconstructed; the most remarkable change was the filling of closed basins with large freshwater lakes. Probabilities of permanent ice cover and floods are discussed. It is believed that a site with minimal probability of climate-related breach can be selected. (DLC)

659 (PNL--2892) ASSESSMENT OF EFFECTIVENESS OF GEOLOGIC ISOLATION SYSTEMS. A CONCEPTUAL SIMULATION MODEL FOR RELEASE SCENARIO ANALYSIS OF A HYPOTHETICAL SITE IN COLUMBIA PLATEAU BASALTS. Stottlemyre, J.A.; Petrie, G.M.; Benson, G.L.; Zellmer, J.T. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jan 1981. Contract AC06-76RL01830. 180p. NTIS, PC A09/MF A01.
This report is a status report for an evolving methodology for release scenario development for underground nuclear waste repositories. As such, it is intended for use as a reference point and a preliminary description of an evolving geoscience methodology. When completed this methodology will be used as a tool in developing disruptive release scenarios for analyzing the long-term safety of geological nuclear waste repositories. While a basalt environment is used as an example, this report is not intended to reflect an actual site safety assessment for a repository in a media. It is rather intended to present a methodology system framework and to provide discussions of the geological phenomena and parameters that must be addressed in order to develop a methodology for potential release scenarios. It is also important to note that the phenomena, their interrelationships, and their relative importance along with the overall current structure of the model will change as new geological information is gathered through additional peer review, geotechnical input, site specific field work, and related research efforts.

660 (PNL--3180) ASSESSMENT OF EFFECTIVENESS OF GEOLOGIC ISOLATION SYSTEMS. ARRRG AND FOOD: COMPUTER PROGRAMS FOR CALCULATING RADIATION DOSE TO MAN FROM RADIONUCLIDES IN THE ENVIRONMENT. Napier, B.A.; Roswell, R.L.; Kennedy, W.E. Jr.; Strenge, D.L. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jun 1980. Contract AC06-76RL01830. 203p. NTIS, PC A10/MF A01.
The computer programs ARRRG and FOOD were written to facilitate the calculation of internal radiation doses to man from the radionuclides in the environment and external radiation doses from radionuclides in the environment. Using ARRRG, radiation doses to man may be calculated for radionuclides released to bodies of water from which people might obtain fish, other aquatic foods, or drinking water, and in which they might fish, swim or boat. With the FOOD program, radiation doses to man may be calculated from deposition on farm or garden soil and crops during either an atmospheric or water release of radionuclides. Deposition may be either directly from the air or from irrigation water. Fifteen crop or animal product pathways may be chosen. ARRAG and FOOD doses may be calculated for either a maximum-exposed individual or for a population group. Doses calculated are a one-year dose and a committed dose from one year of exposure. The exposure is usually considered as chronic; however, equations are included to calculate dose and dose commitment from acute (one-time) exposure. The equations for calculating internal dose and dose commitment are derived from those given by the International Commission on Radiological Protection (ICRP) for body burdens and Maximum Permissible Concentration (MPC) of each radionuclide. The radiation doses from external exposure to contaminated farm fields or shorelines are calculated assuming an infinite flat plane source of radionuclides. A factor of two is included for surface roughness. A modifying factor to compensate for finite extent is included in the shoreline calculations.

661 (PNL--3340) ESTIMATED AIRBORNE RELEASE OF PLUTONIUM FROM THE EXXON NUCLEAR MIXED OXIDE FUEL PLANT AT RICHLAND, WASHINGTON AS A RESULT OF POSTULATED DAMAGE FROM SEVERE WIND AND EARTHQUAKE HAZARD. Mishima, J.; Schwendiman, L.C.; Ayer, J.E.; Owzarski, E.L. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Feb 1980. Contract AC06-76RL01830. 45p. NTIS, PC A03/MF A01.
The potential airborne releases of plutonium from postulated damage sustained by the Exxon Nuclear Company's Mixed Oxide Fabrication Plant at Richland, Washington, as a result of various levels of wind and earthquake hazard, are estimated. The releases are based on damage scenarios that range up to 250 mph for wind hazard and in excess of 1.0 g ground acceleration for seismic hazard, which were developed by other specialists. The approaches and factors used to estimate the releases (inventories of dispersible materials at risk, damage levels and ratios, fractional airborne releases of dispersible materials under stress, atmosphere exchange rates, and source term ranges) are discussed. Release estimates range from less than 10^{-7} g to greater than 14 g of plutonium over a four-day period.

662 (PNL--3514) DECONTAMINATION OF HIGH-LEVEL WASTE CANISTERS. Nesbitt, J.F.; Slate, S.C.; Fetrow, L.K. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Dec 1980. Contract AC06-76RL01830. 77p. NTIS, PC A05/MF A01.
This report presents evaluations of several methods for the in-process decontamination of metallic canisters containing any one of a number of solidified high-level waste (HLW) forms. The use of steam-water, steam, abrasive blasting, electropolishing, liquid honing, vibratory finishing and soaking have been tested or evaluated as potential techniques to decontaminate the outer surfaces of HLW canisters. Either these techniques have been tested or available literature has been examined to assess their applicability to the

decontamination of HLW canisters. Electropolishing has been found to be the most thorough method to remove radionuclides and other foreign material that may be deposited on or in the outer surface of a canister during any of the HLW processes. Steam or steam-water spraying techniques may be adequate for some applications but fail to remove all contaminated forms that could be present in some of the HLW processes. Liquid honing and abrasive blasting remove contamination and foreign material very quickly and effectively from small areas and components although these blasting techniques tend to disperse the material removed from the cleaned surfaces. Vibratory finishing is very capable of removing the bulk of contamination and foreign matter from a variety of materials. However, special vibratory finishing equipment would have to be designed and adapted for a remote process. Soaking techniques take long periods of time and may not remove all of the smearable contamination. If soaking involves pickling baths that use corrosive agents, these agents may cause erosion of grain boundaries that results in rough surfaces.

663 (PNL--3601) ESTIMATED AIRBORNE RELEASE OF PLUTONIUM FROM THE 102 BUILDING AT THE GENERAL ELECTRIC VALLECITOS NUCLEAR CENTER, VALLECITOS, CALIFORNIA, AS A RESULT OF POSTULATED DAMAGE FROM SEVERE WIND AND EARTHQUAKE HAZARD. Mishima, J.; Ayer, J.E.; Hays, I.D. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Dec 1980. Contract AC06-76RL01830. 53p. NTIS, PC A04/MF A01.

This report estimates the potential airborne releases of plutonium as a consequence of various severities of earthquake and wind hazard postulated for the 102 Building at the General Electric Vallecitos Nuclear Center in California. The releases are based on damage scenarios developed by other specialists. The hazard severities presented range up to a nominal velocity of 230 mph for wind hazard and are in excess of 0.8 g linear acceleration for earthquakes. The consequences of thrust faulting are considered. The approaches and factors used to estimate the releases are discussed. Release estimates range from 0.003 to 3 g Pu.

664 (PNL--3683) ENVIRONMENTAL CONSEQUENCES OF POSTULATED PLUTONIUM RELEASES FROM GENERAL ELECTRIC COMPANY VALLECITOS NUCLEAR CENTER, VALLECITOS, CALIFORNIA, AS A RESULT OF SEVERE NATURAL PHENOMENA. Jamison, J.D.; Watson, E.C. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Nov 1980. Contract AC06-76RL01830. 61p. NTIS, PC A04/MF A01.

Potential environmental consequences in terms of radiation dose to people are presented for postulated plutonium releases caused by severe natural phenomena at the General Electric Company Vallecitos Nuclear Center, Vallecitos, California. The severe natural phenomena considered are earthquakes, tornadoes, and high straight-line winds. Maximum plutonium deposition values are given for significant locations around the site. All important potential exposure pathways are examined. The most likely 50-year committed dose equivalents are given for the maximum-exposed individual and the population within a 50-mile radius of the plant. The maximum plutonium deposition values likely to occur offsite are also given. The most likely calculated 50-year collective committed dose equivalents are all much lower than the collective dose equivalent expected from 50 years of exposure to natural background radiation and medical x-rays. The most likely maximum residual plutonium contamination estimated to be deposited offsite following the earthquakes, and the 180-mph and 230-mph tornadoes are above the Environmental Protection Agency's (EPA) proposed guideline for plutonium in the general environment of 0.2 $\mu Ci/m^2$. The deposition values following the 135-mph tornado are below the EPA proposed guidelines.

665 (PNL--3791) FACTORS AFFECTING CRITICALITY FOR SPENT-FUEL MATERIALS IN A GEOLOGIC SETTING. Gore, B.F.; Jenquin, U.P.; Serne, R.J. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Apr 1981. Contract AC06-76RL01830. 219p. NTIS, PC A10/MF A01. Order Number DE81029275.

Following closure of a geologic repository for spent fuel, geologic process may change geometries and spacings, and water may enter the repository. In this study the conditions required for the criticality of spent fuel constituents are determined. Many factors effect criticality, and the effects of various possible post-closure changes are investigated. Factors having the greatest effect on criticality are identified to provide guidance for research programs and for design and evaluation studies. Section II describes the calculational methods and computer codes used to determine critical conditions. Section III of this document addresses effects of the fissile content of spent fuel on criticality. Calculations have been performed to determine the minimum critical mass of spent fuel actinides as a function of the duration of in-reactor fuel exposure for a variety of possible conditions. Section IV addresses the conditions required for criticality under a scenario believed to be highly unlikely but having a unique possibility. Pu quantities and concentrations required for criticality without water were determined for various conditions of Pu separation, rock moderation and reflection, rock impurities and isotopic content of the Pu. Section V addresses the possibility of geochemical processes separating Pu from other spent fuel constituents. Solubilities of U and Pu are calculated for groundwaters characteristic of basalt, tuff, granite, bedded and dome salt. Maximum concentrations which could be adsorbed on geologic media in contact with these groundwaters are then calculated. Comparison of these maximum adsorbed concentrations with the results presented in Section IV yields the conclusion that criticality cannot occur in sorbed deposits of Pu in geologic media due to the low Pu concentrations achievable. The possibility of selective Pu precipitation, however, is not ruled out by these arguments.

666 (PNL--3796) COMPARATIVE ANALYSIS OF RISK CHARACTERISTICS OF NUCLEAR WASTE REPOSITORIES AND OTHER DISPOSAL FACILITIES. Lindell, M.K.; Earle, T.C.; Nealey, S.M. (Battelle Human Affairs Research Center, Seattle, WA (USA)). Jun 1981. Contract AC06-76RL01830. 109p. (B-HARC--411/81/005). NTIS, PC A06/MF A01. Order Number DE81030720.

Three fundamental questions concerning public perception of the measurement of radioactive wastes were addressed in this report. The first question centered on the perceived importance of nuclear waste management as a public issue: how important is nuclear waste management relative to other technological and scientific issues; do different segments of the public disagree on its importance; the second question concerned public attitudes toward a nuclear waste disposal facility: how great a risk to health and safety is a nuclear waste disposal facility relative to other industrial facilities; is there disagreement on its riskiness among various public groups; the third question pertained to the aspects of risks that affect overall risk perception: what are the qualitative aspects of a nuclear waste disposal facility that contribute to overall perceptions of risk; do different segments of the population associate different risk characteristics with hazardous facilities. The questions follow from one another: is the issue important; given the importance of the issue, is the facility designed to deal with it considered risky; given the riskiness of the facility, why is it considered risky. Also addressed in this report, and a main focus of its findings, were the patterns of differences among respondent groups on each of these questions.

667 (PNL--3950) ENVIRONMENTAL CONSEQUENCES OF POSTULATE PLUTONIUM RELEASES FROM ATOMICS INTERNATIONAL'S NUCLEAR MATERIALS DEVELOPMENT FACILITY (NMDF), SANTA SUSANA, CALIFORNIA, AS A RESULT OF SEVERE NATURAL PHENOMENA. Jamison, J.D.; Watson, E.C. (Pacific Northwest Lab., Richland, WA (USA)). Feb 1982. Contract AC06-76RL01830. 59p. NTIS, PC A04/MF A01. Order Number DE82008473.

Potential environmental consequences in terms of radiation dose to people are presented for postulated plutonium releases caused by severe natural phenomena at the Atomics International's Nuclear Materials Development Facility (NMDF), in the Santa Susana site, California. The severe natural phenomena considered are earthquakes, tornadoes, and high straight-line winds. Plutonium deposition values are given for significant locations around the site. All important potential exposure pathways are examined. The most likely 50-year committed dose equivalents are given for the maximum-exposed individual and the population within a 50-mile radius of the plant. The maximum plutonium deposition values likely to occur offsite are also given. The most likely calculated 50-year collective committed dose equivalents are all much lower than the collective dose equivalent expected from 50 years of exposure to natural background radiation and medical x-rays. The most likely maximum residual plutonium contamination estimated to be deposited offsite following the earthquake, and the 150-mph and 170-mph tornadoes are above the Environmental Protection Agency's (EPA) proposed guideline for plutonium in the general environment of 0.2 $\mu Ci/m^2$. The deposition values following the 110-mph and the 130-mph tornadoes are below the EPA proposed guideline.

668 (PNL--4072) SURVEY OF SYSTEMS SAFETY ANALYSIS METHODS AND THEIR APPLICATION TO NUCLEAR WASTE MANAGEMENT SYSTEMS. Pelto, P.J.; Winegardner, W.K.; Gallucci, R.H.V. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Nov 1981. Contract AC06-76RL01830. 114p. NTIS, PC A06/MF A01. Order Number DE82005594.

This report reviews system safety analysis methods and examines their application to nuclear waste management systems. The safety analysis methods examined include expert opinion, maximum credible accident approach, design basis accidents approach, hazard indices, preliminary hazards analysis, failure modes and effects analysis, fault trees, event trees, cause-consequence diagrams, GO methodology, Markov modeling, and a general category of consequence analysis models. Previous and ongoing studies on the safety of waste management systems are discussed along with their limitations and potential improvements. The major safety methods and waste management safety related studies are surveyed. This survey provides information on what safety methods are available, what waste management safety areas have been analyzed, and what are potential areas for future study.

669 (PNL--4099) ENVIRONMENTAL CONSEQUENCES OF POSTULATED RADIONUCLIDE RELEASES FROM THE BATTELLE MEMORIAL INSTITUTE COLUMBUS LABORATORIES JN-1B BUILDING AT THE WEST JEFFERSON SITE AS A RESULT OF SEVERE NATURAL PHENOMENA. Jamison, J.D.; Watson, E.C. (Pacific Northwest Lab., Richland, WA (USA)). Feb 1982. Contract AC06-76RL01830. 59p. NTIS, PC A04/MF A01. Order Number DE82008470.

Potential environmental consequences in terms of radiation dose to people are presented for postulated radionuclide releases caused by severe natural phenomena at the Battelle Memorial Institute Columbus Laboratories JN-1b Building at the West Jefferson site. The severe natural phenomena considered are earthquakes, tornadoes, and high straight-line winds. Maximum radioactive material deposition values are given for significant locations around the site. All important potential exposure pathways are examined. The most likely 50-year committed dose equivalents are given for the maximum-exposed individual and the population within a 50-mile radius of the plant. The maximum radioactive material deposition values likely to occur offsite are also given. The most likely calculated 50-year collective committed dose equivalents are all much lower than the collective dose equivalent expected from 50 years of exposure to natural background radiation and medical x-rays. The most likely maximum residual plutonium contamination estimated to be deposited offsite following the events are well below the Environmental Protection Agency's (EPA) proposed guideline for plutonium in the general environment of 0.2 $\mu Ci/m^2$. The likely maximum residual contamination from beta and gamma emitters are far below the background produced by fallout from nuclear weapons tests in the atmosphere.

670 (PNL-SA--8257, pp 1-3) WISAP RELEASE SCENARIO ANALYSIS WORKSHOP. Benson, G.L. Feb 1981. NTIS, PC A08/MF A01.

From WISAP release scenario analysis workshop; Seattle, WA, USA (13 Sep 1979).

Principles, practices, and procedures are being developed to assess the safety of the repository over a long time. To do that there is a need to come up with new methodology. Two basic assumptions are made, that there will be no human mitigation of a breach nor is there accounting for engineered barriers. Besides developing a methodology, a number of other parts of the waste management safety assessment problems are being dealt with and these are: helping provide criteria for site selection and other related matters; providing input on various reference-site analysis that are currently going on, for example, the salt-dome site; looking at several geologic media including bedded salts, salt domes, and basalt, and plan to start looking at granite next year.

671 (PNL-SA--8257, pp 119-120) PACIFIC NORTHWEST LABORATORY, WASTE ISOLATION SAFETY ASSESSMENT PROGRAM. Harwell, M.A. Feb 1981. NTIS, PC A08/MF A01.

From WISAP release scenario analysis workshop; Seattle, WA, USA (13 Sep 1979).

Some of the areas of interest of Pacific Northwest Laboratory's waste isolation safety assessment program are: adding the statistics part of the scenario model and completing the model based upon basalt geology; reducing computer time; reducing time steps as the simulation goes along; start thinking of how to incorporate the waste-induced aspects into the model; begin looking at (but not in terms of the model) the human-intrusion aspect; incorporating geochemical changes.

672 (PNL-SA--8831) COMPARISON OF POTENTIAL RADIOLOGICAL CONSEQUENCES FROM A SPENT-FUEL REPOSITORY VERSUS NATURAL-URANIUM DEPOSITS. Cloninger, M.O.; Wick, O.J. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Dec 1980. Contract AC06-76RL01830. 15p. (CONF-810217--19). NTIS, PC A02/MF A01. Order Number DE81028232.

From ANS waste management conference; Tucson, AZ, USA (23 Feb 1981).

On the basis of the intrinsic properties comparison, the long-term hazard from spent fuel in a geologic repository is comparable to that of a large ore deposit, although the specific nuclide concentration in an individual fuel element is not duplicated in nature. On the basis of the second comparison, a repository constructed within reasonable constraints presents no greater hazard than a large ore deposit. Even without such constraints, however, the natural hazard due to some observed radioactive releases to the biosphere in the United States far exceeds any that could reasonably be expected from a spent-fuel repository. While at first intuition it is reasonable to use natural ore deposits as a basis for criteria for nuclear waste repositories, the variability in the natural system is so great that it does not allow a specific criterion to be stated in absolute terms. It is suggested that if the naturally radioactive environment is to be used as a basis for a criterion regarding repositories, the criterion should be based on the radiological quality of the environment in the immediate region of a specific repository, and should be in terms of an acceptable potential increase in that radiological content due to the existence of the repository. This criterion could then be used as a basis for developing, through experimentation and modeling, specific criteria for a repository system design that is reasonably assured of meeting the environmental criteria.

673 (PNL-SA--9215) IMPROVED RADON-FLUX-MEASUREMENT SYSTEM FOR URANIUM-TAILINGS PILE MEASUREMENT. Freeman, H.D. (Pacific Northwest Lab., Richland, WA (USA)). Oct 1981. Contract AC06-76RL01830. 6p. (CONF-8110111--3). NTIS, PC A02/MF A01. Order Number DE82003848.

From International conference on radiation hazards in mining: control, measurement, and medical aspects; Golden, CO, USA (4 Oct 1981).

The Pacific Northwest Laboratory (PNL) is developing cover technology for uranium mill tailings that will inhibit the diffusion of radon to the atmosphere. As part of this cover program, an improved radon flux measurement system has been developed. The radon measurement system is a recirculating, pressure-balanced, flow-through system that uses activated carbon at ambient temperatures to collect the radon. With the system, an area of 0.93 m^2 is sampled for periods ranging from 1 to 12 h. The activated carbon is removed from the radon trap and the collected radon is determined by counting the ^{214}Bi daughter product. Development of the system included studies to determine the efficiency of activated carbon, relative calibration measurements and field measurements made during 1980 at the inactive tailings pile in Grand Junction, Colorado. Results of these studies are presented.

674 (PNL-SA--9894) AEGIS METHODOLOGY DEMONSTRATION: CASE EXAMPLE IN BASALT. Dove, F.H. (Pacific Northwest Lab., Richland, WA (USA)). Jan 1982. Contract AC06-76RL01830. 14p. (CONF-820303--15). NTIS, PC A02/MF A01. Order Number DE82008605.

From Waste management conference; Tucson, AZ, USA (8 Mar 1982).

The AEGIS technology has been successfully demonstrated. For the same data, similar unpublished results have been obtained by RHO and INTERA Environmental Consultants, Inc. for contaminant transport. In addition to establishing the utility of computer codes and assessment methodology, the AEGIS technology demonstration in basalt has also produced some practical guidance for future field data gathering programs. The results of this basalt demonstration indicate that the geohydrologic systems separating the nuclear waste from the natural biosphere discharge site mitigate the consequences of the postulated fault intersection event. This analysis suggests that the basalt system satisfies the 1000- and 10,000-yr proposed standards for release to the accessible environment (limited release of ^{129}I and ^{14}C). The reader should be cautioned, however, that the results are valid only for one particular set of parameters and one postulated release scenario. A complete sensitivity analysis must be performed to evaluate the range of effects that might be observed under different release conditions and for the different range in parameters.

675 POTENTIAL AERODYNAMIC ENTRAINMENT OF SOIL FROM EXCAVATION OPERATIONS IN A CONTAMINATED AREA. Sutter, S.L. (Battelle Pacific Northwest Lab., Richland, WA). Nuclear and Chemical Waste Management; 2: No. 1, 39-42(1981).

Experiments were performed to provide an indication of the amount of material that could be made airborne during soil-working operations in a contaminated area. Contaminated soil was collected, dried, and mixed, and particle size distribution and contamination levels characterized. In a 0.6 x 0.6 m wind tunnel, soil was pumped into an airstream moving at 1.4, 4.6, 6.8, and 8.9 m/s. Airborne mass and particle size samples were collected. The fraction of source soil made airborne increased as a linear function of wind speed. Airborne contamination increased with wind speed at a rate about 4 times the soil mass. Source terms (i.e., fraction of material initially made airborne) can be estimated using the information developed. These can, in turn, be used in diffusion models to estimate downwind contamination levels as required for safety assessments of radioactive or chemical waste sites. 3 figures, 2 tables.

676 COMPUTERIZED SIMULATION OF NUCLEAR WASTE REPOSITORIES IN GEOLOGIC MEDIA FOR RELEASE SCENARIO DEVELOPMENT. Zellmer, J.T. (Battelle Pacific Northwest Labs., Richland, WA (USA)). pp 69-78 of Proceedings of the workshop on radionuclide release scenarios for geologic repositories, Paris, 8-12 Sep 1980. Paris, France; OECD (1981).

From International workshop on radionuclide scenarios for geologic repositories; Paris, France (8 Sep 1980).

The long-term, post-closure safety of nuclear waste repositories located in geologic media must be assessed prior to licensing. The safety assessment must include the identification and bounding of natural phenomena that may potentially result in the release of radionuclides to the biosphere. Pacific Northwest Laboratory is developing a computerized methodology to assist in this assessment. The methodology determines and characterizes the various ways that the geologic and hydrologic systems surrounding the repository may evolve over geologic time. This process allows for the identification of events or sequences of events that may cause a release of radionuclides and establishes the initial conditions for release-consequence analyses. The purpose of the methodology is not to predict future geologic states, but rather to characterize and bound probable geologic evolutionary paths that may lead to the release of radionuclides from a nuclear waste repository.

677 WASTE ISOLATION SAFETY ASSESSMENT PROGRAMME. Brandstetter, A.; Harwell, M.A. (Battelle Pacific Northwest Labs., Richland, WA (USA)). pp 423-434 of Underground disposal of radioactive wastes. Proceedings of a symposium jointly organized by the IAEA and the OECD NEA and held at Otaniemi, Finland, 2-6 July 1979. Vienna, Austria; International Atomic Energy Agency (1980).

From International symposium on underground disposal of radioactive wastes; Otaniemi, Finland (2 Jul 1979).

Associated with commercial nuclear power production in the USA is the generation of potentially hazardous radioactive wastes. The Department of Energy (DOE), through the National Waste Terminal Storage (NWTS) Programme, is seeking to develop nuclear waste isolation systems in geologic formations that will preclude contact with the biosphere of waste radionuclides in concentrations which are sufficient to cause deleterious impact on humans or their environments. Comprehensive analyses of specific isolation systems are needed to assess the expectations of meeting that objective. The Waste Isolation Safety Assessment Programme (WISAP) has been established at the Pacific Northwest Laboratory (operated by Battelle Memorial Institute) for developing the capability of making those analyses. Among the analyses required for isolation system evaluation is the detailed assessment of the post-closure performance of nuclear waste repositories in geologic formations. This assessment is essential, since it is concerned with aspects of the nuclear power programme which previously have not been addressed. Specifically, the nature of the isolation systems (e.g. involving breach scenarios and transport through the geosphere), and the time-scales necessary for isolation, dictate the development, demonstration and application of novel assessment capabilities. The assessment methodology needs to be thorough, flexible, objective, and scientifically defensible. Further, the data utilized must be accurate, documented, reproducible, and based on sound scientific principles.

678 DEVELOPMENT AND TESTING OF LONG-TERM SAFETY ASSESSMENT METHODOLOGY FOR GEOLOGIC WASTE REPOSITORIES. Brandstetter, A. (Battelle Pacific Northwest Labs., Richland, WA). Transactions of the American Nuclear Society ; 33: 140(1979). (CONF-791103--).

From American Nuclear Society meeting; San Francisco, CA, USA (12 Nov 1979).

RADIOACTIVE WASTE STORAGE;UNDERGROUND DISPOSAL;SAFETY;RISK ASSESSMENT;RADIOACTIVE WASTE DISPOSAL;GEOLOGIC DEPOSITS;ENVIRONMENTAL EXPOSURE PATHWAY;RADIOACTIVE WASTE FACILITIES; HIGH-LEVEL RADIOACTIVE WASTES

679 ANALYTIC METHODS FOR FUEL-CYCLE SAFETY STUDIES. Smith, T.H.; Hall, R.J.; Williams, L.D. (Battelle-Northwest, Richland, Wash). IEEE (Institute of Electrical and Electronics Engineers) Transactions on Reliability; R-25: No. 3, 184-190(Aug 1976).

Battelle-Northwest (BNW) is conducting safety studies of the following nuclear fuel cycle operations: reprocessing, transportation of radioactive material, decommissioning of facilities, and waste management. Various methods and depths of analysis are used in these studies, but all involve safety quantification in terms of risk, relating probabilities and consequences of potential accidents. This paper describes the methods used for these safety studies. Highlighted are areas in which BNW contributions have extended the analytic methods: a risk-based fault-tree analytic method for identification, preliminary evaluation, and screening of potential release sequences; improved treatment of deposition and resuspension in airborne transport of radionuclides; and groundwater transport models of radionuclide chains in soil columns, including the effects of convection, diffusion, sorption, and generation and decay. Areas needing additional development are identified. 42 refs.

Regulations

Accountability and Safeguards

680 (PB--282169) CONSIDERATIONS FOR SAMPLING NUCLEAR MATERIALS FOR SNM ACCOUNTING MEASUREMENTS. SPECIAL NUCLEAR MATERIAL ACCOUNTABILITY REPORT. Brouns, R.J.; Roberts, F.P.; Upson, U.L. (Battelle Pacific Northwest Labs., Richland, WA (USA)). May 1978. Contract ERDA-EY-76-C-06-1830. 43p. NTIS PC A03/MF A01.

This report presents principles and guidelines for sampling nuclear materials to measure chemical and isotopic content of the material. Development of sampling plans and procedures that maintain the random and systematic errors of sampling within acceptable limits for SNM(Special Nuclear Materials) accounting purposes are emphasized.

681 IN-SITU TRANSURANIUM ELEMENT MEASUREMENT TECHNIQUE FOR WASTES ASSOCIATED WITH POWER REACTOR FUELS. Nielson, K.K.; Brodzinski, R.L.; Wogman, N.A. (Battelle, Pac Northwest Lab, Richland, Wash). National Bureau of Standards (U.S.), Special Publication; 201-206(May 1978). (CONF-780522--).

From ANS topical meeting; Williamsburg, VA, USA (15 May 1978).

A planar, 19 cmsup 2 intrinsic germanium detector has been used for in-situ analysis of plutonium and americium in contaminated laboratories and buildings. Detection limits depend on local background activity, but in typical surface measurements for decontamination work are about 0.005 nCi/cmsup 2 for sup 241Am (59.5 keV) and 0.5 nCi/cmsup 2 for sup 239Pu (17.2 keV). Specific analyses of sup 238Pu, sup 239Pu, sup 240Pu, and sup 241Pu are also possible using various gamma-rays. Attenuations equivalent to 10 cm of concrete can be tolerated for high levels of sup 239Pu and sup 241Am. 3 refs.

682 INSTRUMENT FOR MONITORING THE TRANSURANIC CONTENT OF CHOPPED LEACHED HULLS FROM SPENT NUCLEAR FUEL ELEMENTS. Wogman, N.A.; Brodzinski, R.L.; Brown, D.P. (Battelle, Northwest Lab, Richland, Wash). National Bureau of Standards (U.S.), Special Publication; 284-288(May 1978). (CONF-780522--).
From ANS topical meeting; Williamsburg, VA, USA (15 May 1978).
This communication describes a neutron detection system for the direct measurement of the transuranic elements in the hulls of chopped leached fuel elements and some associated effluent wastes. The monitor detects neutrons arising from spontaneous fission and from (alpha,n) reactions on oxygen in the hulls and wastes. The system is constructed using a massive external neutron shield, an internal lead gamma-ray shield between the sample and the neutron detectors, and an electronic system which records all single and coincidence neutron events which occur during a preset time interval. Both the transuranic neutron flux and the cosmic-ray produced neutron background are determined simultaneously. The estimated detection limit of the system is 2 mg of Pu in a 10sup 4 second counting period.

Administrative and Regulatory

683 (PNL--2666) ANALYSIS OF PRINT MEDIA COVERAGE OF NUCLEAR POWER ISSUES. Rankin, W.L.; Nealey, S.M.; Montano, D.E. (Battelle Human Affairs Research Center, Seattle, WA (USA)). Apr 1978. Contract EW-78-C-06-1076. 67p. Dep. NTIS, PC A04/MF A01.
224 newspaper articles were sampled from four newspapers, extrapolating to about 2850 nuclear power articles from 1972 to 1976. 124 magazine articles were found in four magazines (Time, Business Week, Scientific American, Environment). The newspapers wee found to run close coverage of nuclear power issues. The articles were mainly (45%) focused on reactors and reactor operation. The articles were judged to be antinuclear more often than pronuclear (only Scientific American discussed benefits more than costs). The same general conclusion was reached for articles on nuclear waste management. (DLC)

Radioactive Materials Monitoring and Transport

684 (BNWL--2207) DETAILED ACCIDENT ANALYSIS FOR THE HIGH LEVEL WASTE PREPARATION PHASE OF THE COMMERCIAL NUCLEAR WASTE VITRIFICATION PROJECT. Oscarson, E.E.; Mishima, J.; Waite, D.A. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Apr 1977. Contract EY-76-C-06-1830. 46p. Dep. NTIS, PC A03/MF A01.
The environmental consequences and the probability of conceivable accidents occurring during the high level waste preparation (HLWP) phase of the Commercial Nuclear Waste Vitrification Project (CNWVP) were analyzed. The maximum environmental consequences of postulated accidents were calculated to result in low radiation doses: a 50-year dose commitment of 0.3 rem to the whole body for a maximum individual, and of 20 man-rem to the whole body for the surrounding population. This may be compared to Department of Energy (DOE) values, in Manual Chapter Appendix 0524, of 0.5-rem whole-body annual dose commitment to individuals at points of maximum probable exposure in uncontrolled areas. The calculated whole-body relative dose risk to individuals from accidents is low (1.6×10^{-8} rem/yr) as compared to that received from natural background radiation (approximately 1.5×10^{-1} rem/yr) and as compared to levels specified in Nuclear Regulatory Commission (NRC) requirements for routine releases from nuclear power reactors (5.0×10^{-3} rem/yr). Therefore, the design and operation plans for the HLWP phase are judged not to represent an undue environmental risk from accident conditions.

685 (BNWL--2219) ENVIRONMENTAL ASSESSMENT METHODOLOGY FOR THE NUCLEAR FUEL CYCLE. Brenchley, D.L.; Soldat, J.K.; McNeese, J.A.; Watson, E.C. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jul 1977. Contract EY-76-C-06-1830. 153p. Dep. NTIS, PC A08/MF A01.
Separate abstracts were prepared for two sections of this report.

686 (BNWL--2219, pp 3-30) ASSESSMENT METHODOLOGY FOR RADIOACTIVE EFFLUENTS. Jul 1977.
Environmental assessment methodology for the nuclear fuel cycle.
The objective of this environmental assessment is to define and rank the needs for controlling radioactive effluents from nuclear fuel cycle facilities. The assessment is based on environmental standards and dose-to-man calculations. The study includes three calculations for each isotope from each facility: maximum individual dose for a 50-year dose commitment from a 1-yr exposure according to the organ affected; population dose for a 50-yr dose commitment from a 1-yr exposure according to the organ affected; and annual dose rate for the maximally exposed individual. The relative contribution of a specific nuclide and source to the total dose provides a method of ranking the nuclides, which in turn identifies the sources that should receive the greatest control in the future. These results will be used in subsequent tasks to assess the environmental impact of the total nuclear fuel cycle.

687 (BNWL-SA--6114) IMPROVED METHOD FOR CALCULATION OF POPULATION DOSES FROM NUCLEAR COMPLEXES OVER LARGE GEOGRAPHICAL AREAS. Corley, J.P.; Baker, D.A.; Hill, E.R.; Wendell, L.L. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Sep 1977. Contract EY-76-C-06-1830. 18p. (CONF-770720--16). Dep. NTIS, PC A02/MF A01.
From 22. annual meeting of the Health Physics Society; Atlanta, GA, USA (3 Jul 1977).
To simplify the calculation of potential long-distance environmental impacts, an overall average population exposure coefficient (P.E.C.) for the entire contiguous United States was calculated for releases to the atmosphere from Hanford facilities. The method, requiring machine computation, combines Bureau of Census population data by census enumeration district and an annual average atmospheric dilution factor (anti chi/Q') derived from 12-hourly gridded wind analyses provided by the NOAA's National Meteorological Center. A variable-trajectory puff-advection model was used to calculate an hourly anti chi/Q' for each grid square, assuming uniform hourly releases; seasonal and annual averages were then calculated. For Hanford, using 1970 census data, a P.E.C. of 2×10^{-3} man-seconds per cubic meter was calculated. The P.E.C. is useful for both radioactive and nonradioactive releases. To calculate population doses for the entire contiguous United States, the P.E.C. is multiplied by the annual average release rate

and then by the dose factor (rem/yr per Ci/m^3) for each radionuclide, and the dose contribution in man-rem is summed for all radionuclides. For multiple pathways, the P.E.C. is still useful, provided that doses from a unit release can be obtained from a set of atmospheric dose factors. The methodology is applicable to any point source, any set of population data by map grid coordinates, and any geographical area covered by equivalent meteorological data.

688 (EPRI-EA--2045) TRANSURANIUM AND OTHER LONG-LIVED RADIONUCLIDES IN THE TERRESTRIAL ENVIRONS OF NUCLEAR POWER PLANTS. FINAL REPORT. Robertson, D.E.; Thomas, C.W.; Perkins, R.W.; Thomas, V.W. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Sep 1981. Contract AC06-76RL01830. 181p. NTIS, PC A09/MF A01. Order Number DE82003074.

The purpose of this research was to ascertain the contribution of certain radioactive elements, especially transuranics, to the environment from nuclear power plants as opposed to that from fallout due to weapon testing. Of the radioactive nuclides, alpha emitter, e.g., plutonium, are considered the most hazardous to human health; hence, the extent to which power plants may release plutonium to the environment is a matter of concern. Samples of soil, plants, and air were taken at four nuclear power plants: Rancho Seco (California), Zion (Illinois), Browns Ferry (Alabama), and Quad-Cities (Illinois). For background purposes, samples were also collected at 10 sites in Idaho, Indiana, Mississippi, Missouri, Montana, North Dakota, Oklahoma, Texas, Washington, and West Virginia that were remote from nuclear plants. Samples were analyzed for plutonium, americium, and curium as well as for a number of other species. Emphasis was on plutonium because the ratio of ^{238}Pu to ^{239,240}Pu can be used to distinguish reactor plutonium from fallout plutonium. The power plant contribution to the environment of the elements tested is negligible. With plutonium, for example, the absolute concentration of the element differed greatly from plant to plant and even from site to site at a given plant; however, the ^{238}Pu/^{239,240}Pu ratio was remarkably constant and, with rare exceptions, could not be distinguished from the ratios of background samples. At a few localities, for some elements, a power plant contribution could be recognized. In such cases, despite a power plant contribution, the absolute concentration of the element was below a national background value.

689 (NUREG/CR--0523) MESODIF-II: A VARIABLE TRAJECTORY PLUME SEGMENT MODEL TO ASSESS GROUND-LEVEL AIR CONCENTRATIONS AND DEPOSITION OF EFFLUENT RELEASES FROM NUCLEAR POWER FACILITIES. Powell, D.C.; Wegley, H.L.; Fox, T.D. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Nov 1978. Contract EY-76-C-06-1830. 319p. (PNL--2419). Dep. NTIS, PC A14/MF A01.

MESODIF-II is a variable trajectory plume segment atmospheric transport model designed to predict normalized air concentrations and deposition of radioactive, but otherwise non-reactive, effluents released from one or two levels over the same position in an xy-plane. In such a model, calculated particle trajectories vary as synoptic scale wind varies. At all sampling times the particles are connected to form a segmented plume centerline; the lateral and vertical dimensions of the plume are determined by a parameterization of turbulence scale diffusion. The model development arose from the need to assess the radiological effects of routine operations of commercial nuclear power reactors. The purpose of the present document is to make MESODIF-II and its peripheral programs available to those who wish to use it directly. The theory and mathematics in the model are explained. A user's guide, which treats not only MESODIF-II descriptively, but also other programs that have to be used to generate the input wind and stability data is included. Two other peripheral programs that are used in conjunction with MESODIF-II are also described. One is Program Stagg, the other is Program RERITE. A complete test run of MESODIF-II is included, as is the complete code for the program and the control cards used on the CDC-6400. A glossary for the MESODIF-II code, and a reference list arealso provided.

690 (PNL--2432) MASTER SCHEDULE FOR CY-1978. HANFORD ENVIRONMENTAL SURVEILLANCE ROUTINE PROGRAM. Blumer, P.J.; Myers, D.A.; Fix, J.J. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Dec 1977. Contract EY-76-C-06-1830. 28p. Dep. NTIS, PC A03/MF A01.

This report provides the current schedule of data collection for the routine environmental surveillance program at the Hanford Site. No results are presented in this report. The data collected are available in routine reports issued by the Environmental Evaluations staff.

691 (PNL--2439) POPULATION DOSE COMMITMENTS DUE TO RADIOACTIVE RELEASES FROM NUCLEAR POWER PLANT SITES IN 1975. Baker, D.A.; Soldat, J.K.; Watson, E.C. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Oct 1977. Contract EY-76-C-06-1830. 133p. Dep. NTIS, PC A07/MF A01.

Population radiation dose commitments were estimated from reported radionuclide releases from commercial power reactors operating during 1975. Fifty-year dose commitments from one year exposure were calculated from both liquid and atmospheric releases for four population groups (infant, child, teenager and adult) residing between 2 and 80 km from each site. Results are given in the form of tables giving the dose commitments for both liquid and airborne pathways for each age group and organ. Also included for each site is a histogram showing the fraction of the total population within the 2 to 80-km region around each site receiving various average dose commitments from the airborne pathways. The total dose commitment from both liquid and airborne pathways ranged from a high of 750 person-rem to a low of 0.008 person-rem with an arithmetic mean of 34 person-rem.

692 (PNL--2500(Pt.5), pp 4.1-4.9) OPERATIONAL AND ENVIRONMENTAL SAFETY. Feb 1978.

Pacific Northwest Laboratory annual report for 1977 to the DOE Assistant Secretary for Environment. Part 5. Control technology, overview, health, safety and policy analysis.

The responsibility of the DOE Office of Operational and Environmental Safety is to assure that DOE-controlled activities are conducted in a manner that will minimize risks to the public and employees and will provide protection for property and the environment. The program supports the various energy technologies by identifying and resolving safety problems; developing and issuing safety policies, standards, and criteria; assuring compliance with DOE, Federal, and state safety regulations; and establishing procedures for reporting and investigating accidents in DOE operations. Guidelines for the radiation protection of personnel; radiation monitoring at nuclear facilities; an assessment of criticality accidents by fault tree analysis; and the preparation of environmental, safety, and health standards applicable to geothermal energy development are discussed.

693 (PNL--2614) ENVIRONMENTAL SURVEILLANCE AT HANFORD FOR CY 1977. Houston, J.R.; Blumer, P.J. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Apr 1978. Contract EY-76-C-06-1830. 58p. Dep. NTIS, PC A04/MF A01.

Environmental data collected during 1977 show continued compliance by Hanford with all applicable state and federal regulations. Data were collected for most environmental media including air, Columbia River water, external radiation, foodstuffs (milk, beef, eggs, poultry, and produce) and wildlife (deer, fish, game birds, and oysters from Willapa Bay), as well as soil and vegetation samples. In general, offsite levels of radionuclides attributable to Hanford operations during 1977 were indistinguishable from background levels.

694 (PNL--2670) RADIOLOGICAL AND TOXICOLOGICAL ASSESSMENT OF AN EXTERNAL HEAT (BURN) TEST OF THE 105MM CARTRIDGE, APFSDS-T, XM-744. Gilchrist, R.L.; Parker, G.B.; Mishima, J. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Mar 1978. Contract EY-76-C-06-1830. 37p. Dep. NTIS, PC A03/MF A01.

The potential radiological and toxicological hazard of depleted uranium aerosol release was investigated. This type of release might arise from accidents with XM-774 ammunition involving great heat. Twelve rounds of packaged ammunition were subjected to an external heat (burn) test. Examination of the site on the day following the test revealed that all 12 depleted uranium penetrators were completely intact. Oxidation of the penetrators was not apparent, even on the most severely burned projectile located at ground zero. Eleven of the 12 projectiles were recovered with the sabots intact; some sabots appeared charred. It was concluded that no airborne release of depleted uranium had occurred and subsequently there had been no radiological or toxicological hazard from DU during this test. However, this conclusion may not apply to the release of depleted uranium in other types of fires involving this ammunition because other factors may affect the fire. These factors include type of fuel, number of ammunition rounds, and type of structure housing the ammunition.

695 (PNL--2677) ENVIRONMENTAL STATUS OF THE HANFORD SITE FOR CY 1977. Houston, J.R.; Blumer, P.J. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jun 1978. Contract EY-76-C-06-1830. 63p. Dep. NTIS, PC A04/MF A01.

Environmental data collected during 1977 showed continued compliance of Hanford operations with all applicable state and federal regulations. Included in the environmental data collected were measurements of external radiation and radionuclide analysis of air samples, Columbia River water, wildlife, soil, vegetation, and surface waste waters. In addition, all roadways, railways, and active as well as retired burial grounds were surveyed periodically to detect any abnormal levels of radioactivity.

696 (PNL--2801) MASTER SCHEDULE FOR CY-1979 HANFORD ENVIRONMENTAL SURVEILLANCE ROUTINE PROGRAM. Blumer, P.J.; Houston, J.R.; Eddy, P.A. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Dec 1978. Contract EY-76-C-06-1830. 31p. Dep. NTIS, PC A03/MF A01.

The current schedule of data collection for the routine environmental surveillance program at the Hanford Site, as conducted by the Environmental Evaluation Section of Battelle, Pacific Northwest Laboratory for the Department of Energy (DOE), is given. Modifications to the schedule are made during the year and special areas of study, usually of short duration, are not scheduled. The environmental surveillance program objectives are to evaluate the levels of radioactive and nonradioactive pollutants in the Hanford environs, and to monitor Hanford operations for compliance with applicable environmental criteria and Washington State Water Quality Standards. Air quality data are obtained in a separate program administered by the Hanford Environmental Health Foundation. The collection schedule for potable water is shown but it is not part of the routine environmental surveillance program. Water quality data for Hanford Site potable water systems are published each year by the Hanford Environmental Health Foundation. The data collected are available in routine reports issued by the Environmental Evaluations staff. Groundwater data and evaluation are reported in the series, ''Radiological Status of the Groundwater Beneath the Hanford Project for...,'' the latest issue being PNL-2624 for CY-1977. Data from locations within the plant boundaries are presented in the annual ''Environmental Status of the Hanford Site for...'' report series, the most recent report being PNL-2677 for 1977. Data from offsite locations are presented in the annual ''Environmental Surveillance at Hanford for...'' series of reports, the latest being PNL-2614 for 1977.

697 (PNL--3128) SUPPLEMENTAL REPORT ON POPULATION ESTIMATES FOR HANFORD HIGH-LEVEL DEFENSE WASTE DRAFT PROGRAMMATIC ENVIRONMENTAL IMPACT STATEMENT. Yandon, K.E.; Landstrom, D.K. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jun 1980. Contract AC06-76RL01830. 31p. NTIS, PC A03/MF A01.

Current and revised population projections based on those previously published in the document Population Distribution in 90-mile Radius of Hanford Meteorological Station and Projections to Year 2300 by Compass Sector and 10 Mile Radii are presented. In addition, there was a need to extend the population estimates out to 1000 and 10,000 years into the future to permit estimation of population radiation doses from accidents affecting the Hanford Facilities directly related to the defense high-level waste disposal alternatives. The methodology used in making the estimates is presented along with the detailed population matrix data required for performing the dose calculations. Although the near-term overall population projections are probably reasonably correct, no claim is made for the accuracy of the detailed data within each individual sector. Long-term estimates are made using reasonable assumptions about the growth potential and possibilities in the Hanford area. No claim of accuracy of these figures is made since they are so highly dependent on actions and conditions that are not predictable. For example, if a major climate change were to occur, the entire Hanford area might be uninhabited at 10,000 years in the future. To provide conservative dose estimates, it was assumed that the Hanford population will experience reasonable and continuous growth throughout the 10,000 year period.

698 (PNL--3222) MASTER SCHEDULE FOR CY-1980 HANFORD ENVIRONMENTAL SURVEILLANCE ROUTINE PROGRAM. Blumer, P.J.; Houston, J.R.; Eddy, P.A. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Dec 1979. Contract EY-76-C-06-1830. 41p. Dep. NTIS, PC A03/MF A01.

The current schedule of data collection for the routine environmental surveillance program at the Hanford Site is presented. The enviromental surveillance program objectives are to evaluate the levels of radioactive and nonradioactive pollutants in the Hanford environs, as required in Manual Chapter 0513, and to monitor Hanford operations for compliance with applicable environmental criteria given in Manual Chapter 0524 and Washington State Water Quality Standards. Data are reported on the following topics: air; Columbia River; sanitary water; surface water; ground water; foodstuffs; wildlife; soil and vegetation; external radiation measurement; portable instrument surveys; and surveillance of waste disposal sites;

699 (PNL--3283) ENVIRONMENTAL SURVEILLANCE AT HANFORD FOR CY-1979. Houston, J.R.; Blumer, P.J. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Apr 1980. Contract AC06-76RL01830. 72p. NTIS, PC A04/MF A01.

Environmental data were collected for most environmental media including air, Columbia River water, external radiation, foodstuffs (milk, beef, eggs, poultry, and produce) and wildlife (deer, fish, and game birds), as well as soil and vegetation samples. In general, offsite levels of radionuclides attributable to Hanford operations during 1979 were indistinguishable from background levels. The data are summarized in the following highlights. Air quality measurements of NO_2 in the vicinity of the Hanford Site and releases of SO_2 onsite were well within the applicable federal and state standards. Particulate air concentrations exceed the standards primarily because of agricultural activities in the area. Discharges of waste water from Hanford facilities in the Columbia River under the National Pollution Discharge Elimination System (NPDES) permit were all within the perameter limits on the permit.

700 (PNL--3284) ENVIRONMENTAL STATUS OF THE HANFORD SITE FOR CY 1979. Houston, J.R.; Blumer, P.J. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Aug 1980. Contract AC06-76RL01830. 80p. NTIS, PC A05/MF A01.

The Hanford environmental surveillance program provides for the measurement, interpretation, and evaluation of the radiological impact of Hanford operations on its onsite and offsite environs. The program is designed to evaluate all significant potential pathways for the release of radioactivity, especially those that may result in direct exposure of the public and those in which environmental reconcentration of radionuclides is likely to occur. Presented in this report are the results of effluent and environmental monitoring and sampling conducted to determine the onsite impact of ongoing operations.

701 (PNL--3300(Pt.2), pp 147) WASTE ISOLATION SAFETY ASSESSMENT PROGRAM. Soldat, J.K.; Napier, B.A.; Strenge, D.L.; Watson, E.C. Feb 1980. Dep. NTIS, PC A11/MF A01.

Pacific Northwest Laboratory annual report for 1979 to the DOE Assistant Secretary for Environment. Part 2. Ecological sciences.

Several computer codes available for calculation of aerial radiation doses to people were evaluated and those suitable for the WISAP project were modified and converted to the new DOE/RL 1100/44 computer. The codes were used to calculate potential doses from consumption of salt obtained by solution mining of a salt dome. The potential radiation doses were significantly high.

702 (PNL--3300(Pt.2), pp 151) WASTE MANAGEMENT SAFETY STUDIES. Mellinger, P.J.; Hoenes, G.R.; Brackenbush, L.W.; Greenborg, J. Feb 1980. Dep. NTIS, PC A11/MF A01.

Pacific Northwest Laboratory annual report for 1979 to the DOE Assistant Secretary for Environment. Part 2. Ecological sciences.

Until now, all of t ^{85}Kr from nuclear fuel reprocessing has been intentionally released to the atmosphere; however, federal regulations for commercial fuel irradiated after 1982 will require collection and long-term storage of about 85% of the ^{85}Kr previously released during reprocessing. In this project, we compare individual and population radiation doses from the presently accepted practice of total release of ^{85}Kr with doses resulting from the operation and maintenance of krypton recovery, transportation and storage operations required to comply with the forthcoming federal regulations. Results indicate that it makes little difference to the world population dose whether ^{85}Kr is captured and stored or chronically released to the environment.

703 (PNL--3300(Pt.5), pp 71) HANDBOOKS ON EFFLUENT AND ENVIRONMENTAL MONITORING. Feb 1980. Dep. NTIS, PC A07/MF A01.

Pacific Northwest Laboratory annual report for 1979 to the DOE Assistant Secretary for Environment. Part 5. Environmental assessment, control, health, and safety.

Issue of the draft Guide for Effluent Monitoring at DOE Installations was postponed pending resolution of further regulatory requirements. Analysis and evaluation of CY-1978 annual environmental surveillance reports from Department of Energy (DOE) nuclear sites were approximately 50% completed. A draft Executive Summary of 1977 environmental impacts from all DOE nuclear sites was completed.

704 (PNL--3524) ALLDOS: A COMPUTER PROGRAM FOR CALCULATION OF RADIATION DOSES FROM AIRBORNE AND WATERBORNE RELEASES. Strenge, D.L.; Napier, B.A.; Pelquin, R.A.; Zimmerman, M.G. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Oct 1980. Contract AC06-76RL01830. 154p. NTIS, PC A08/MF A01.

The computer code ALLDOS is described and instructions for its use are presented. ALLDOS generates tables of radiation doses to the maximum individual and the population in the region of the release site. Acute or chronic release of radionuclides may be considered to airborne and waterborne pathways. The code relies heavily on data files of dose conversion factors and environmental transport factors for generating the radiation doses. A source inventory data library may also be used to generate the release terms for each pathway. Codes available for preparation of the dose conversion factors are described and a complete sample problem is provided describing preparation of data files and execution of ALLDOS.

705 (PNL--3639) MASTER SCHEDULE FOR CY-1981 HANFORD ENVIRONMENTAL SURVEILLANCE ROUTINE PROGRAM. Blumer, P.J.; Sula, M.J.; Eddy, P.A. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Dec 1980. Contract AC06-76RL01830. 41p. NTIS, PC A03/MF A01.

The current schedule of data collection for the routine environmental surveillance program at the Hanford Site is provided. Questions about specific entries should be referred to the authors since modifications to the schedule are made during the year and special areas of study, usually of short duration, are not scheduled. The environmental surveillance program objectives are to evaluate the levels of radioactive and nonradioactive pollutants in the Hanford environs, as required in Manual Chapter 0513, and to monitor Hanford operations for compliance with applicable environmental criteria given in Manual Chapter 0524 and Washington State Water Quality Standards. Air quality data obtained in a separate program are also reported. The collection schedule for potable water is shown but it is not part of the routine environmental surveillance program. Schedules are presented for the following subjects: air, Columbia River, sanitary water, surface water, ground water, foodstuffs, wildlife, soil and vegetation, external radiation measurement, portable instrument surveys, and surveillance of waste disposal sites. (JGB)

706 (PNL--3728, pp 33-37) EXTERNAL RADIATION. Apr 1981. NTIS, PC A05/MF A01.

Environmental surveillance at Hanford for CY-1980.

External radiation levels were measured using thermoluminescent dosimeters at all air sampling locations in the Hanford environs. Dosimeters were also used to measure the dose rates along the Columbia River islands and shoreline near the Hanford Site and the immersion dose in Columbia River water upstream and downstream of Hanford liquid effluent discharge points. Measurements during 1980 at the air sampling stations could not distinguish any difference in external dose rates between the perimeter and distant stations. The number of Columbia River Island and Shoreline dose measurement locations was increased during 1980 to include areas with the highest dose rates measured during an extensive radiological survey of the river in 1979. Based on measurements at these locations, the maximum potential increase in radiation dose to recreational users of the river was 3 mrem/yr. The radioactivity causing these dose rates is from pre-1971 operation of production reactors at Hanford.

707 (PNL--3728, pp 39-43) RADIOLOGICAL IMPACT OF HANFORD OPERATIONS. Apr 1981. NTIS, PC A05/MF A01.

Environmental surveillance at Hanford for CY-1980.

The radiological impact of Hanford operations during 1980 was calculated based upon the quantity of radionuclides measured in effluents from operating facilities. The first-year dose to the hypothetical maximum-exposed individual was calculated to be 0.72 mrem to the thyroid, 0.5% of the applicable DOE Radiation Protection Standard. The 50-yr whole-body dose commitment to the population within an 80-km (50-mile) radius of the Hanford Site was calculated to be 0.60 man-rem. A comparison of the estimated potential impact from 1980 Hanford operations with the impact from other sources of radiation exposure routinely encountered demonstrates the comparatively small impact of current operations.

708 (PNL--4120) MASTER SCHEDULE FOR CY-1982 HANFORD ENVIRONMENTAL SURVEILLANCE ROUTINE PROGRAM. Blumer, P.J.; Sula, M.J.; Eddy, P.A. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Dec 1981. Contract AC06-76RL01830. 44p. NTIS, PC A03/MF A01. Order Number DE82005722.

This report provides the current schedule of data collection for the routine environmental surveillance program at the Hanford Site. The environmental surveillance program objectives are to evaluate and report the levels of radioactive and nonradioactive pollutants in the Hanford environs, as required in DOE Order 5484.1. The routine sampling schedule provided does not include samples which are planned to be collected during FY-1982 in support of special studies or for quality control purposes. In addition, the routine program outlined in this schedule is subject to modification during the year in response to changes in Site operations, program requirements, or unusual sample results. Sampling schedules are presented for the following: air; Columbia River; sanitary water; surface water; ground water; foodstuffs; wildlife; soil and vegetation; external radiation measurements; portable instrument surveys; and surveillance of waste disposal sites. (ATT)

709 (PNL-SA--6771) AIRBORNE PLUTONIUM-239 AND AMERICIUM-241 CONCENTRATIONS MEASURED FROM THE 125-METER HANFORD METEOROLOGICAL TOWER. Sehmel, G.A. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jan 1978. Contract EY-76-C-06-1830. 15p. (CONF-780212--7). Dep. NTIS, PC A02/MF A01.

From Plutonium information conference; San Diego, CA, USA (28 Feb 1978).

Airborne plutonium-239 and americium-241 concentrations and fluxes were measured at six heights from 1.9 to 122 m on the Hanford meteorological tower. The data show that plutonium-239 was transported on nonrespirable and small particles at all heights. Airborne americium-241 concentrations on small particles were maximum at the 91 m height.

710 (PNL-SA--6832) CHARACTERIZING DISPERSION ON A CLIMATOLOGICAL BASIS. Sandusky, W.F.; Nickola, P.W. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jun 1978. Contract EY-76-C-06-1830. 18p. (CONF-7806109--2). Dep. NTIS, PC A02/MF A01.

From American Association for the Advancement of Science; Seattle, WA, USA (13 Jun 1978).

Normalized concentrations of pollutants downwind of various nuclear power reactor sites have been predicted with onsite meteorological data and a computer code developed by the U.S. Nuclear Regulatory Commission (Sagendorf and Goll, 1977). These results, grouped by calendar year of meteorological data, were compared and the maximum concentrations were found to vary by a factor of approx. 5 between groups and approx. 3 within groups. Mean values of normalized concentrations of pollutant for each group were found to vary by a factor of approx. 2. Results of this study confirm earlier analysis by Hosler (1964) which indicated that differences in atmospheric dilution among data sites, based on the average effects of wind speed and vertical thermal stability, are small.

711 (PNL-SA--8033) DIFFERENTIAL MONITORING

OF TRITIUM AND CARBON-14 COMPOUNDS. Goles, R.W.; Brauer, F.P. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Apr 1980. Contract AC06-76RL01830. 17p. (CONF-800427--22). NTIS, PC A02/MF A01.

From Tritium technology in fission, fusion, and isotopic applications; Dayton, OH, USA (29 Apr 1980).

A gaseous sampling system was developed to differentially collect all major volatile forms of tritium and carbon-14 according to chemical class. These chemical forms include: tritiated forms of water, hydrogen and organics; as well as ^{14}C-containing carbon monoxide, carbon dioxide and organics. Sampling campaigns involving the use of this differential ^{3}H and ^{14}C collection system have been successfully conducted at a high level liquid waste solidification plant, at a spent fuel storage facility and in the vicinity of power reactors.

712 (PNL-SA--9140) ENVIRONMENTAL CONCENTRATION AND MIGRATION OF ^{129}I. Brauer, F.P.; Strebin, R.S. Jr. (Pacific Northwest Lab., Richland, WA (USA)). 1981. Contract AC06-76RL01830. 22p. (CONF-810722--12). NTIS, PC A02/MF A01. Order Number DE81030093.

From International symposium on migration in the terrestrial environment of long-lived radionuclides from the nuclear fuel cycle; Knoxville, TN, USA (27 Jul 1981).

Local sources of ^{129}I suitable for study of migration in the environment are uranium ore bodies for long periods and nuclear fuel reprocessing operations for recent time periods. Uranium ores produce ^{129}I from spontaneous and neutron-induced fission. Nuclear fuel reprocessing has resulted in releases of ^{129}I to both the atmosphere and the hydrosphere. The long term fate of surface-deposited and groundwater ^{129}I was studied. Neutron activation analysis was used to measure the low levels of ^{129}I and natural ^{127}I in samples near and remote from nuclear fuel reprocessing facilities. Measurements were made on air samples, vegetation, soil, surface water, groundwater, and uranium ore samples. The ^{129}I was found to accumulate in the top soil and litter layer. The surface concentrations of ^{129}I in forest communities were found to be several times higher than for nearby grass communities. The ^{129}I levels were found to persist in the environment after reprocessing activities had been discontinued. Measurable ^{129}I was found to be contained in uranium ore samples from a variety of locations and with a range of uranium concentrations. Thus ore bodies constitute a source of ^{129}I suitable for study of migration over long time periods. Iodine-129 has entered surface and groundwater as the result of liquid and atmospheric discharges from nuclear fuel reprocessing facilities. The iodine in water samples was found to be anionic and travel with the water flow. Thus ^{129}I was found to provide high sensitivity for hydrology tracer studies and radioactivity migration surveillance. Once ^{129}I enters the hydrosphere it migrates with the water with little or no hold-up by exchange with cationic exchangers such as soil or silt.

713 (PNL-SA--9318) INDEPENDENT DETERMINATION OF THE ACCURACY OF A RESEARCH REACTOR STACK-GAS MONITOR. Pickett, B.D.; Johnson, A.G. (Battelle Pacific Northwest Labs., Richland, WA (USA); Oregon State Univ., Corvallis (USA). Radiation Center). Feb 1981. Contract AC06-76RL01830. 12p. (CONF-810644--1). NTIS, PC A02/MF A01. Order Number DE81028194.

From 5. international workshop on rare earth-cobalt magnets and their application; Dayton, OH, USA (6 Jun 1981).

During normal operation of the Oregon State University TRIGA reactor (OSTR), argon-41 is produced and released to the atmosphere as a gaseous effluent through the reactor building exhaust stack. The reactor facility's stack gas monitor (SGM) continuously samples the effluent in the exhaust stack and simultaneously measures the concentration of argon-41 being released. This study was undertaken to determine the accuracy of the stack gas monitor, using techniques which were independent of the monitoring system itself. The results indicated that the argon-41 concentrations displayed by the stack gas monitor were approximately 50% of those predicted by analysis of individual samples from the exhaust system.

714 TRANSURANIC WASTES FROM THE COMMERCIAL LIGHT-WATER-REACTOR CYCLE. Kreiter, M.R.; Mendel, J.E.; McKee, R.W. (Battelle Pacific Northwest Lab., Richland, WA). pp 92-106 of Transuranic elements in the environment. Hanson, W.C. (ed.). Oak Ridge, TN: Technical Information Center (1980).

Airborne and transuranic-contaminated wastes generated in postfission activities are identified by quantity and radioactivity for the case in which spent fuel is declared waste (once-through cycle) and that in which spent fuel is reprocessed and the recovered uranium and plutonium are recycled. Because no standard defining transuranic wastes is available at this time, in this chapter the waste source is used as the basis for such a definition. Radioactive wastes are generally treated to reduce their volume and/or mobility. For convenience the radioactive wastes discussed are categorized according to the treatment they require. A selected treatment process as well as the final treated volume is presented for each of the seven categories of waste.

715 (BNWL--1794) DISTRIBUTION OF RADIOACTIVE JACKRABBIT PELLETS IN THE VICINITY OF THE B-C CRIBS, 200 EAST AREA, USAEC HANFORD RESERVATION. O'Farrell, T.P.; Fitzner, R.E.; Gilbert, R.O. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Sep 1973. Contract EY-76-C-06-1830. 79p. Dep. NTIS, PC A04/MF A01.

During 1972 and 1973 a study was conducted in the Hanford B-C Cribs, 200 East Area, to learn the extent to which jackrabbits (Lepus californicus) and their predators had dispersed buried radioactive wastes in their fecal pellets and scats. The specific objective was to gather sufficient data on the pattern of dispersal so that statistically valid sampling strategies could be developed in future programs, depending upon management planning objectives for the area. A secondary objective was to relate these data with parameters, such as topography, wind direction, vegetation types, animal behavior, that might help explain the pattern of dispersal. In 1972, 2625 circular sampling sites were surveyed along 30 transects radiating out 2.4 to 3.2 km from the B-C Cribs. Radioactive contaminated feces, urine, soil and vegetation were distributed in all directions from the cribs, but the area to the south and southwest was more densely and uniformly contaminated. Of the ultimate sampling units surveyed, 278 or 10.6% had activity in excess of 10,000 counts per minute (cpm) measured with a Geiger-Mueller counter.

716 (BNWL--2117, pp 505-530) FIELD AND LABORATORY OBSERVATIONS ON PLUTONIUM OXIDATION STATES. Bondietti, E.A.; Reynolds, S.A.

1976.

From Conference on actinide sediment reaction; Seattle, WA, USA (10 Feb 1976).

Test methods for evaluating Pu oxidation state species in experimental samples were developed and tested. These methods, which are based on solvent extraction and coprecipitation principles, enabled the elucidation of Pu(VI) and Pu(IV) monomeric species in laboratory soil-water equilibration studies, plutonium dioxide dissolution studies, and in seepage solution collected near a solid waste burial ground. The methods employed also allowed the identification of soluble polymeric Pu species which do not revert to monomeric forms upon acidification.

717 (BNWL--2117, pp 39-71) ACTINIDE OCCURRENCES IN SEDIMENTS FOLLOWING GROUND DISPOSAL OF ACID WASTES AT 216-Z-9. Ames, L.L. 1976.

From Conference on actinide sediment reaction; Seattle, WA, USA (10 Feb 1976).

Liquid acid wastes from a Pu recovery facility at Hanford were released to the ground via structures collectively termed trenches from 1955 through 1962. Data are presented from a study of the microdistribution of Am and Pu in samples from the 216-Z-9 trench. Solution sediment relationships and associated actinide removal mechanisms under acid conditions were studied. Core wells were drilled into the sediments in which this covered trench is located and in the immediate vicinity to obtain samples for quantitative mineralogical analysis and comparison of sediments from various depths of contaminated and noncontaminated areas. Analytical techniques are described and results are reported.

718 (BNWL--2127) MULTICOMPONENT MASS TRANSPORT MODEL: THEORY AND NUMERICAL IMPLEMENTATION (DISCRETE-PARCEL-RANDOM-WALK VERSION). Ahlstrom, S.W.; Foote, H.P.; Arnett, R.C.; Cole, C.R.; Serne, R.J. (Battelle Pacific Northwest Labs., Richland, WA (USA)). May 1977. Contract EY-76-C-06-1830. 128p. Dep. NTIS, PC A07/MF A01.

The Multicomponent Mass Transfer (MMT) Model is a generic computer code, currently in its third generation, that was developed to predict the movement of radiocontaminants in the saturated and unsaturated sediments of the Hanford Site. This model was designed to use the water movement patterns produced by the unsaturated and saturated flow models coupled with dispersion and soil-waste reaction submodels to predict contaminant transport. This report documents the theorical foundation and the numerical solution procedure of the current (third) generation of the MMT Model. The present model simulates mass transport processes using an analog referred to as the Discrete-Parcel-Random-Walk (DPRW) algorithm. The basic concepts of this solution technique are described and the advantages and disadvantages of the DPRW scheme are discussed in relation to more conventional numerical techniques such as the finite-difference and finite-element methods. Verification of the numerical algorithm is demonstrated by comparing model results with known closed-form solutions. A brief error and sensitivity analysis of the algorithm with respect to numerical parameters is also presented. A simulation of the tritium plume beneath the Hanford Site is included to illustrate the use of the model in a typical application. 32 figs.

719 (PNL--2454) REVEGETATION OF A WASTE DISPOSAL SITE IN THE 200 AREAS OF THE HANFORD RESERVATION. Cline, J.F.; Uresk, V.A. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Nov 1977. Contract EY-76-C-06-1830. 22p. Dep. NTIS, PC A02/MF A01.

The purpose of the experiment was to test and compare the efficacy of various soil amendments in establishing a self-sustaining vegetative cover on soil fill overlying a portion of a former radioactive waste pond. Data are presented for content of ^{90}Sr, ^{106}Ru, ^{137}Cs, U, and Pu in the pond, percent frequency of occurrence of cheatgrass, rye, Russian thistle, bursage, and tansy mustard for three growing seasons; biomass summary by species from the treatments of straw, clay, and control; chemical analysis of young leaf tissue of cheatgrass; length, width, height, and volume measurements of mature Russian thistle plants; ^{90}Sr content of Russian thistle plants; and vegetative composition of black-tailed hare fecal pellets. (HLN)

720 (PNL--2850(Pt.2), pp 5.17-5.23) RADIOECOLOGY OF NUCLEAR FUEL CYCLES. Schreckhise, R.G.; Emery, R.M.; Rogers, L.E.; Cadwell, L.L.; Cline, J.F.; Fitzner, R.E.; Gano, K.A.; Poston, T.M.; Rickard, W.H. Feb 1979.

Pacific Northwest Laboratory annual report for 1978 to the DOE Assistant Secretary for Environment. Part 2. Ecological Sciences.

Sites where radioactive wastes are found are solid waste burial ground, soils below liquid storage areas, surface ditches and ponds, and the terrestrial environment around chemical processing facilities that discharge airborne radioactive debris from stacks. This study provides information to help assess the environmental impacts and certain potential human hazards associated with nuclear fuel cycles. A data base is being developed to define and quantify biological transport routes which will permit credible predictions and assessment of routine and potential large-scale releases of radionuclides and other toxic materials. Progress is reported in studies on terrestrial and aquatic radiology of waste management areas, biotic transport parameters (uptake and distribution of ^{232}U in peas and barley), and chemical speciation for uptake studies (^{233}U movement within a soil column). (JGB)

721 (PNL--3700(Pt.2)) PACIFIC NORTHWEST LABORATORY ANNUAL REPORT FOR 1980 TO THE DOE ASSISTANT SECRETARY FOR ENVIRONMENT. PART 2. ECOLOGICAL SCIENCES. Vaughan, B.E. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Feb 1981. 190p. NTIS, PC A09/MF A01.

Separate abstracts were prepared for 33 topics discussed in this progress report. Section 11 which deals with the energy-related research for other agencies such as EPA, NOAA, NSF, and NRC is not represented by a separate abstract. (KRM)

722 (PNL-SA--7497) POTENTIAL RADIOLOGICAL IMPACT OF A CONCEPTUAL HANFORD NUCLEAR ENERGY CENTER. Soldat, J.K. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Oct 1978. Contract EY-76-C-06-1830. 32p. (CONF-790728--5). Dep. NTIS, PC A03/MF A01.

From 24. annual meeting of the Health Physics Society; Philadelphia, PA, USA (7 Jul 1979).

The potential radiological impact of the siting of 20 light-water reactors and associated nuclear fuel cycle facilities on the Hanford reservation was evaluated by calculating the potential radiation doses received by individuals and populations in the vicinity of the reservation. The largest contributor to the potential radiation doses, to both the individual and the 50-mile population, were the effluents from the conceptual 1500 MT/yr fuel reprocessing plant. The effluents from the 20 reactors combined was the second largest contributor. The radiation dose contributions from the 300 MT/yr mixed oxide fuel fabrication plant were insignificant. The highest organ dose from all facilities combined was 24 mrem/yr to the child thyroid; followed by 8 mrem/yr to the adult thyroid. The 50-year collective dose commitment to the population within 50 miles was about 50 man-rem for most organs of reference, while the estimate for bone was 70 man-rem. With the exception of ^{85}Kr, the release rates of radionuclides were within the EPA guidelines. Removal of about 90% of the 4 x 10^5 Ci/yr per gigawatt-year of electricity of ^{85}Kr from the fuel reprocessing plant gaseous effluents would be required for compliance with the EPA guidelines.

Soil

723 (BNWL-SA--671) TRANSPORT ANALYSIS: BASIC PREDICTIVE APPROACH OF THE MOVEMENT OF POLLUTANTS THROUGH SOIL. Cearlock, D.B. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 3 May 1966. 18p. (CONF-660523--2). Battelle Pacific Northwest Labs., Richland, WA.

From 21. annual Purdue industrial waste conference; Lafayette, IN, USA (3 May 1966).

Transport analysis for determining distribution of pollutants through soil is discussed. Since the transport equation describes two independent phenomena, fluid movement and pollutant reactions (the reactions of the pollutant with its environment), the analysis was simplified by investigating them separately. The transport equation was then used to combine these into one interrelated equation which yields the concentration distribution of the pollutants involved.

724 (PNL--2479) ANALYSIS OF SMALL MAMMAL POPULATIONS INHABITING THE ENVIRONS OF A LOW-LEVEL RADIOACTIVE WASTE POND. Gano, K.A. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Mar 1979. Contract EY-76-C-06-1830. 64p. Dep. NTIS, PC A04/MF A01.

This study was designed to determine the kinds of small mammals living adjacent to 216-U-10 Pond, the radiation exposures these mice received, and the level and type of radionuclides assimilated while living next to this pond and the 216-Z-19 Ditch. Four species of mice were trapped including the Great Basin pocket mouse, deer mouse, house mouse, and the western harvest mouse. Animals were collected throughout the study and composite tissue samples were analyzed by gamma spectrometry. Also, an analysis for ^{238}Pu, ^{239,240}Pu, and ^{241}Am was performed. The most abundant gamma emitter was ^{137}Cs with the highest levels occurring at three trapping locations: one near the 216-Z-19 Ditch and two locations adjacent to the pond. House mice captured near the 216-Z-19 Ditch showed the highest levels with one gastrointestinal (GI) tract sample having 1600 pCi ^{137}Cs/g dry weight. Four tissue types from resident mice were analyzed for Pu and Am concentrations. The tissues analyzed were fur-skin, liver, lung, and muscle-bone. The highest concentration detected was 2.03 pCi ^{239,240}Pu/g dry weight in a fur-skin sample from house mice captured on the meadow transect near the pond. Results from radiochemical analyses of mouse tissues showed that pocket mice have the lowest concentrations of radionuclides. Another part of this study involved dosimeters implanted in resident mice to determine gamma exposure. Analyses revealed that mice living in the meadow transect adjacent to the pond receive the highest exposure. Again, house mice had the highest, with an average 54.9 R/yr. Dosimeters were placed in the soil along the trapping transects to measure gamma and thermal neutron exposure rates. The top decimeter of soil had the highest exposure rate with a mean of 75 R/yr in the meadow. Neutron dose in the soil was also highest near the surface with 37 mrad/yr average in the meadow. (ERB)

725 (PNL--2724) VERTICAL CONTAMINATION IN THE UNCONFINED GROUNDWATER AT THE HANFORD SITE, WASHINGTON. Eddy, P.A.; Myers, D.A.; Raymond, J.R. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Aug 1978. Contract EY-76-C-06-1830. 44p. Dep. NTIS, PC A03/MF A01.

Disposal to the ground at Hanford of large volumes of low- and intermediate-level wastes in the local unconfined groundwater flow system has raised concern about the movement and distribution of this waste. Previous work produced information on the horizontal movement of the waste, but little or no information exists on its vertical distribution within the unconfined groundwater flow system. In 1975 Phase I of a study was started to determine the vertical distribution of contaminants in three existing wells (699-23-40, 699-31-31, and 699-37-43). Because of negative results, only one well that produced positive results (699-31-31) was chosen for Phase II. Phase II consisted of tests conducted on this well by a testing company, with samples cross-checked by two different laboratories. Phase III was a cooperative study with Rockwell Hanford Operations, which included the installation, testing, and sampling of piezometers. The data were then analyzed using predictive codes and models in order to determine if vertical movement did occur. The present groundwater flow system shows some vertical contamination. However, concentrations are relatively higher near the surface of the flow system, indicating possible radial flow patterns from the groundwater mounds known to have developed under the chemical processing area disposal sites. Upward flow from deeper aquifers may be diluting the contaminant and masking a possible downward migration of contaminants.

726 (PNL--2850(Pt.2), pp 10.3-10.8) ROCKWELL SUPPORT STUDIES. Rogers, L.E.; Cadwell, L.L. Feb 1979.

Pacific Northwest Laboratory annual report for 1978 to the DOE Assistant Secretary for Environment. Part 2. Ecological Sciences.

Studies were designed either to (1) clarify the role of specific biota in the uptake and transport of radionuclides from shallow, low-level waste burial sites or (2) evaluate the radioactive exposure levels to biota inhabiting such sites. Tasks include studies on burrowing animals and shrub-inhabiting insects, dose distribution evaluation, environmental assessment of raptors, study of radionuclide levels in blacktailed hares, field plant uptake studies, plant uptake (laboratory) studies and burial ground stabilization studies.

727 (PNL--3498) POTENTIAL AIRBORNE RELEASE FROM SOIL-WORKING OPERATIONS IN A CONTAMINATED AREA. Sutter, S.L. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Aug 1980. Contract AC06-76RL01830. 30p. NTIS,

PC A03/MF A01.

Experiments were performed to provide an indication of how much material could be made airborne during soil-working operations in a contaminated area. Approximately 50 kg of contaminated soil were collected, dried, and mixed, and particle size distribution and ^{137}Cs content were characterized. In four experiments performed in a 2 ft x 2 ft wind tunnel at the Radioactive Aerosol Release Test Facility, soil was pumped into an airstream moving at 3.2, 10.4, 15.2, and 20 mph. These experiments were designed to maximize airborne releases by fluidizing the soil as it was pumped into the wind tunnel. Thus the airborne releases should represent upper limit values for soil-working operations. Airborne concentration and particle size samples were collected and all of the material deposited downstream was collected to calculate a mass balence. The fraction airborne was calculated using these measurements.

728 (PNL--3548) SUMMARY OF FOUR RELEASE CONSEQUENCE ANALYSES FOR HYPOTHETICAL NUCLEAR WASTE REPOSITORIES IN SALT AND GRANITE. Cole, C.R.; Bond, F.W. (Battelle-Northwest, Richland, WA (USA). Pacific Northwest Lab.). Dec 1980. Contract AC06-76RL01830. 131p. NTIS, PC A07/MF A01.

Release consequence methology developed under the Assessment of Effectiveness of Geologic Isolation Systems (AEGIS) program has now been applied to four hypothetical repository sites. This paper summarizes the results of these four studies in order to demonstrate that the far-field methodology developed under the AEGIS program offers a practical approach to the post-closure safety assessment of nuclear waste repositories sited in deep continental geologic formations. The four studies are briefly described and compared according to the following general categories: physical description of the repository (size, inventory, emplacement depth); geologic and hydrologic description of the site and the conceptual hydrologic model for the site; description of release scenario; hydrologic model implementation and results; engineered barriers and leach rate modeling; transport model implementation and results; and dose model implementation and results. These studies indicate the following: numerical modeling is a practical approach to post-closure safety assessment analysis for nuclear waste repositories; near-field modeling capability needs improvement to permit assessment of the consequences of human intrusion and pumping well scenarios; engineered barrier systems can be useful in mitigating consequences for postulated release scenarios that short-circuit the geohydrologic system; geohydrologic systems separating a repository from the natural biosphere discharge sites act to mitigate the consequences of postulated breaches in containment; and engineered barriers of types other than the containment or absorptive type may be useful.

729 (PNL--3616) UNSATURATED MOISTURE AND RADIONUCLIDE TRANSPORT: LABORATORY ANALYSIS AND MODELING. Gee, G.W.; Campbell, A.C.; Wierenga, P.J.; Jones, T.L. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jun 1981. Contract AT06-76RL01830. 44p. NTIS, PC A03/MF A01. Order Number DE81029397.

This report describes several laboratory procedures and computer model simulations used to evaluate the transport of water and radionuclides through unsaturated Hanford soils. The unsaturated hydraulic conductivity was measured using the steady state methods of Klute and the transient state method of Rose. These experimental data were compared to other conductivity models. Good agreement was found between all methods in the wet range; however, disagreement was found in the dry range. None of the conductivity models explicity addresses the water vapor component of the conductivity. This may explain the under prediction of the hydraulic conductivity in the dry range where vapor transport is important. Radionuclide transport through unsaturated media was investigated by using two solute transport models to describe the transport of tritium and strontium-85 in laboratory columns. A two parameter convective-dispersive model was compared with a four parameter mobile-immobile water model. Both models adequately described the movement of tritium and strontium through small (5 cm x 27.5 cm) columns and the movement of tritium and strontium through a large (0.5 m x 1.7) column. The dispersion coefficient was found to be sensitive to changes in both velocity and column length. The mobile-immoble water equations were not as sensitive to changes in experimental scales as the convective-dispersive equation. Both models were relatively successful in describing the rapid flush of strontium-85 from a column initially leached with a low salt solution followed by a high salt solution, a phenomona called the snow plow effect. The four parameter mobile-immobile water model predicted the initial release of the strontium more accurately than the two parameter convective-dispersive model. Both models confirm enhanced mobility of strontium-85 with leaching solutions of increased salt concentration.

730 (PNL--3647) PLUTONIUM IN SURFACE SOIL NEAR THE SOUTHWESTERN BOUNDARY OF THE HANFORD PROJECT. Price, K.R.; Dirkes, R.L. (Battelle Pacific Northwest Labs., Richland, WA (USA)). May 1981. Contract AC06-76RL01830. 21p. NTIS, PC A02/MF A01.

Samples of airborne particles collected near the Prosser Barricade in another study showed low $^{240}Pu/^{239}Pu$ ratios that are indicative of Hanford-produced plutonium. In an effort to locate evidence of a trail or the remains of a large short-term release of plutonium that may have occurred during past Hanford operations, surface soil samples were collected along the southweastern boundary of the Hanford Site in December 1979. Results indicated the possibility of slightly elevated levels of $^{239,240}Pu$ (0.016 pCi/g) occurring in the general vicinity of the Arid Land Ecology Field lab extending to the junction of Highway 240 and Horn Rapids Road as compared to lower levels (0.006 pCi/g) in a northwesterly direction along the base of Rattlesnake Mountain and the eastern slope of Yakima Ridge. Assuming the worldwide average $^{240}Pu/^{239}Pu$ ratio of 0.18 for soil of the Pacific Northwest, the plutonium in these soil samples may be slightly less enriched with ^{240}Pu ($^{240}Pu/^{239}Pu$ = 0.16). No evidence was discovered of an acute release remaining intact and crossing the southwestern boundary during the operating history of plutonium facilities in the 200 Areas.

731 (PNL-SA--9018) LONG TERM EROSION RESISTANCE OF UNPROTECTED RADON SUPPRESSION EARTH COVERS TO WIND STRESSES. Bander, T.J. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Oct 1980. Contract AC06-76RL01830. 10p. (CONF-801155--2). NTIS, PC A02/MF A01.

From 3. annual symposium on uranium mill tailings manegement; Ft Collins, CO, USA (24 Nov 1980).

Estimates of the amount of soil cover lost due to wind erosion are obtained using three different models. One model developed by W. S.

Chepil employs a wind erosion equation which is based on wind tunnel and agricultural field measurements. The second model is based on an analytical solution of the saltation process obtained by P. R. Owen and agrees well with the data of Bagnold and Zingg. The third model was developed by Gillette and is based on a modification of the horizontal flux equation used by Bagnold and Chepil. Basic differences between the models are examined and a comparison of the soil movement predicted by each is presented. The usefulness of these wind erosion models as guidance in designing long term erosion-resistant covers for tailings piles is considered.

732 (PNL-SA--9122) CHEMICAL SPECIES OF MIGRATING RADIONUCLIDES AT A SHALLOW LAND LOW-LEVEL RADIOACTIVE-WASTE BURIAL SITE. Kirby, L.J.; Toste, A.P.; Wilkerson, C.L. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 1981. Contract AC06-76RL01830. 9p. (CONF-810722--9). NTIS, PC A02/MF A01. Order Number DE81028011.

From International symposium on migration in the terrestrial environment of long-lived radionuclides from the nuclear fuel cycle; Knoxville, TN, USA (27 Jul 1981).

A research program at the Maxey Flats, Kentucky (U.S.A.) waste disposal site has been undertaken to define the chemical species contributing to the migration or retention of radionuclides contained in waste buried at that site. An experimental trench and inert atmosphere sampling wells were installed to sample water for determination of the chemical species of migrating radionuclides. The organic ligands are studied by gas chromatography, steric exclusion chromatography and mass spectrometry; and the data correlated with specific radionuclide counting data to determine precise chemical species. Preliminary data are reported in the text.

733 (PNL-SA--9887) RESEARCH SUMMARY: CHARACTERIZATION OF RADIONUCLIDE AND MOISTURE MOVEMENT THROUGH ARID REGION SEDIMENTS. Gee, G.W.; Jones, T.L.; Rai, D. (Pacific Northwest Lab., Richland, WA (USA)). Sep 1981. Contract AC06-76RL01830. 12p. NTIS, PC A02/MF A01. Order Number DE82003840.

This project has the task of understanding the movement of moisture and radionuclides under arid region conditions. This understanding will be used to maximize the isolation of low level waste from the environment. Specific objectives include: field monitoring of moisture and radionuclide transport at an arid region site; assessment of the interaction of radionuclides with unsaturated soils in arid regions; evaluation of radionuclide transport in unsaturated soils by appropriate mathematical models; and assessment of the importance of upward migration of radionuclides by evaporation and diffusion processes. The Burial Waste Test Facility (BWTF) located near Richland, Washington, on the Department of Energy (DOE) Hanford Site has been monitored for water content and radionuclide transport for the past two years. Tritium movement has been observed to depths of 7.6 m in both irrigated and nonirrigated lysimeters. Laboratory tests were conducted to determine how leachate from uranium tailings interacts with geologic materials. Acid leach tailings and tailings solution and geologic materials typical of mill site tailing pits were physically and chemically characterized. Investigation was made of the sorption characteristics of heavy metals and radionuclides on the geologic materials under low and neutral pH conditions. From solubility tests conducted at Pacific Northwest Laboratory, thermodynamic considerations predicted that for the Eh-pH range of natural aqueous environment, the dominant species of Pu is likely to be Pu(V) in relatively oxidizing environments and Pu(III) in reducing environments. Radionuclide transport through unsaturated media was investigated by using two solute transport models to describe the transport of tritium and strontium-85 in laboratory columns. A new approach was used to analyze radon emissions from uranium mill tailings.

734 ^{241}AM, ^{237}NP, AND ^{99}TC SORPTION ON TWO UNITED STATES SUBSOILS FROM DIFFERING WEATHERING INTENSITY AREAS. Routson, R.C.; Jansen, G.; Robinson, A.V. (Battelle Pacific Northwest Labs., Richland, Wash. (USA)). Health Physics; 33: No. 4, 311-317(Oct 1977).

For the purpose of assessing the consequences of a potential release event from a high-level waste repository, the sorption of ^{241}Am, ^{237}Np, and ^{99}Tc was measured on two soils as a function of solution concentration of Na and Ca from extremes of weathering intensity areas. Distribution coefficients (Kd values) were determined on subsoil samples from Washington and South Carolina for ^{241}Am, ^{237}Np, and ^{99}Tc, as a function of equilibrium solution concentration of calcium (Ca^{+2}) and of sodium (Na^{+}). Kd values decreased in all cases with increasing solution concentrations of (Ca^{+2}) and (Na^{+}). For the South Carolina subsoil, Kd values ranged from 1.0 to 67 ml/g for ^{241}Am as a function of (Ca^{+2}), from 0.2 to 0.002 M, respectively; 1.6 to 280 ml/g for ^{241}Am as a function of (Na^{+}); 0.43 to 0.66 ml/g for ^{237}Np as a function of (Ca^{+2}); and 0.16 to 0.25 ml/g for ^{237}Np as a function of (Na^{+}), from 3.0 to 0.015 M, respectively. For the Washington soil Kd values were greater than 1200 ml/g for ^{241}Am and ranged from 0.36 to 2.37 ml/g as a function of (Ca^{+2}) and from 3.10 to 3.90 ml/g for ^{237}Np as a function of (Na^{+}), over the above concentration ranges. Kd values for ^{99}Tc were essentially zero at all $NaHCO_3$ concentrations on the South Carolina subsoil. This study demonstrated that sorption assessments based on plutonium data are inadequate for estimating other actinide sorption. ^{241}Am and ^{237}Np were found to be poorly sorbed in comparison to Pu.

735 CHARACTERIZATION OF ACTINIDE-BEARING SEDIMENTS UNDERLYING LIQUID WASTE DISPOSAL FACILITIES AT HANFORD. Price, S.M. (Atlantic Richfield Hanford Co., Richland, Wash. (USA)); Ames, L.L. (Battelle Pacific Northwest Labs., Richland, Wash. (USA)). pp 191-210 of Transuranium nuclides in the environment. Vienna: International Atomic Energy Agency (1976).

From IAEA international symposium on transuranium nuclides in the environment; San Francisco, CA, USA (17 Nov 1975).

See CONF-751105--.

Past liquid waste disposal practices at the US Energy Research and Development Administration's Hanford Reservation have included the discharges of solutions containing low concentrations of actinides directly into the ground via structures collectively termed ''trenches''. Characterization of samples from two of these trenches, the 216-Z-9 and the 216-Z-1A(a), has been initiated to determine the present form and migration potential of plutonium stored in sediments which received high salt, acidic waste liquids. Analysis of samples acquired by drilling has revealed that the greatest measured concentration of plutonium, approximately 10^6 $\mu Ci^{239}Pu/l$ of sediment, occurs in both facilities just below the points of release of the waste liquids. This concentration decreases to approximately 10^3 μCi ^{239}Pu/l of sediment within the first 2 m of the underlying sediment columns and to approximately 10 μCi ^{239}Pu/l of sediment at the maximum depth sampled (9m). Examination of relatively undisturbed sediment cores illustrated two types of Pu occurrence responsible for this distribution. One of these types is composed of Pu particles (>70 wt% PuO_2) added to the disposal site in the same form. This ''particulate'' type was ''filtered out'' within the upper 1 m of the sediment column, accounting for the high concentration of Pu/l of sediment in this region. The second type of Pu (<0.5 wt% PuO_2) was originally disposed of as soluble Pu(IV). This ''nonparticulate'' type penetrated deeper within the sediment profile and was deposited in association with silicate hydrolysis of the sediment fragments.

Terrestrial Ecosystems and Food Chains

736 (BNWL--90) EVALUATION OF RADIOLOGICAL CONDITIONS IN THE VICINITY OF HANFORD, 1964. Wilson, R.H. (ed.). (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jul 1965. Contract AT(45-1)-1830. 206p. NTIS.

The evaluation of results obtained from the Hanford environmental surveillance program for 1964 indicates that most of the environmental radiation exposure for the majority of persons living in the neighborhood of the Hanford project was due to natural sources and world-wide fallout rather than to Hanford operations. Of the low-level wastes that are released to the environment from the Hanford plants, neutron-induced radionuclides present in reactor cooling water discharged to the Columbia River continued to be the source of greatest potential exposure to people in the environs. The primary pathways of exposure from this source are drinking water derived from the river, consumption of fish and waterfowl which inhabit the river, and foodstuffs grown on land irrigated with water pumped from the Columbia downstream from Hanford. ^{131}Iodine in the Hanford environs remained at very low concentrations in 1964. The Chinese nuclear test on October 16 caused a brief increase in ^{131}I, but concentrations soon returned to the low levels experienced during most of 1964. The postulated maximum exposure from ^{131}I to the thyroid of a small child amounted to only about 5% of the Radiation Protection Guide recommended for individuals by the Federal Radiation Council.

737 (PNL--2383) RADIONUCLIDE CONCENTRATIONS IN SELECTED FOODSTUFFS AND WILDLIFE FROM THE HANFORD ENVIRONS, 1971--1975. Fix, J.J.; Leete, S.C.; Bramson, P.E. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Nov 1977. Contract EY-76-C-06-1830. 35p. Dep. NTIS, PC A03/MF A01.

To assess the public's potential exposure to radionuclides from Hanford operations, a variety of foodstuffs including milk, meat, chickens, eggs, and leafy vegetables have routinely been collected from the Hanford environs. Radionuclides observed included naturally-occurring ^{40}K, as well as the radionuclides ^{90}Sr and ^{137}Cs contributed by worldwide fallout. Radionuclides of Hanford origin would be similar to those from fallout and must necessarily be present in greater levels than the fallout nuclides to be detectable. Data on radioactivity for each foodstuff category is examined seperately.

738 (PNL--3300(Pt.2), pp 149) HANFORD HIGH-LEVEL WASTE. Soldat, J.K.; Strenge, D.L.; Napier, B.A.; Zimmerman, M.C.; Kennedy, W.E. Jr. Feb 1980. Dep. NTIS, PC A11/MF A01.

Pacific Northwest Laboratory annual report for 1979 to the DOE Assistant Secretary for Environment. Part 2. Ecological sciences.

Methodology was developed for estimating the amount of land which might require control or decontamination because of contamination above specified limits following accidental releases of radionuclides at Hanford. Provisional limits were derived by calculating potential doses via food pathways and external exposure. The results of the calculations were published in a preliminary draft report. The report also included the results of calculations of potential doses from onsite and offsite transportation accidents.

739 (PNL--3300(Pt.2), pp 153-159) ROCKWELL SUPPORT STUDIES. Cadwell, L.L. Feb 1980. Dep. NTIS, PC A11/MF A01.

Pacific Northwest Laboratory annual report for 1979 to the DOE Assistant Secretary for Environment. Part 2. Ecological sciences.

Studies performed for the Rockwell Hanford Operations (RHO) were designed to either (1) identify the role of biota in the uptake and transport of radionuclides from low-level waste managemnt areas or (2) design and/or evaluate methods for reducing biological transport of radionuclides away from waste management areas. The completion and publication of documents reporting the results of previous studies was also emphasized this fiscal year. Field studies were designed to characterize insect and plant communities occupying low-level waste management areas and to evaluate pond ecosystem response to herbicide application. Laboratory studies involved the examination of (1) germination response in plant species proposed for surface stabilization on burial grounds, (2) a synthetic polymer matrix as a carrier and delivery system for phytotoxins to act as a barrier to plant root penetration of shallow-land low-level waste burial sites, and (3) factors responsible for differences in radionuclide availability to plants resulting from the chemical equilibration of radionuclides occurring under field conditions.

740 (PNL--3315) ENVIRONMENTAL CONSEQUENCES OF POSTULATED PLUTONIUM RELEASES FROM EXXON NUCLEAR MOFP, RICHLAND, WASHINGTON, AS A RESULT OF SEVERE NATURAL PHENOMENA. Jamison, J.D.; Watson, E.C. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Feb 1980. Contract AC06-76RL01830. 69p. NTIS, PC A04/MF A01.

Potential environmental consequences in terms of radiation dose to people are presented for postulated plutonium releases caused by severe natural phenomena at the Exxon Nuclear Company Mixed Oxide Fabrication Plant (MOFP), Richland, Washington. The severe natural phenomena considered are earthquakes, tornadoes, high straight-line winds, and floods. Maximum plutonium deposition values are given for significant locations around the site. All important potential exposure pathways are examined. The most likely 50-year committed dose equivalents are given for the maximum-exposed individual and the population within a 50-mile radius of the plant. The maximum plutonium deposition values most likely to occur offsite are also given.

741 (PNL--3475) ENVIRONMENTAL CONTROL TECHNOLOGY FOR MINING AND MILLING LOW-GRADE

URANIUM RESOURCES. Weakley, S.A.; Blannik, D.E.; Long, L.W.; Bloomster, C.H. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Apr 1981. Contract AC06-76RL01830. 151p. NTIS, PC A08/MF A01.

This study examined the type and level of wastes that would be generated in the mining and milling of U_3O_8 from four potential domestic sources of uranium. The estimated costs of the technology to control these wastes to different degrees of stringency are presented.

742 (PNL--3509(Pt.1)) STANDARDIZED INPUT FOR HANFORD ENVIRONMENTAL IMPACT STATEMENTS. Napier, B.A. (Battelle Pacific Northwest Labs., Richland, WA (USA)). May 1981. Contract AC06-76RL01830. 45p. NTIS, PC A03/MF A01.

Models and computer programs for simulating the environmental behavior of radionuclides in the environment and the resulting radiation dose to humans have been developed over the years by the Environmental Analysis Section staff, Ecological Sciences Department at the Pacific Northwest Laboratory (PNL). Methodologies have evolved for calculating raidation doses from many exposure pathways for any type of release mechanism. Depending on the situation or process being simulated, different sets of computer programs, assumptions, and modeling techniques must be used. This report is a compilation of recommended computer programs and necessary input information for use in calculating doses to members of the general public for environmental impact statements prepared for DOE activities to be conducted on or near the Hanford Reservation.

743 (PNL--3729) ENVIRONMENTAL STATUS OF THE HANFORD SITE FOR CY-1980. Sula, M.J.; Blumer, P.J.; Dirkes, R.L. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Aug 1981. Contract AC06-76RL01830. 70p. NTIS, PC A04/MF A01. Order Number DE81028358.

Samples of air, surface water, soil, vegetation, and wildlife were collected and external penetrating radiation dose measurements were made in the vicinity of the major operating areas on the Hanford Site. The samples were analyzed for radioactive constituents including tritium, strontium-90, plutonium, and gamma-emitting radionuclides. In addition, site roads, railroad tracks, and burial grounds were surveyed periodically to detect any abnormal levels of radioactivity. Radioactive and nonradioactive waste discharges and environmentally related unusual occurrences reported for the major operating areas were summarized.

744 (PNL-SA--9123) RADIONUCLIDES IN A DECIDUOUS FOREST SURROUNDING A SHALLOW-LAND-BURIAL SITE IN THE EASTERN UNITED STATES. Rickard, W.H.; Kirby, L.J.; McShane, M.C. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jun 1981. Contract AC06-76RL01830. 15p. (CONF-810722--11). NTIS, PC A02/MF A01. Order Number DE81028009.

From International symposium on migration in the terrestrial environment of long-lived radionuclides from the nuclear fuel cycle; Knoxville, TN, USA (27 Jul 1981).

The objective of this study was to determine if radioactive materials buried in trenches at the Maxey Flats burial ground in eastern Kentucky have migrated into the surrounding oak-hickory forest. Forest floor litter, minearl soil, and tree leaves were sampled and the radionuclide content measured. (ACR)

Radioactive Materials Monitoring and Transport

745 (BNWL--2305) ASSOCIATION OF HANFORD ORIGIN RADIONUCLIDES WITH COLUMBIA RIVER SEDIMENT. Robertson, D.E.; Fix, J.J. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Aug 1977. Contract EY-76-C-06-1830. 102p. Dep. NTIS, PC A06/MF A01.

Results are reported from measurements of present radionuclide concentrations in Columbia River sediments. A description of the rates and mechanisms governing the decrease in radioactivity levels in the river sediments between 1971 and 1976; a characterization of the areal and depth distribution of fine grain, silty sediments in the McNary Reservoir mode by sub-bottom seismic surveying; and analysis of deep piston cores which penetrate the entire thickness of sediment deposits at selected sites in McNary Reservoir to provide radionuclide data for estimating the inventories of radionuclides in the sediment deposits in the McNary Reservoir are included. The results of these studies showed that the short- and intermediate-lived radionuclides have now decayed to insignificant levels, and only a few long-lived radionuclides (i.e., ^{54}Mn, ^{55}Fe, ^{60}Co, ^{137}Cs, $^{152-154}Eu$, ^{238}Pu, $^{239-240}Pu$, and ^{241}Am) at trace concentrations remain buried in the sediment deposits. The surface sediments in the McNary Reservoir contain much lower radionuclide concentrations then deeper sediments because of the build-up of 40 to 80 cm of fresh, relatively uncontaminated new sediment deposits since 1971.

746 (BNWL-SA--406) SEQUENCE FOR PREDICTING WASTE TRANSPORT BY GROUND WATER. Nelson, R.W. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 15 Apr 1965. 17p. (CONF-651207--1). NTIS.

From American Water Resources Association, 1st annual meeting; Chicago, IL, USA (2 Dec 1965).

An analysis sequence is presented to enable rational predictions of containment concentrations entering streams, rivers, or other potable waters traversing the region of study by ground water transport. The purposes are to give an over-all description of the sequence, briefly describe the methods developed, and indicate the computer programs available to assist in accomplishing the analysis. Throughout the analysis, major emphasis was on realistically evaluating actual situations. The natural sediments are assumed heterogeneous with respect to the permeability, potential measurements can be irregularly located, and boundary conditions may be both mixed and nonuniform.

747 (BNWL-SA--6244) MIGRATION OF PLUTONIUM FROM FRESHWATER ECOSYSTEM AT HANFORD. Emery, R.M.; Klopfer, D.C.; McShane, M.C. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Sep 1977. Contract EY-76-C-06-1830. 44p. Dep. NTIS, PC A03/MF A01.

A reprocessing waste pond at Hanford has been inventoried to determine quantities of plutonium (Pu) that have been accumulated since its formation in 1944. Expressions of export were developed from these inventory data and from informed assumptions about the vectors which act to mobilize material containing Pu. This 14-acre pond provides a realistic illustration of the mobility of Pu in a lentic ecosystem. The ecological behavior of Pu in this pond is similar to that in other contaminated aquatic systems having widely differing limnological characteristics. Since its creation, this pond has received about one Ci of $^{239,240}Pu$ and ^{238}Pu, most of which has been retained by its sediments. Submerged plants, mainly diatoms and Potamogeton, accumulate >95% of the Pu contained in biota. Emergent insects are the only direct biological route of export, mobilizing about 5 x 10^3 nCi of Pu annually, which is also the estimated maximum quantity of the Pu exported by waterfowl, birds and mammals collectively. There is no apparent significant export by wind, and it is not likely that Pu has migrated to the ground water below U-Pond via percolation. Although this pond has a rapid flushing rate, a eutrophic nutrient supply with a diverse biotic profile, and interacts with an active terrestrial environment, it appears to effectively bind Pu and prevent it from entering pathways to man and other life.

748 (PNL--2423) DESIGN AND ANALYSIS OF AQUATIC MONITORING PROGRAMS AT NUCLEAR POWER PLANTS. McKenzie, D.H.; Kannberg, L.D.; Gore, K.L.; Arnold, E.M.; Watson, D.G. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Nov 1977. Contract EY-76-C-06-1830. 139p. Dep. NTIS, PC A07/MF A01.

This report addresses some of the problems of designing, conducting, and analyzing aquatic environmental monitoring programs for impact assessment of nuclear power plants. The concepts discussed are applicable to monitoring the effects of chemical, radioactive, or thermal effluents. The concept of control and treatment station pairs is the fundamental basis for the experimental method proposed. This concept is based on the hypothesis that the relationship between the two stations forming the pair can be estimated from the preoperational period and that this relationship holds during the operational period. Any changes observed in this relationship during the operational period are assumed to be the result of the power plant impacts. Thus, it is important that station pairs are selected so it can be assumed that they respond to natural environmental changes in a manner that maintains that relationship. The major problem in establishing the station pairs will be the location of the control station. The universal heterogeneity in the environment will prevent the establishment of identical station pairs. The requirement that the control station remain unaffected by the operation of the power plant dictates a spacial separation with its associated differences in habitat. Thus, selection of the control station will be based upon balancing the following two criteria: (1) far enough away from the plant site to be beyond the plant influence, and (2) close enough to the treatment station that the biological communities will respond to natural environmental changes consistently in the same manner.

749 (BNWL-tr--268) GEOSPHERE MODEL: DESCRIPTION, OPINIONS, AND INPUT REQUIREMENTS. Grundfelt, B. (Kemakta Konsult AB (Sweden)). 30 Sep 1977. Contract EY-76-C-06-1830. Translation of KBS-TR--10 39p. Dep. NTIS, PC A03/MF A01.

The Geosphere computer model described is used to compute the amount of radioactivity released per year (the release rate) to a receiver from a repository in a geologic formation through which ground water is flowing. The release rate is computed as a function of time, so that a chromatographic spectrum is obtained. In its present state, the program evaluates 56 radionuclides that may be present in reactor effluents or naturally occurring in rocks. The output, in the form of chromatographic spectra, obtained from the Geosphere model is stored in a data file in such a way that the file can be used directly as input for the Biosphere model.

Water

750 (BNWL--63) SEQUENCE FOR PREDICTING WASTE TRANSPORT BY GROUND WATER. Nelson, R.W. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 15 Apr 1965. Contract AT(45-1)-1830. 12p. NTIS.

Tabular field data and fitting functions submitted to a digital computer give the ground water potential equations for heterogeneous soils. Added to special boundary conditions, this potential equation is the key for computing the permeability distribution. The second computer program has four sequential operations. The first two provide permeability distribution and the last two give water travel time and flux distribution which enables water waste transport calculations. The nonlinear boundary value problem interrelates all pertinent variables, and when solved, gives the new ground water potential for flow system changes. The nonlinear equations are then solved digitally by a piecewise method for computations of predicted distribution of ground water potentials. The present computer program has a limit on problem size (3000 equations), but an enlarged program is being developed. Travel paths and times of flow and subsequently, the water arrival distribution are determined from the water potential function. An arrival curve is then used to plot the concentration of wastes entering streams.

751 (NUREG/CR--0576) SEDIMENT AND RADIONUCLIDE TRANSPORT IN RIVERS FIELD SAMPLING PROGRAM CATTARAUGUS AND BUTTERMILK CREEKS, NEW YORK. ANNUAL PROGRESS REPORT, OCTOBER 1977--SEPTEMBER 1978. Ecker, R.M.; Onishi, Y. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Dec 1978. Contract EY-76-C-06-1830. 73p. (PNL--2551). Dep. NTIS, PC A04/MF A01.

One mechanism affecting the dispersal of radioactive materials in water bodies is radionuclide adsorption by sediment. Consequently, sediment transport is a major factor to consider when evaluating radionuclide migration. As a part of a study on sediment and radionuclide transport in rivers, Pacific Northwest Laboratory (PNL) is investigating the effect of sediment on the transport of radionuclides in Cattaraugus and Buttermilk Creeks, New York, during different flow conditions. Sources of radioactivity in these creeks were a low-level waste disposal site and a nuclear fuel reprocessing plant. Reprocessing operations were terminated in 1972 and waste disposal was discontinued in 1975. Other sources of radioactivity include fallout from worldwide weapons testing and natural background radioactivity. The major objective of the PNL Field Sampling Program is to provide data on sediment and radionuclide characteristics in Cattaraugus and Buttermilk Creeks to verify the use of the Sediment and Radionuclide Transport model, SERATRA, for nontidal rivers. The program is divided into three phases: Phase 1, medium-flow condition; Phase 2, low-flow condition; and Phase 3, high-flow condition. To date, results have been obtained primarily for the Phase 1 portion. For the Phase 1 sampling, 10 transects were established to collect data on flow and channel, water, sediment, and radionuclide characteristics. Some radiological analyses were made on samples of water, suspended sediment, and bed sediment.

752 (PNL--2624) ENVIRONMENTAL MONITORING REPORT ON THE STATUS OF GROUND WATER BENEATH THE HANFORD SITE, JANUARY--DECEMBER 1977. Myers, D.A. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Apr 1978. Contract EY-76-C-06-1830. 104p. Dep. NTIS, PC A06/MF A01.

An evaluation of the status of ground-water contamination resulting from Hanford's onsite discharges is presented. Data collected during 1977 describe the movement of the major plumes (β/sub t/, ^3H, NO^-_3) that respond to the influences of ground-water flow, ionic dispersion and radioactive decay. The total beta plume continues to recede. A continuation of the study to ascertain the contributors to this plume indicates that the suspected highly mobile species do not exist. The tritium plume continues to expand and is mapped as having reached the Columbia River, although its contribution to the river cannot be distinguished from that attributable to atmospheric fallout. Nitrate concentrations in the vicinity of the 100-H Area have risen rapidly due to a leaking evaporation facility. The results of a study to determine the vertical distribution of contaminants in the Hanford ground-water system indicate that the vast majority of contaminants are stratified in the upper portions of the unconfined aquifer.

753 (PNL--2635) WELL MAINTENANCE EVALUATION. McGhan, V.L. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Oct 1978. Contract EY-76-C-06-1830. 52p. Dep. NTIS, PC A04/MF A01.

The ground-water monitoring program is an integral part of the total environmental surveillance program for the Hanford Site. Extensive ground-water data have been collected and studied over the past several decades. All of this data is, of course, dependent upon the quality of the ground-water sampling structures. A program to upgrade the quality of ground-water sampling was initiated in early 1974. That program also included changes to avoid cross-contamination of ground-water samples by installing permanently mounted individual pumps in many of the sampling wells. These two programs have resulted in increased reliability of both the ground-water samples and the analytical data. This basic quality assurance effort has provided a high level of confidence in ground-water surveillance. The current program is providing data with reliabilities not previously attainable.

754 (PNL--2899) RADIOLOGICAL STATUS OF THE GROUND-WATER BENEATH THE HANFORD PROJECT, JANUARY--DECEMBER 1978. Eddy, P.A. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Apr 1979. Contract EY-76-C-06-1830. 81p. Dep. NTIS, PC A05/MF A01.

An evaluation of the status of ground-water contamination resulting from Hanford's onsite discharges is presented. Data collected during 1978 describe the movement of major plumes (β/sub t/, ^3H, NO_3) that respond to the influences of ground-water flow, ionic dispersion, and radioactive decay. The total beta plume continues to recede, with the exception of a beta source that is beginning to show up in the 300 Area, a result of minor spills and leaks. The tritium plume continues to expand and is mapped as having reached the Columbia River, although its contribution to the river cannot be distinguished from that attributable to atmospheric fallout. The plume now shows much the same configuration as in 1977. The nitrate plume shows general stability relative to its size with concentrations in the vicinity of the 100-H Area continuing to be high as a result of leaks from the evaporation facility. The results of a study to determine the vertical distribution of contaminants in the Hanford ground-water system indicate that the majority of contaminants are stratified in the upper portions of the unconfined aquifer.

755 (PNL--3160-1) VARIABLE THICKNESS TRANSIENT GROUND-WATER FLOW MODEL. VOLUME 1. FORMULATION. Reisenauer, A.E. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Dec 1979. Contract EY-76-C-06-1830. 35p. Dep. NTIS, PC A03/MF A01.

Mathematical formulation for the variable thickness transient (VTT) model of an aquifer system is presented. The basic assumptions are described. Specific data requirements for the physical parameters are discussed. The boundary definitions and solution techniques of the numerical formulation of the system of equations are presented.

756 (PNL--3346) RADIOLOGICAL STATUS OF THE GROUND WATER BENEATH THE HANFORD PROJECT, JANUARY-DECEMBER 1979. Eddy, P.A.; Wilbur, J.S. (Battelle Columbus Labs., OH (USA)). Apr 1980. Contract AC06-76RL01830. 114p. NTIS, PC A06/MF A01.

Operations on the Hanford Site since 1944 have resulted in discharge of large volumes of process cooling water and low-level liquid radioactive waste to the ground. Radioactivity and chemical substances have been carried with these discharges and have reached the Hanford ground water. For may years wells have been used as groundwater sampling structures to gather data on the distribution and movement of these discharges as they interact with the unconfined ground water beneath the site. During 1979, 317 wells were sampled on various frequencies from weekly to annually. This report is one of a series prepared annually to document the evaluation of the status of ground water on the Hanford Site. Data collected during 1979 describe the movement of radionuclide (Tritium and Beta) and nitrate plumes that respond to the influence of groundwater flow, ionic dispersion and radioactive decay.

757 (PNL--3768) RADIOLOGICAL STATUS OF THE GROUND WATER BENEATH THE HANFORD SITE, JANUARY-DECEMBER 1980. Eddy, P.A.; Wilbur, J.S. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Apr 1981. Contract AC06-76RL01830. 117p. NTIS, PC A06/MF A01.

Operations at the Hanford Site since 1944 have resulted in the discharge to the ground of large volumes of process cooling water and low-level liquid radioactive waste. Radioactivity and chemical substances have been carried with these discharges and have reached the Hanford ground water. For many years wells have been used as ground-water sampling structures to gather data on the distribution and movement of these discharges as they interact with the unconfined ground water beneath the Hanford Site. During 1980, 317 such structures were sampled at various times for radionuclide and chemical contaminants. Data collected during 1980 describe the movement of tritium and ruthenium-106 and the nonradioactive nitrate plume as well as their response to the influences of ground-water flow, ionic dispersion, and radioactive decay.

758 DUMPING OF LOW-LEVEL RADIOACTIVE WASTE IN

THE DEEP OCEAN. Templeton, W.L. (Battelle Pacific Northwest Labs., Richland, WA (USA)). pp 451-464 of Impacts of radionuclide releases into the marine environment. Proceedings of an international symposium jointly organized by the IAEA and the OECD NEA and held in Vienna 6-10 October 1980. Vienna, Austria; IAEA (1981).

From International symposium on the impacts of radionuclide releases into the marine environment; Vienna, Austria (6 Oct 1980).

Two international agreements relate to the dumping of packaged radioactive waste into the oceans - the Convention on the Prevention of Marine Pollution by Dumping Wastes and Other Matter of 1972 (London Convention) and the Multilateral Consultation and Surveillance Mechanism for Sea Dumping of Radioactive Waste of 1977 under the Organization for Economic Co-operation and Development (OECD). The International Atomic Energy Agency was given the responsibility to define high-level radioactive wastes which are unsuitable for dumping in the oceans and to make recommendations for the dumping of other radioactive wastes. A revised Definition and Recommendations was submitted and accepted by the London Convention. The technical basis for the Definition is reviewed here and a description is given of how it has been applied to the radiological assessment of the only operational dumping site in the North East Atlantic.

759 SOLUTION SPECIES OF PLUTONIUM IN THE ENVIRONMENT. Rai, D.; Serne, R.J.; Swanson, J.L. (Batelle Pacific Northwest Lab., Richland, WA). Journal of Environmental Quality; 9: No. 3, 417-420(Jul 1980).

Information regarding the oxidation states of Pu in environmental samples is needed for estimating its migration through the geologic media. Predictions based upon thermodynamic data indicate that in the Eh-pH range of natural aqueous environments, the dominant species of Pu is likely to be Pu[V] in relatively oxidizing environments and Pu[III] in reducing environments. Because of the lack of methods of determining Pu[V] in environmental samples containing trace concentrations of Pu, Pu[V] has not been previously identified in these samples. Plutonium [VI] has generally been assumed to be the dominant species in relatively oxidizing environments. However, the results presented show that solutions in equilibrium with ^{239}Pu[IV] hydroxide contain Pu[V]. The presence of Pu[V] in solutions contacting $^{239}PuO_2(s)$ is also inferred from the data.

760 MODEL ASSESSMENT OF CONTAMINANT SEEPAGE FROM BURIED URANIUM MILL TAILINGS. Nelson, R.W.; Reisenauer, A.E.; Gee, G.W. (Battelle, Pac Northwest Lab, Richland, Wash). pp 245-269 of International conference on uranium mine waste disposal, 1st, 1980. Brawner, C.O. (ed.). New York, NY; Soc. of Min. Eng. of the AIME (1980).

From 1. international conference on uranium mine waste disposal; New York, NY, USA (19 May 1980).

Assessment was made to evaluate contaminant transport to ground water from clay-lined and unlined tailings pits. Hydrologic soil characteristics were measured on materials at the Morton Ranch uranium mill in Wyoming. Contaminant flow paths, the advancing contaminant flow fronts for sorbed and non-sorbed constituents, and travel times of contaminants along flow paths enable evaluation of the environmental consequences for alternative control methods. 8 refs.

761 RADIOACTIVITY IN THE OCEANS. Templeton, W.L. (Battelle Pacific Northwest Labs., Richland, WA (USA)). pp 949-957 of Proceedings of the 6th International Congress of Radiation Research. Okada, S. (Tokyo Univ. (Japan). Faculty of Medicine); Imamura, M.; Terashima, T.; Yamaguchi, H. (eds.). Tokyo, Japan; Japanese Association for Radiation Research (1979).

From 6. international congress symposium on radiation research and stem cells; Tokyo, Japan (13 May 1979).

While the revised ''Definition and Recommendations'' of the International Atomic Energy Agency (IAEA) restricts the dumping of the radioactive wastes that exceed specified concentration/mass limits, the acceptance of the concept of applying the release rate limits as developed by the IAEA provides a rational basis for further considering the emplacement of radioactive wastes in seabed as an attractive and acceptable alternative to terrestrial geological repositories. The technical basis for the present radiological assessment is on release rate limits and not on dumping rates. However, to meet the present requirements of the London Convention, it is necessary to express to Definition in terms of the concentration in a single site and the assumed upper limit on mass dumping rate at a single site of 100,000 tons/year with the added proviso of release rate limits for the finite ocean volume of 10^{17} m^3. This results in the concentration limits of a) 1 Ci/ton for α-emitters but limited to 10^{-1} Ci/ton ^{226}Ra and supported ^{210}Po; b) 10^2 Ci/ton for β/γ-emitters with half-lives of at least 0.5 yr (excluding ^{3}H) and the mixtures of β/γ-emitters of unknown half-lives; and c) 10^6 Ci/ton for ^{3}H and the β/γ-emitters with half-lives less than 0.5 yr.

Aquatic Ecosystems and Food Chains

762 (BNWL--165) EVALUATION OF RADIOLOGICAL CONDITIONS IN THE VICINITY OF HANFORD, JANUARY-JUNE 1965. Wilson, R.H. (ed.). (Battelle Pacific Northwest Labs., Richland, WA (USA)). Nov 1965. Contract AC06-76RL01830. 29p. NTIS.

Surveillance of the Hanford environs during the first half of 1965 showed that the amounts of radioactive materials present during this period were well within the nationally accepted limits at all times and that releases of radioactive materials from the Hanford plant were adequately controlled. ^{32}P released with Hanford effluent in the Columbia River continued to be the radioisotope that contributed the most exposure to individuals who ate locally caught fish in quantity.

763 (PB--296947, pp 471-484) USE OF HANFORD WASTE WATER PONDS BY WATERFOWL. Price, K.R.; Fitzner, R.E. (Battelle Pacific Northwest Labs., Richland, WA). May 1979. NTIS, PC A99/MF A01.

From 12. Health Physics Society midyear topical symposium; Williamsburg, VA, USA (12 Feb 1979).

Census and environmental surveillance information on waterfowl that use the Hanford Site 200 Area waste water ponds are described and evaluated. Physical features of the ponds are discussed in relation to their use and suitability for waterfowl. Seasonal distributions observed for the years 1971 through 1974 indicate that the highest use by waterfowl occurs during the spring and fall migratory periods. ðese population estimates are 300 to 400 resident waterfowl with a few tens of pairs nesting during the summer. Environmental surveillance data on ^{137}Cs in muscle tissue are presented for the years 1971 through 1977. Comparisons are made between Columbia River and waste water pond waterfowl, between waterfowl groups, and among ponds. Waterfowl collected from ponds frequently have easily detected levels of ^{137}Cs in muscle tissue. However, those waterfowl collected from the Columbia River seldom show a ^{137}Cs level above that expected from worldwide fallout. Waterfowl collected from the pond with the smallest ^{137}Cs inventory and the poorest waterfowl habitat contained the lowest levels of ^{137}Cs in muscle tissue.

764 (PNL--2462) AMERICAN COOT (FULICA AMERICANA) ON THE HANFORD SITE. PART 1. NESTING BIOLOGY. Fitzner, R.E.; Schreckhise, R.G. (Battelle Pacific Northwest Labs., Richland, WA (USA)). May 1979. Contract EY-76-C-06-1830. 40p. Dep. NTIS, PC A03/MF A01.

The nesting biology of the American coot was studied on low-level radioactive waste ponds located on the Hanford DOE Site and on control ponds located in the Columbia National Wildlife Refuge in southeastern Washington from 1974 through 1976. The objective was to discover any differences in the nesting biology of the birds which could be attributed to the low-level radioactive wastes present in the Hanford DOE Site ponds. Coots nesting on the Hanford ponds and those nesting on the wildlife refuge were found to have similar nesting habits. Nesting habitats were also similar. There were no apparent differences in nesting chronology between birds from the different study sites. Clutch size also showed no significant differences. The average number of eggs per nest for all ponds was 6.7. Egg and chick weights and percent hatching success were similar among coots from both study sites. Feeding habits of the coots from the two sites did show some differences. However, this is probably related to the availability of food items in each pond.

765 (PNL--2924) SEQUIM MARINE RESEARCH LABORATORY ROUTINE ENVIRONMENTAL MEASUREMENTS DURING CY-1978. Houston, J.R.; Blumer, P.J. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Mar 1979. Contract EY-76-C-06-1830. 24p. Dep. NTIS, PC A02/MF A01.

Environmental data collected during 1978 in the vicinity of the Marine Research Laboratory show continued compliance with all applicable state and federal regulations and furthermore show no detectable change from conditions that existed in previous years. Samples collected for radiological analysis included soil, drinking water, bay water, clams, and seaweed. Radiation dose rates at 1 meter aboveground were also measured.

766 (PNL--3127) RADIOLOGICAL SURVEY OF EXPOSED SHORELINES AND ISLANDS OF THE COLUMBIA RIVER BETWEEN VERNITA AND THE SNAKE RIVER CONFLUENCE. Sula, M.J. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Apr 1980. Contract AC06-76RL01830. 163p. NTIS, PC A08/MF A01.

This document describes a radiological survey which was performed to evaluate the magnitude and distribution of radioactive contamination on the exposed shorelines of the Columbia River along and downstream of the Hanford Site. The area encompassed by the survey includes the low-lying exposed land on both sides of the river from the uppermost point of production reactor discharge into the river at 100-B Area to the confluence of the Snake and Columbia Rivers, almost 60 miles downstream of the starting point. External exposure rate measurements were made at nearly 30,000 locations during the survey - accounting for approximately 60% of the land in the study area. Measurable radioactive contamination, resulting from past Hanford operations was found to be present on the shorelines of the Columbia River along the study area. The absence of short-lived radionuclides in the shore sediments and the presence of contamination several meters above recent maximum river levels indicate that the material was deposited some years ago.

767 (PNL--3300(Pt.2), pp 71-78) RADIOECOLOGY OF NUCLEAR FUEL CYCLES. Schreckhise, R.G.; Cadwell, L.L.; Emery, R.M. Feb 1980. Dep. NTIS, PC A11/MF A01.

Pacific Northwest Laboratory annual report for 1979 to the DOE Assistant Secretary for Environment. Part 2. Ecological sciences.

Sites where radioactive wastes are found are solid waste burial grounds, soils below liquid stoage areas, surface ditches and ponds, and the terrestrial environment around chemical processing facilities that discharge airborne radioactive debris from stacks. This study provides information to help assess the environmental impacts and certain potential, human hazards associated with nuclear fuel cycles. A data base is being developed to define and quantify biological transport routes which will permit credible predictions and assessment of routine and potential large-scale releases of radionuclides and other toxic materials. These data, used in assessment models, will increase the accuracy of estimating radiation doses to man and other life forms. Information obtained from existing storage and disposal sites will provide a meaningful radioecological perspective with which to improve the effectiveness of waste management practices. Results will provide information to determine if waste management procedures on the Hanford Site have caused ecological perturbations, and if so, to determine the source, nature, and magnitude of such disturbances.

768 (PNL-SA--8152) ARTIFICIAL RADIONUCLIDES IN THE OCEANS. Templeton, W.L. (Pacific Northwest Lab., Richland, WA (USA)). Oct 1979. Contract AC06-76RL01830. 21p. (CONF-8009234--1). NTIS, PC A02/MF A01. Order Number DE82007067.

From 3. international congress on the history of oceanography; Woods Hole, MA, USA (22 Sep 1980).

The report highlights the areas of major contributions that the nuclear era has made to the understanding of oceanography and the marine sciences, and in particular the application to the public health problems that arise through anthropogenic exploitation of the oceans for the disposal of radioactive materials. (ACR)

769 NUCLEAR WASTE PONDS AND STREAMS ON THE HANFORD SITE: AN ECOLOGICAL SEARCH FOR RADIATION EFFECTS. Emery, R.M.; McShane, M.C. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Health Physics; 38: No. 5, 787-809(May 1980).

Limnological and radiological parameters were investigated in ponds and streams on the Hanford Site to develop general radioecological profiles. All but one aquatic system receive low-level aqueous radioactive wastes from nuclear facilities. The remaining system is a pond formed by surfacing groundwater and contains radioactivity as a result of evaporative concentration of naturally occurring nuclides. Attempts were made to determine whether the amounts of radioactivity present in each aquatic system could be related to ecological variation occurring among them. Maximum dose from the sediments and nuclide concentrations in the water were used to differentiate these systems radiologically. These aquatic environments support populations of commonly occurring algae, macrophytes, invertebrates and in some cases, fish. Since these ponds and streams cannot be clearly differentiated from offsite systems or among themselves on the basis of a general ecological profile, this survey provides no conclusive evidence that the nuclear wastes discharged into Hanford ponds and streams have affected the colonization, diversity and activity of biota that appear in them.

770 ECOLOGICAL EXPORT OF PLUTONIUM FROM A REPROCESSING WASTE POND. Emery, R.M.; Klopfer, D.C.; McShane, M.C. (Battelle Pacific Northwest Labs., Richland, Wash. (USA)). Health Physics; 4: No. 3, 255-269(Mar 1978).

A reprocessing waste pond at Hanford has been inventoried to determine quantities of plutonium that have been accumulated since its formation in 1944. Expressions of export were developed from these inventory data and from informed assumptions about the vectors that act to mobilize material containing Pu. This 14-acre pond provides a realistic illustration of the mobility of Pu in a lentic ecosystem. The ecological behavior of Pu in this pond is similar to that in other contaminated aquatic systems having widely differing limnological characteristics. Since its creation, this pond has received about 1 Ci of $^{239,240}Pu$ and ^{238}Pu, most of which has been retained by its sediments. Submerged plants, mainly diatoms and Potamogeton, accumulate > 95% of the Pu contained in biota. Emergent insects are the only direct biological route of export, mobilizing about 5 x 10^3 nCi of Pu annually, which is also the estimated maximum quantity of the Pu exported by waterfowl, birds, and mammals collectively. There is no apparent significant export by wind, and it is not likely that Pu has migrated to the ground water below U-Pond via percolation. Although this pond has a rapid flushing rate, a eutrophic nutrient supply with a diverse biotic profile, and is in contact with an active terrestrial environment, it appears to effectively bind Pu and prevent it from entering pathways to man and other life.

GEOSCIENCES

Geology and Hydrology

771 (BNWL-SA--140) PROBLEMS ASSOCIATED WITH THE EXTENSION OF THE STRATIGRAPHIC UNITS OF SOUTH-CENTRAL WASHINGTON. PART II. THE POST BASALT SEDIMENTS. Brown, D.J.; Brown, R.E. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 26 Mar 1965. Contract AT(45-1)-1830. 9p. NTIS.

There are four district stratigraphic units within the Pasco Basin which if interpreted on the basis of recent field evidence and described properly could be traced over most of southeastern Washington. At the present time these post-basalt sediments are described in the literature in such a way as to make extensive correlations impossible. Some descriptions overlap more than one unit whereas others describe only a part of the unit. It is proposed that all interested geologists in the Pacific Northwest now consider reconciling the differences which exist in the names and descriptions of those sedimentary deposits in light of this new evidence.

772 (PNL--2851, pp VII.1-VII.8) PRELIMINARY SUBSURFACE HYDROLOGIC CONSIDERATIONS: COLUMBIA RIVER PLATEAU PHYSIOGRAPHIC PROVINCE. Veatch, M.D. (Shannon and Wilson, Inc., Seattle, WA). Nov 1979. Dep. NTIS, PC A09/MF A01.

Summary of FY-1978 consultation input for Scenario Methodology Development.

Subsurface hydrologic conditions in the Pacific Northwest are strongly controlled by the structural and stratigraphic framework of subregions. A significant portion of the Pacific Northwest is underlain by the Columbia River Plateau basalt sequence. This discussion is limited to hydrologic conditions as they relate to the Columbia River Plateau physiographic province and specifically to the Pasco Basin in the central part of the province.

773 (PNL--2851, pp VIII.1-VIII.9) GENERAL SUBSURFACE GEOLOGY OF SOUTHEASTERN NEW MEXICO WITH SPECIAL REFERENCE TO ITS STRUCTURAL STABILITY. Hills, J. (Univ. of Texas, El Paso). Nov 1979. Dep. NTIS, PC A09/MF A01.

Summary of FY-1978 consultation input for Scenario Methodology Development.

Southeastern New Mexico east of the Pecos River is a tectonically stable area with no major fault movements for the past 250 million years. No major tectonic adjustments are expected in the next million years. Minor tectonic adjustments during Triassic time (200 to 180 million years ago) and early Tertiary time (70 million years ago) have resulted in deepening and eastward tilting of the Delaware basin. These tectonic adjustments have resulted in the opening of joints and consequent solution of the upper part of the Salado salt. Compaction of Pennsylvanian and Lower Permian sediments has also contributed to the basin subsidence. This effect has been most severe on the west edge of the basin near the Pecos River and in the southeast edge of the basin adjoining Winkler County, Texas. Recent minor seismic activity is probably the result of petroleum secondary recovery operations. This seismic activity may result in the release of strain and may contribute to the long term tectonic stability of the area.

774 (PNL--2949) GEOHYDROLOGY AND GROUND-WATER QUALITY BENEATH THE 300 AREA, HANFORD SITE, WASHINGTON. Lindberg, J.W.; Bond, F.W. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jun 1979. Contract EY-76-C-06-1830. 165p. Dep. NTIS, PC A08/MF A01.

Ground water enters the 300 Area from the northwest, west, and southwest. However, throughout most of the 300 Area, the flow is to the east and southeast. Ground water flows to the northeast only in the southern portion of the 300 Area. Variations in level of the Columbia River affected the ground-water system by altering the level and shape of the 300 Area watertable. Large quantities of process waste water, when warmed during summer months by solar radiation or cooled during winter months by ambient air temperature, influenced the temperature of the ground water. Leaking pipes and the intentional discharge of waste water (or withdrawal of ground water) affected the ground-water system in the 300 Area. Water quality tests of Hanford ground water in and adjacent to the 300 Area showed that in the area of the Process Water Trenches and Sanitary Leaching Trenches, calcium, magnesium, sodium, bicarbonate, and sulfate ions are more dilute, and nitrate and chloride ions are more concentrated than in surrounding areas. Fluoride, uranium, and beta emitters are more concentrated in ground water along the bank of the Columbia River in the central and southern portions of the 300 Area and near the 340 Building. Test wells and routine ground-water sampling are adequate to point out contamination. The Variable Thickness Transient (VTT) Model of ground-water flow in the unconfined aquifer underlying the 300 Area has been set up, calibrated, and verified. The Multicomponent Mass Transfer (MMT) Model of distribution of contaminants in the saturated regime under the 300 Area has been set up, calibrated, and tested.

775 (PNL--3198) DURABILITY OF METALS FROM ARCHAEOLOGICAL OBJECTS, METAL METEORITES, AND NATIVE METALS. Johnson, A.B. Jr.; Francis, B. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jan 1980. Contract EY-76-C-06-1830. 117p. Dep. NTIS, PC A06/MF A01.

Metal durability is an important consideration in the multi-barrier nuclear waste storage concept. This study summarizes the ancient metals, the environments, and factors which appear to have contributed to metal longevity. Archaeological and radiochemical dating suggest that human use of metals began in the period 6000 to 7000 BC. Gold is clearly the most durable, but many objects fashioned from silver, copper, bronze, iron, lead, and tin have survived for several thousand years. Dry environments, such as tombs, appear to be optimum for metal preservation, but some metals have survived in shipwrecks for over a thousand years. The metal meteorites are Fe-base alloys with 5 to 60 wt% Ni and minor amounts of Co, I, and S. Some meteoritic masses with ages estimated to be 5,000 to 20,000 years have weathered very little, while other masses from the same meteorites are in advanced stages of weathering. Native metals are natural metallic ores. Approximately five million tonnes were mined from native copper deposits in Michigan. Copper masses from the Michigan deposits were transported by the Pleistocene glaciers. Areas on the copper surfaces which appear to represent glacial abrasion show minimal corrosion. Dry cooling tower technology has demonstrated that in pollution-free moist environments, metals fare better at temperatures above than below the dewpoint. Thus, in moderate temperature regimes, elevated temperatures may be useful rather than detrimental for exposures of metal to air. In liquid environments, relatively complex radiolysis reactions can occur, particularly where multiple species are present. A dry environment largely obviates radiolysis effects.

776 (PNL--4005) ASSESSMENT OF EFFECTIVENESS OF GEOLOGIC ISOLATION SYSTEMS. ANALYTIC MODELING OF FLOW IN A PERMEABLE FISSURED MEDIUM. Strack, O.D.L. (Pacific Northwest Lab., Richland, WA (USA)). Feb 1982. Contract AC06-76RL01830. 117p. NTIS, PC A06/MF A01. Order Number DE82009289.

An analytic model has been developed for two dimensional steady flow through infinite fissured porous media, and is implemented in a computer program. The model is the first, and major, step toward the development of a model with finite boundaries, intended for use as a tool for numerical experiments. These experiments may serve to verify some of the simplifying assumptions made in continuum models and to gain insight in the mechanics of the flow. The model is formulated in terms of complex variables and the analytic functions presented are closed-form expressions obtained from singular Cauchy integrals. An exact solution is given for the case of a single crack in an infinite porous medium. The exact solution is compared with the result obtained by the use of an independent method, which assumes Darcian flow in the crack and models the crack as an inhomogeneity in the permeability, in order to verify the simplifying assumptions. The approximate model is compared with solutions obtained from the above independent method for some cases of intersecting cracks. The agreement is good, provided that a sufficient number of elements are used to model the cracks.

777 (PNL-SA--9096) GEOSTATISTICAL MODELING OF PORE VELOCITY. Devary, J.L.; Doctor, P.G. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Mar 1981. Contract AC06-76RL01830. 38p. (CONF-810372--2). NTIS, PC A03/MF A01.

From Symposium on uncertainties associated with the regulation of the geologic disposal of high-level radioactive waste; Gatlinburg, TN, USA (9 Mar 1981).

A significant part of evaluating a geologic formation as a hazardous waste repository involves the modeling of contaminant transport in the surrounding media in the event the repository is breached. The commonly used contaminant transport models are deterministic. However, the spatial variability of hydrologic field parameters introduces uncertainties into contaminant transport predictions. This paper discusses the application of geostatistical techniques to the modeling of spatially varying hydrologic field parameters required as input to contaminant transport analyses. Kriging estimation techniques were applied to Hanford Reservation field data to calculate media conductivity and the ground-water potential gradients. These quantities were statistically combined to estimate the ground-water pore velocity and to characterize the pore velocity estimation error. Combining geostatistical modeling techniques with product error propagation techniques results in an effective stochastic characterization of ground-water pore velocity, a hydrologic parameter required for contaminant transport analyses.

778 (PNL-SA--9889) COMPUTER SIMULATION OF GEOLOGIC SYSTEMS. Foley, M.G. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jan 1982. Contract AC06-76RL01830. 9p. (CONF-820303--21). NTIS, PC A02/MF A01. Order Number DE82008678.

From Waste management conference; Tucson, AZ, USA (8 Mar 1982).

The Geologic Simulation Model (GSM) developed under the Assessment of Effectiveness of Geologic Isolation Systems (AEGIS) project at the Pacific Northwest Laboratory for the Department of Energy is a quasi-deterministic process-response model which simulates the development of the geologic and hydrologic systems of a groundwater basin for a million years into the future. Effects of natural processes on the groundwater hydrologic system are modeled principally by rate equations. The combined effects and synergistic interactions of different processes are approximated by linear superposition of their effects during discrete time intervals in a stepwise-integration approach. The results of the GSM simulations are not yet defensible. They are promising, and the general behavior of the GSM over the near-term (20,000 years) and long-term (million years) is plausible. Thus, in terms of a demonstration of the GSM technology alone, the results indicate that the development effort was a success, and this report indicates what additional effort is required to make the GSM defensible. However, the GSM is a part of a coordinated performance analysis which involves other models as well, and is intended as a primary guide to analyses to be performed in addition to that of the present system. The usefulness of the GSM results to the demonstration of a coordinated performance analysis technology must be determined by considering the validity of the results and how they may be applied realistically (unmodified) to guiding more detailed analyses. (DMC)

Geophysics

Seismology and Tectonics

779 (BNWL--47) GEOPHYSICAL SEISMIC EVALUATION STUDY AT HANFORD. Brown, R.E.; Raymond, J.R. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Dec 1964. 49p. NTIS.

A geophysical research program was conducted at Hanford to determine the feasibility and desirability of using seismic methods in geohydrologic studies. Detection and delineation were desired of seven different geological features that in some sites affect the movement of liquid radioactive wastes discharged to the ground. Rotary drilling method exceeded expectations. They permitted in-well logging methods and produced information not obtainable from cable-tool methods. By extending the information laterally, the seismic program provided line data where only point data had been available. The importance of seismic methods as an adjunct to other exploration methods used at Hanford was proven. Refraction methods proved the most usable and least expensive seismic techniques for Hanford studies and also minimized drilling needs. The use of fertilizer-grade ammonium nitrate primed with diesel oil was fully adequate and cheaper than other explosives. Surface detonation was completely satisfactory and minimized the drilling of shot holes.

780 (PNL--2851, pp XII.1-XII.13) EVIDENCE AND THEORY FOR THE PREDICTION OF TECTONIC ACTIVITY IN THE BASIN AND RANGE PROVINCE OF NEVADA AND UTAH FOR THE NEXT ONE MILLION YEARS. Lovejoy, E.M.P. Nov 1979. Dep. NTIS, PC A09/MF A01.

Summary of FY-1978 consultation input for Scenario Methodology Development.

Major conclusions of the report are: Important seismic activity in the next one million years will be restricted to the Intermountain Seismic Belt. Minor seismic activity in the same period will be restricted to the Nevada Seismic Belt, Sierra Nevada front, and Reno-Yellowstone lineament. There will be seismic inactivity in the same period in the rest of the Basin and Range Province except locally along high mountain frontal fault zones. In these zones, isostatic unloading will produce slow, secular, mild seismic activity for many millions of years to come.

781 (PNL--2851, pp XIV.1-XIV.10) ASSESSMENT OF TECTONIC HAZARDS TO WASTE STORAGE IN INTERIOR-BASIN SALT DOMES. Kehle, R. Nov 1979. Dep. NTIS, PC A09/MF A01.

Summary of FY-1978 consultation input for Scenario Methodology Development.

Salt domes in the northern Gulf of Mexico may make ideal sites for storage of radioactive waste because the area is tectonically quiet. The stability of such salt domes and the tectonic activity are discussed. (DLC)

Volcanology

782 (PNL--2882) DISRUPTIVE EVENT ANALYSIS: VOLCANISM AND IGNEOUS INTRUSION. Crowe, B.M. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Aug 1980. Contract AC06-76RL01830. 46p. NTIS, PC A03/MF A01.

An evaluation is made of the disruptive effects of volcanic activity with respect to long term isolation of radioactive waste through deep geologic storage. Three major questions are considered. First, what is the range of disruption effects of a radioactive waste repository by volcanic activity. Second, is it possible, by selective siting of a repository, to reduce the risk of disruption by future volcanic activity. And third, can the probability of repository disruption by volcanic activity be quantified. The main variables involved in the evaluation of the consequences of repository disruption by volcanic activity are the geometry of the magma-repository intersection (partly controlled by depth of burial) and the nature of volcanism. Potential radionuclide dispersal by volcanic transport within the biosphere ranges in distance from several kilometers to global. Risk from the most catastrophic types of eruptions can be reduced by careful site selection to maximize lag time prior to the onset of activity. Certain areas or volcanic provinces within the western United States have been sites of significant volcanism and should be avoided as potential sites for a radioactive waste repository. Examples of projection of future sites of active volcanism are discussed for three areas of the western United States. Probability calculations require two types of data: a numerical rate or frequency of volcanic activity and a numerical evaluation of the areal extent of volcanic disruption for a designated region. The former is clearly beyond the current state of art in volcanology. The latter can be approximated with a reasonable degree of satisfaction. In this report, simplified probability calculations are attempted for areas of past volcanic activity.

Mineralogy, Petrology, and Rock Mechanics

783 (PNL--2847) HANFORD BASALT FLOW MINERALOGY. Ames, L.L. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Sep 1980. Contract AC06-76RL01830. 469p. NTIS, PC A20/MF A01.

Mineralogy of the core samples from five core wells was examined in some detail. The primary mineralogy study included an optical examination of polished mounts, photomicrographs, chemical analyses of feldspars, pyroxenes, metallic oxides and microcrystalline groundmasses and determination from the chemical analyses of the varieties of feldspars, pyroxenes and metallic oxides. From the primary mineralogy data, a firm understanding of the average Hanford basalt flow primary mineralogy emerged. The average primary feldspar was a laboradorite, the average pyroxene was an augite and the average metallic oxide was a solid solution of ilmenite and magnetite. Secondary mineralization consisted of vug filling and joint coating, chiefly with a nontronite-beidellite clay, several zeolites, quartz, calcite, and opal. Specific flow units also were examined to determine the possibility of using the mineralogy to trace flows between core wells. These included units of the Pomona, the Umatilla and a high chromium flow just below the Hurtzinger. In the Umatilla, or high barium flow, the compositional variation of the feldspars was unique in range. The pyroxenes in the Pomona were relatively highly zoned and accumulated chromium. The high chromium flow contained chromium spinels that graded in chromium content into simple magnetites very low in chromium content. A study of the statistical relationships of flow unit chemical constituents showed that flow unit constituents could be roughly correlated between wells. The probable cause of the correlation was on-going physical-chemical changes in the source magma.

784 (PNL--2864) IDENTIFYING SUITABLE PIERCEMENT SALT DOMES FOR NUCLEAR WASTE STORAGE SITES. Kehle, R.:. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Aug 1980. Contract AC06-76RL01830. 46p. NTIS, PC A03/MF A01.

Piercement salt domes of the northern interior salt basins of the Gulf of Mexico are being considered as permanent storage sites for both nuclear and chemically toxic wastes. The suitable domes are stable and inactive, having reached their final evolutionary configuration at least 30 million years ago. They are buried to depths far below the level to which erosion will penetrate during the prescribed storage period and are not subject to possible future reactivation. The salt cores of these domes are themselves impermeable, permitting neither the entry nor exit of ground water or other unwanted materials. In part, a stable dome may be recognized by its present geometric configuration, but conclusive proof depends on establishing its evolutionary state. The evolutionary state of a dome is obtained by reconstructing the growth history of the dome as revealed by the configuration of sedimentary strata in a large area (commonly 3,000 square miles or more) surrounding the dome. A high quality, multifold CDP reflection seismic profile across a candidate dome will provide much of the necessary information when integrated with available subsurface control. Additional seismic profiles may be required to confirm an apparent configuration of the surrounding strata and an interpreted evolutionary history. High frequency seismic data collected in the near vicinity of a dome are also needed as a supplement to the CDP data to permit accurate depiction of the configuration of shallow strata. Such data must be tied to shallow drill hole control to confirm the geologic age at which dome growth ceased. If it is determined that a dome reached a terminal configuration many millions of years ago, such a dome is incapable of reactivation and thus constitutes a stable storage site for nuclear wastes.

Geochemistry

785 (PNL-SA--9848) SOLID PHASES LIMITING THE CONCENTRATION OF DISSOLVED CONSTITUENTS IN BASALT AQUIFERS OF THE COLUMBIA PLATEAU IN EASTERN WASHINGTON. Deutsch, W.J.; Jenne, E.A.; Krupka, K.M. (Pacific Northwest Lab., Richland, WA (USA)). 1981. Contract AC06-76RL01830. 8p. NTIS, PC A02/MF A01. Order Number DE82005874.

The purposes of this study were: (1) to provide information on the solid phases which are in apparent equilibrium with ground waters of basalt aquifers, and (2) to further develop the capability of geochemical modeling to support solute transport studies and performance assessments of nuclear waste repositories. The basalt aquifers of the Columbia Plateau in eastern Washington were chosen as the study area because: (1) regional ground-water analyses are readily available, (2) these basalts are a potential medium for a nuclear-waste repository, and (3) mineralogical analyses from local site studies are available.

Oceanography

786 (PNL--2851, pp IX.1-IX.8) SEA LEVEL REPORT. Schwartz, M.L. (Western Washington Univ., Bellingham). Nov 1979. Dep. NTIS, PC A09/MF A01.

Summary of FY-1978 consultation input for Scenario Methodology Development.

Study of Cenozoic Era sea levels shows a continual lowering of sea level through the Tertiary Period. This overall drop in sea level accompanied the Pleistocene Epoch glacio-eustatic fluctuations. The considerable change of Pleistocene Epoch sea level is most directly attributable to the glacio-eustatic factor, with a time span of 10^5 years and an amplitude or range of approximately 200 m. The lowering of sea level since the end of the Cretaceous Period is attributed to subsidence and mid-ocean ridges. The maximum rate for sea level change is 4 cm/y. At present, mean sea level is rising at about 3 to 4 mm/y. Glacio-eustacy and tectono-eustacy are the parameters for predicting sea level changes in the next 1 my. Glacio-eustatic sea level changes may be projected on the basis of the Milankovitch Theory. Predictions about tectono-eustatic sea level changes, however, involve predictions about future tectonic activity and are therefore somewhat difficult to make. Coastal erosion and sedimentation are affected by changes in sea level. Erosion rates for soft sediments may be as much as 50 m/y. The maximum sedimentation accumulation rate is 20 m/100 y.

GENERAL

787 (BNWL--122, pp 198-201) REACTOR EFFLUENT MONITORING-1964. Olson, P.A.; Nakatani, R.E. (Battelle-Northwest, Richland, WA). Jan 1965. NTIS.

Hanford biology research. Annual report for 1964.

Since monitoring of Hanford's reactor effluent started in 1945, the biological effects of the primary factors (heat, chemical toxicity, and radiation) have been defined in terms of growth and mortaility for young chinook salmon. Comparisons between groups of chinook fingerlings reared in regular and cooled reactor effluent showed the importance of the accumulative effect of small temperature increments. Higher mortalities were observed in the present studies with local stock chinooks than in past studies with Puget Sound chinooks, due, perhaps, to physiological differences in racial stocks. Data are included on the content of ^{32}P and ^{65}Zn in fish reared in regular effluent for 10 weeks.

788 (BNWL--197) PURIFICATION OF RARE EARTHS FROM HANFORD WASTES BY SOLVENT EXTRACTION. Bray, L.A.; Roberts, F.P. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Feb 1966. 31p. NTIS.

Laboratory and hot-cell experiments showed that a good separation of Mn and the rare earths is possible by controlled pH solvent extraction. Because pH control is very difficult in the plant, unreliable results and high product losses occurred in the full-scale tests; the D2EHPA solvent extraction scheme for Mn removal proved impractical. An alternate scheme, based on sulfate precipitation was developed. This process, although complicated by the lack of a centrifuge in the plant, does not require pH control. Preliminary results indicated 85 to 95% rare earth recovery and an adequate Mn decontamination factor of 50 to 100. The separation of cerium from trivalent rare earths with D2EHPA was improved by the use of a straight chain hydrocarbon diluent. The effects of radiolysis were markedly reduced and high cerium distribution ratios were maintained.

789 (BNWL--2023) BIOLOGICAL EFFECTS OF ACTIVATION PRODUCTS AND OTHER CHEMICALS RELEASED FROM FUSION POWER PLANTS. Strand, J.A.; Poston, T.M. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Sep 1976. Contract E(45-1)-1830. 40p. Dep. NTIS $5.00.

Literature reviews indicate that existing information is incomplete, often contradictory, and of questionable value for the prediction and assessment of ultimate impact from fusion-associated activation products and other chemical releases. It is still uncertain which structural materials will be used in the blanket and first wall of fusion power plants. However, niobium, vanadium, vanadium-chromium alloy, vanadium-titanium alloy, sintered aluminum product, and stainless steel have been suggested. The activation products of principal concern will be the longer-lived isotopes of ^{26}Al, ^{49}V, ^{51}Cr, ^{54}Mn, ^{55}Fe, ^{58}Co, ^{60}Co, ^{93}Nb, and ^{94}Nb. Lithium released to the environment either during the mining cycle, from power plant operation or accident, may be in the form of a number of compound types varying in solubility and affinity for biological organisms. The effects of a severe liquid metal fire or explosion involving Na or K will vary according to inherent abiotic and biotic features of the affected site. Saline, saline-alkaline, and sodic soils of arid lands would be particularly susceptible to alkaline stress. Beryllium released to the environment during the mining cycle or reactor accident situation could be in the form of a number of compound types. Adverse effects to aquatic species from routine chemical releases (biocides, corrosion inhibitors, dissolution products) may occur in the discharge of both fission and fusion power plant designs.

790 (BNWL--2042) SrF_2 CAPSULE DESIGN FOR HEAT ENGINE APPLICATIONS. Lester, D.H. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Apr 1976. Contract EY-76-C-06-1830. 53p. Dep. NTIS, PC A04/MF A01.

Portions of document are illegible.

A number of design changes were considered to improve heat transfer characteristics of the WESF capsule. This capsule was evaluated in a design concept for use as a heat source in a helium-working fluid, Stirling heat engine. Throughout the study a heat block concept was used. The helium was assumed to be at 1200°F and 200 atm. The upper temperature limit at the fuel-metal interface was assumed to be 800°C because of material compatibility considerations. A 0.6-in. thick outer can was considered since it may be required for impact resistance and high pressure accident environments. The modifications considered were: (1) filling all gaps with helium rather than air, (2) filling gaps with powdered metal, and (3) adding a third can to the existing capsule. Also, enhancement of emissivity on metal surfaces was considered as a possible modification.

791 (BNWL--2064) SOURCE EFFICIENCY CALCULATIONS FOR ^{137}CS IRRADIATORS. Libby, R.A. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Sep 1977. Contract EY-76-S-06-1830. 26p. Dep. NTIS, PC A03/MF A01.

The efficiency of radiation sources used for radiation treatment is important in the overall economics of irradiators. The report examines design optimization for a ^{137}Cs radiation source. The initial source design used is currently being fabricated at the Waste Encapsulation and Storage Facility (WESF). Although the current source was intended for long term storage and isolation of the cesium, it has potential for use as an irradiator. Modifications to the source which were examined include: optimum cylinder dia is 1 in.; optimum stainless steel wall thickness is about 0.1 in., thicker walls will reduce source efficiency, while thinner walls increase the likelihood of wall failure; and a cesium-titanate ceramic combines the advantages of good source efficiency with low solubility and low leachability. The cost to modify the present WESF facility does not appear to justify these design changes. However, new facilities could incorporate this design to obtain optimum source efficiency.

792 (BNWL--2100(Pt.5), pp 29-50) TECHNOLOGY OVERVIEW. Jun 1977.

Pacific Northwest Laboratory annual report for 1976 to the ERDA Assistant Administrator for Environment and Safety. Part 5. Control technology, overview, safety, and policy analysis.

The Integrated Assessment Program, funded by the ERDA Division of Technology Overview, is the mechanism by which health, environmental, social, economic and institutional factors are combined into a form useful for energy planning and decision making. This program selectively combines information about effects of alternative energy technologies (such as waste releases, land and water use, and social effects) to produce broad-based assessments of the advantages and disadvantages of energy and conservation options. As a corollary, needs for further research, development, and technology transfer are identified. The program is focused on four interrelated activities: supporting systems analysis to develop and improve methods for use in assessing and comparing impacts of energy and conservation options; integrated technological impact assessment, applying these methods to help select technologies for development that are safe, clean, and environmentally acceptable; regional comparative assessments, applying the results of the technological impact assessments to identification of regional energy strategies; and a regional outreach effort to assist regional and state agencies in their energy planning programs.

793 (BNWL--2266) COST ANALYSIS FOR APPLICATION OF SOLIDIFIED WASTE FISSION PRODUCT CANISTERS IN U.S. ARMY STEAM PLANTS. Sande, W.E.; Bjorklund, W.J.; Brooks, N.A. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Apr 1977. Contract EY-76-C-06-1830. 165p. Dep. NTIS, PC A08/MF A01.

The main objectives of the present study are to design steam plants using projected waste fission product canister characteristics, to analyze the overall impact and cost/benefit to the nuclear fuel cycle associated with these plants, and to develop plans for this application if the cost analysis so warrants it. The construction and operation of a steam plant fueled with waste fission product canisters would require the involvement and cooperation of various government agencies and private industry; thus the philosophies of these groups were studied. These philosophies are discussed, followed by a forecast of canister supply, canister characteristics, and strategies for Army canister use. Another section describes the safety and licensing of these steam plants since this affects design and capital costs. The discussion of steam plant design includes boiler concepts, boiler heat transfer, canister temperature distributions, steam plant size, and steam plant operation. Also, canister transportation is discussed since this influences operating costs. Details of economics of Army steam plants are provided including steam plant capital costs, operating costs, fuel reprocessor savings due to Army canister storage, and overall economics. Recommendations are made in the final section.

794 (BNWL-CC--2297) STRONTIUM FLUORIDE FLOWSHEET DEVELOPMENT STUDIES FOR THE WASTE PACKAGING PROGRAM. Fullam, H.T.; Bray, L.A. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 25 May 1972. Contract EY-76-C-06-1830. 31p. Dep. NTIS, PC A03/MF A01.

It was demonstrated that the modified version of the Martin flowsheet can produce SrF_2 of acceptable quality using a simulated feed of the same composition as the anticipated packaging plant feed. By careful control of operating conditions, strontium losses in the process can be reduced to an acceptable level. Using a nonrepresentative radioactive feed (which contained impurities not present in the simulated feed) high strontium losses were encountered in the feed purification step, although the purification obtained was excellent. The other steps in the process presented no difficulties. Additional tests of the feed purification step will be necessary when a representative feed is available to ensure that strontium losses can be reduced to an acceptable level. The proposed flowsheet differs from the Martin flowsheet in that a feed purification step is used in the proposed process. All the remaining steps in the process are essentially identical and the data and recommendations reported in the Martin publications appear applicable to the proposed process. Based on the results of this study (and including data published by the Martin Company), a detailed flowsheet with recommended operating conditions is given.

795 (BNWL-SA--6070) CHEMICAL DURABILITY OF ZINC BOROSILICATE NUCLEAR WASTE GLASS. Westsik, J.H. Jr.; Mendel, J.E. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Mar 1977. Contract EY-76-C-06-1830. 20p. (CONF-770416--10). Dep. NTIS, PC A02/MF A01.

From 79. annual meeting of American Ceramic Society; Chicago, IL, USA (23 Apr 1977).

Chemical durability is of primary concern when evaluating the safety of waste glass. For this reason, testing the leachability of waste glasses is a fundamental part of their development and characterization. The leachability is also very much a function of glass composition as previously discussed. This discussion is limited to a representative waste glass composition, a high-zinc borosilicate formulation which has been studied in detail by Battelle Pacific Northwest Laboratories. (CHT)

796 (BNWL-SA--6302) SPECIATION STUDIES OF RADIONUCLIDES IN LOW LEVEL WASTES AND PROCESS WATERS FROM PRESSURIZED WATER REACTOR. Abel, K.H.; Robertson, D.E.; Crecelius, E.A.; Silker, W.B. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Oct 1977. Contract EY-76-C-06-1830. 7p. (CONF-771113--14). Dep. NTIS, MF A01.

From 4. joint conference on sensing of environmental pollutants; New Orleans, LA, USA (6 Nov 1977).

Portions of document are illegible.

Physicochemical characterization studies of aqueous process streams at San Onofre Generating Station Unit No. 1 are providing important source term information concerning the forms of radionuclides being released into the marine environment. The primary coolant, secondary steam condensate, processed low level wastes and tertiary coolant were sampled at several different times in the nuclear fuel cycle. Rdionuclides were partitioned into particulate, cationic, anionic, and nonionic species in the reactor process streams, and into particulate and soluble species in the tertiary seawater coolant. Characterization of the particulate species has included a detailed size distribution. The purpose of this research was to provide information concerning chemical and physical forms of the radionuclides being released to the coastal zone from a nuclear generating station in order to facilitate design of radiobiological studies necessary for assessment of their environmental significance.

797 (NUREG/CR--0572) APPLICATION OF FISHERIES MANAGEMENT TECHNIQUES TO ASSESSING IMPACTS: TASK I REPORT. McKenzie, D.H.; Baker, K.S.; Fickeisen, D.H.; Metzger, R.M.; Skalski, J.R. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Mar 1979. Contract EY-76-C-06-1830. 89p. (PNL--2811). Dep. NTIS, PC A05/MF A01.

Task I efforts examined the available fisheries management techniques and assessed their potential application in a confirmatory monitoring program. The objective of such monitoring programs is to confirm that the prediction of an insignificant impact (usually made in the FES) was correct. Fisheries resource managers have developed several tools for assessing the fish population response to stress (exploitation) and they were thought potentially useful for detecting nuclear power plant impacts. Techniques in three categories were examined: catch removal, population dynamics, and nondestructive censuses, and the report contains their description, examples of application, advantages, and disadvantages. The techniques applied at nuclear power plant sites were examined in detail to provide information on implementation and variability of specific approaches. The most suitable techniques to incorporate into a monitoring program confirming no impact appear to be those based on Catch Per Unity Effort (CPUE) and hydroacoustic data. In some specific cases, age and growth studies and indirect census techniques may be beneficial. Recommendations for task II efforts to incorporate these techniques into monitoring program designs are presented. These include development of guidelines for: (1) designing and implementing a data collection program; (2) interpreting these data and assessing the occurrence of impact, and (3) establishment of the monitoring program's ability to detect changes in the affected populations.

798 (NUREG/CR--0798) EVALUATION OF EMPIRICAL ATMOSPHERIC DIFFUSION DATA. Horst, T.W.; Doran, J.C.; Nickola, P.W. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Oct 1979. Contract EY-76-C-06-1830. 151p. (PNL--2599). Dep. NTIS, PC A08/MF A01.

A study has been made of atmospheric diffusion over level, homogeneous terrain of contaminants released from non-buoyant point sources up to 100 m in height. Current theories of diffusion are compared to empirical diffusion data, and specific dispersion estimation techniques are recommended which can be implemented with the on-site meteorological instrumentation required by the Nuclear Regulatory Commission. A comparison of both the recommended diffusion model and the NRC diffusion model with the empirical data demonstrates that the predictions of the recommended model have both smaller scatter and less bias, particularly for groundlevel sources.

799 (PB--296947, pp 356-365) ASSESSMENT OF RADON PROGENY INHALATION EXPOSURE FROM LOW-LEVEL WASTES OF PHOSPHATE MINING IN FLORIDA. Fisher, D.R. (Battelle Pacific Northwest Labs., Richland, WA); Roessler, C.E. May 1979. NTIS, PC A99/MF A01.

From 12. Health Physics Society midyear topical symposium; Williamsburg, VA, USA (12 Feb 1979).

The redistribution of naturally-occurring uranium series radionuclides as a result of phosphate mining, processing, product use, and waste disposal presents several potential radiation pathways to man. Of particular importance is exposure to radon-222 progeny in structures built on reclaimed lands in Florida. Indoor radon daughter sampling data from Polk County were analyzed, and the data categorized by land and structure type. The average population-weighted concentration in about 4400 homes was determined to be about 0.009 working level (WL) in addition to a background of 0.0003 WL. It was also determined that the average annual cumulative indoor exposure on reclaimed land was approximately 0.02 working level months (WLM). A relatively small number of houses on high-activity overburden accounted for 38% of the total population exposure. A generally applicable model is proposed to relate lund cancer risk to the average annual exposure, the risk coefficient, the expected lung cancer mortality from all other causes, the duration of the exposure and the number of years for observation of the effects. Health risk estimates were performed for present levels and population size, and also for several scenarios anticipating new growth and construction - with and without imposed standards to limit indoor radon progeny levels. The model suggests that for an equilibrium condition, about one additional case of radiogenic lung cancer every two years in the Polk County population might be expected.

800 (PB--296947, pp 485-491) CESIUM-137 IN COOTS (FULICA AMERICANA) ON HANFORD WASTE PONDS: CONTRIBUTION TO POPULATION DOSE AND OFFSITE TRANSPORT ESTIMATES. Cadwell, L.L.; Schreckhise, R.G.; Fitzner, R.E. (Battelle Pacific Northwest Labs., Richland, WA). May 1979. NTIS, PC A99/MF A01.

From 12. Health Physics Society midyear topical symposium; Williamsburg, VA, USA (12 Feb 1979).

American coots (Fulica americana) were periodically collected from ponds receiving low-level radioactive waste on the Hanford Site and from ponds on a control area. Gut contents and selected tissues were removed and analyzed for ^{137}Cs, ^{90}Sr and gross Pu. The concentration of ^{137}Cs in coot flesh was the highest of the radioelements measured. Coots from the pond receiving the greatest quantity of fission products had the highest ^{137}Cs concentrations. One- and fifty-year ^{137}Cs total-body dose commitments to an individual from ingesting all edible tissues of one Gable Mountain Pond coot were calculated at 1.9 and 2.1 mrem, respectively. The 50-year population dose from the projected harvest of Gable Mountain Pond coots during one year was estimated to be 0.13 person-rem, which is likely high because the coots must leave the Site to be harvested. Total ^{137}Cs export from Hanford Site ponds via coots, having an average body burden of 0.092 μCi, was estimated to be 46 μCi per year.

801 (PNL--1845-37, pp 27-30) TERRESTRIAL RADIOISOTOPE APPLICATIONS PROGRAM. Van Tuyl, H.H.; Sande, W.E. Jan 1978.

Quarterly report on the strontium heat source development program and the terrestrial radioisotope applications program, Advanced Systems and Materials Production Division for October--December 1977.

The major objective of this program is to identify potential beneficial uses of nuclear reactor by-product through three principal tasks: (1) evaluation of long-term availability and cost of useful isotopes from commercial suppliers or military wastes; (2) evaluation of beneficial applications of isotopes; and (3) identification and evaluation of the actions required to optimize the $^{90}SrF_2$ and $^{137}CsCl$ products from the Hanford Waste Encapsulation and Storage Facility (WESF) for beneficial uses.

802 (PNL--1845-37, pp 1-25) STRONTIUM HEAT SOURCE DEVELOPMENT PROGRAM. Van Tuyl, H.H.; Fullam, H.T.; Atteridge, D.G.; Simonen, F.A. Jan 1978.

Quarterly report on the strontium heat source development program and the terrestrial radioisotope applications program, Advanced Systems and Materials Production Division for October--December 1977.

At Hanford, strontium is separated from the high-level waste, converted to the fluoride, and doubly encapsulated in small, high-integrity containers for subsequent long-term storage. The fluoride conversion, encapsulation and storage take place in the Waste Encapsulation and Storage Facilities (WESF). The encapsulated strontium fluoride represents an economical source of ^{90}Sr if the WESF capsule can be licensed for heat source applications under anticipated use conditions. The objectives of this program are to obtain the data needed to license $^{90}SrF_2$ heat sources and specifically the WESF $^{90}SrF_2$ capsules. Topics discussed include chemical and physical properties of $^{90}SrF_2$, $^{90}SrF_2$ compatibility studies, and capsule qualification and licensing.

803 (PNL--1845-50) QUARTERLY REPORT ON THE STRONTIUM HEAT SOURCE DEVELOPMENT PROGRAM, SPACE AND TERRESTRIAL SYSTEMS DIVISION FOR JANUARY-MARCH 1981. Fullam, H.T. (Battelle Pacific Northwest Labs., Richland, WA (USA)). May 1981. Contract AC06-76RL01830. 13p. NTIS, PC A02/MF A01. Order Number DE81028404.

At Hanford, strontium is separated from the high-level waste, converted to the fluo-ride, and doubly encapsulated in small, high-integrity containers for subsequent long-term storage. The fluoride conversion, encapsulation, and storage takes place in the Waste Encapsulation and Storage Facilities

(WESF). The encapsulated strontium fluoride represents an economical source of ^{90}Sr if the WESF capsule can be licensed for heat-source applications under anticipated-use conditions. The objects of this program are to obtain the data needed to license $^{90}SrF_2$ heat sources and specifically the WESF $^{90}SrF_2$ capsules. The information needed for licensing can be divided into three general task areas: Task 1--Chemical and Physical Properties of $^{90}SrF_2$; Task 2--$^{90}SrF_2$ Compatibility Studies; and Task 3--Capsule Qualification and Licening. Efforts are proceeding concurrently on all three tasks to obtain the required information.

804 (PNL--2286(App.)) DESCRIPTIONS OF REFERENCE LWR FACILITIES FOR ANALYSIS OF NUCLEAR FUEL CYCLES. APPENDIXES. Schneider, K.J.; Kabele, T.J. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Sep 1979. Contract EY-76-C-06-1830. 232p. Dep. NTIS, PC A11/MF A01.

The appendixes present the calculations that were used to derive the release factors discussed for each fuel cycle facility in Volume I. Appendix A presents release factor calculations for a surface mine, underground mine, milling facility, conversion facility, diffusion enrichment facility, fuel fabrication facility, PWP, BWR, and reprocessing facility. Appendix B contains additional release factors calculated for a BWR, PWR, and a reprocessing facility. Appendix C presents release factors for a UO_2 fuel fabrication facility.

805 (PNL--2429) ENFORM II: A CALCULATIONAL SYSTEM FOR LIGHT WATER REACTOR LOGISTICS AND EFFLUENT ANALYSIS. Heeb, C.M.; Lewallen, M.A.; Purcell, W.L.; Cole, B.M. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Sep 1979. Contract EY-76-C-06-1830. 133p. Dep. NTIS, PC A07/MF A01.

ENFORM is a computer-based information system that addresses the material logistics, environmental releases and economics of light water reactor (LWR) operation. The most important system inputs consist of electric energy generation requirements, details of plant construction scheduling, unit costs, and environmental release factors. From these inputs the ENFORM system computes the mass balances and generates the environmental release information for noxious chemicals and radionuclides from various fuel cycle facilities (except waste disposal). Fuel cycle costs and electric power costs are also computed. All code development subsequent to 1977 is summarized. Programming instructions are provided for the modules that are comprised in the ENFORM system. ENGEN, a code that uses a generation schedule specified by the user and isotopic data generated by ORIGEN, has been developed to produce a scenario-specific data base. Other codes (ENMAT, ENRAD, etc) have been developed to use data base information to estimate radioactive and nonradioactive release information.

806 (PNL--2719) FUSION FUEL CYCLE SOLID RADIOACTIVE WASTES. Gore, B.F.; Kaser, J.D.; Kabele, T.J. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jun 1978. Contract EY-76-C-06-1830. 39p. Dep. NTIS, PC A03/MF A01.

Eight conceptual deuterium-tritium fueled fusion power plant designs have been analyzed to identify waste sources, materials and quantities. All plant designs include the entire D-T fuel cycle within each plant. Wastes identified include radiation-damaged structural, moderating, and fertile materials; getter materials for removing corrosion products and other impurities from coolants; absorbents for removing tritium from ventilation air; getter materials for tritium recovery from fertile materials; vacuum pump oil and mercury sludge; failed equipment; decontamination wastes; and laundry waste. Radioactivity in these materials results primarily from neutron activation and from tritium contamination. For the designs analyzed annual radwaste volume was estimated to be 150 to 600 m^3/GWe. This may be compared to 500 to 1300 m^3/GWe estimated for the LMFBR fuel cycle. Major waste sources are replaced reactor structures and decontamination waste.

807 (PNL--2749) SAFETY ANALYSIS REPORT FOR THE NEUTRON MULTIPLIER FACILITY, 329 BUILDING. Rieck, H.G. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Sep 1978. Contract EY-76-C-06-1830. 118p. Dep. NTIS, PC A06/MF A01.

Neutron multiplication is a process wherein the flux of a neutron source such as ^{252}Cf is enhanced by fission reactions that occur in a subcritical assemblage of fissile material. The multiplication factor of the device depends upon the consequences of neutron reactions with matter and is independent of the initial number of neutrons present. Safe utilization of such a device demands that the fissile material assemblage be maintained in a subcritical state throughout all normal and credibly abnormal conditions. Examples of things that can alter the multiplication factor (and degree of subcriticality) are temperature fluctuations, changes in moderator material such as voiding or composition, addition of fissile materials, and change in assembly configuration. The Neutron Multiplier Facility (NMF) utilizes a multiplier-^{252}Cf assembly to produce neutrons for activation analysis of organic and inorganic environmental samples and for on-line mass spectrometry analysis of fission products which diffuse from a stationary fissile target (less than or equal to 4 g fissile material) located in the Neutron Multiplier. The NMF annex to the 329 Building provides close proximity to related counting equipment, and delay between sample irradiation and counting is minimized.

808 (PNL--2821) EFFECTS OF THE BEN FRANKLIN DAM ON THE HANFORD SITE. Harty, H. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Apr 1979. Contract EY-76-C-06-1830. 230p. Dep. NTIS, PC A11/MF A01.

A previous assessment of the effects of a Ben Franklin Dam on the Hanford Site made in 1967 was updated so that the potential adverse effects may be better understood in light of existing operations, current environmental and safety standards, and proposed facilities and operations. The major effects would probably arise from flooding of portions of the site by the reservoir associated with the dam and by the raising of the ground water table under the site. A preliminary analysis of the effects of the dam is presented, and a number of studies are recommended in order to fully evaluate and understand these potential impacts. The following seven tasks are identified and discussed: groundwater - hydrology analysis; soil liquefaction analysis; hydrostatic uplift and soil effects on structures; assessment of the potential for landsliding and sloughing; facility decommissioning; hydrothermal analysis; and, meteorological effects. Four other aspects commented upon in this report are: aquatic ecology, terrestrial ecology, socioeconomic effects, and public interaction. Possible effects on ongoing DOE-sponsored R and D are also noted. To the extent possible, cost estimates are developed for corrective actions which must be taken on the Hanford Site to accommodate the dam. Where this was not possible, appropriate courses of action leading to cost estimates are presented.

809 (PNL--2830) MATERIALS FLOW, RECYCLE AND DISPOSAL FOR DEUTERIUM--TRITIUM FUSION. Willenberg, H.J.; Kabele, T.J.; May, R.P.; Willingham, C.E. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Dec 1978. Contract EY-76-C-06-1830. 106p. Dep. NTIS, PC A06/MF A01.
The flow of materials to and from a deuterium--tritium fusion power plant is investigated. Three tokamak conceptual reactor designs are described and materials requirements are detailed. Various process options are considered for the reprocessing segment of materials cycle. A conceptual materials resource cycle is selected from these options and details of a conceptual reprocessing segment are described. Environmental control technology considerations associated with the reprocessing segment are outlined.

810 (PNL--2935, pp 101-109) ALTERNATIVE FUELS AND TECHNOLOGIES. Shupe, J.W.; Kreid, D.K.; DeHaan, G.L. Jun 1979.
Compressed Air Energy Storage Technology Program. Annual report, 1978.
During 1978, two alternatives were being studied: utilization of nuclear waste decay heat and utilization of coal-fired technologies. In addition, a cursory examination of the benefits of integrating CAES with magnetohydrodynamics was made. The nuclear waste heat and magnetohydrodynamics studies were completed this past year. The coal-fired technology studies are continuing, and a report of the preliminary findings will be contained in a topical report to be prepared during FY-1979.

811 (PNL--3159) NUCLEAR ENERGY INFORMATION FLOW FROM DOE TO THE PUBLIC. Simmons, J.L.; Rankin, W.L.; Nealey, S.M. (Battelle Human Affairs Research Center, Seattle, WA (USA)). Jun 1980. Contract AC06-76RL01830. 43p. (B-HARC--411/014). NTIS, PC A03/MF A01.
The objective of this research was to study the DOE's program for educating the public about nuclear power and nuclear waste management. DOE's organizational structuree and the procedures used within this structure to disseminate information were studied and readability tests on nuclear information distributed by DOE were conducted. Initial information was obtained through interviews with 29 local, state, and federal DOE representatives. This was supplemented with additional information as it was released by the DOE. The primary goals of the DOE's information program are to encourage two-way communication between the DOE and the public and to encourage public participation in policy-making decisions. Most of this communication, however, is presented orally. Relative to other energy technologies and conservation, very few nuclear brochures are currently being distributed by the DOE. This is especially true with regard to information about nuclear waste. A recent public survey found that a majority of the public wants to learn more about nuclear power and that, with regard to the nuclear fuel cycle, the public wants most to learn about nuclear waste management. Thus, the DOE appears to be missing an eager audience.

812 (PNL--3182) FUSION FUEL CYCLE: MATERIAL REQUIREMENTS AND POTENTIAL EFFLUENTS. Teofilo, V.L.; Bickford, W.E.; Long, L.W.; Price, B.A.; Mellinger, P.J.; Willingham, C.E.; Young, J.K. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Oct 1980. Contract AC06-76RL01830. 176p. NTIS, PC A09/MF A01.
Environmental effluents that may be associated with the fusion fuel cycle are identified. Existing standards for controlling their release are summarized and anticipated regulatory changes are identified. The ability of existing and planned environmental control technology to limit effluent releases to acceptable levels is evaluated. Reference tokamak fusion system concepts are described and the principal materials required of the associated fuel cycle are analyzed. These materials include the fusion fuels deuterium and tritium; helium, which is used as a coolant for both the blanket and superconducting magnets; lithium and beryllium used in the blanket; and niobium used in the magnets. The chemical and physical processes used to prepare these materials are also described.

813 (PNL--3276) SIMULATION OF ATMOSPHERIC TURBULENCE. Cliff, W.C.; Hall, D.L. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jan 1980. Contract EY-76-C-06-1830. 29p. Dep. NTIS, PC A03/MF A01.
A method is described for constructing an artificial temporal sequence of velocity values for simulating atmospheric turbulence. The method develops the conditional probability density distribution of velocity given that the previous value of velocity is known. A value is randomly selected from the conditional probability density which then serves as a known value for the next conditional probability density function. Continuing this procedure provides the desired simulated sequence of velocity values. Based upon this simulation method a program is developed for the Hewlett-Packard HP-29C hand-held calculator for simulating the longitudinal component of atmospheric turbulence during neutral or strong wind conditions.

814 (PNL--3300(Pt.5), pp 19-20) ENERGY MATERIAL TRANSPORT, NOW THROUGH 2000. Feb 1980. Dep. NTIS, PC A07/MF A01.
Pacific Northwest Laboratory annual report for 1979 to the DOE Assistant Secretary for Environment. Part 5. Environmental assessment, control, health, and safety.
The overall objectives of this project are to assess potential problems which may inhibit the safe and environmentally acceptable development of nuclear and fossil energy material transportation systems from now to the year 2000 and to recommend research, development, and other necessary action to mitigate the potential adverse impact of these problems. Effort in FY 1979 addressed the domestic transportation of petroleum, synfuels, nuclear fuel cycle materials, and legal and regulatory concerns. Results of the studies on petroleum, synfuels, and legal and regulatory concerns were published. Research continued on spent fuel cask productivity, coal sludge transportation, the Northern Tier Pipeline, and joint utility corridors.

815 (PNL--3511) LMFBR CONCEPTUAL DESIGN STUDY: AN OVERVIEW OF ENVIRONMENTAL AND SAFETY CONCERNS. Brenchley, D.L. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jun 1981. Contract AC06-RL01830. 108p. NTIS,

PC A06/MF A01. Order Number DE81028260.
The US Department of Energy (DOE) initiated the Liquid Metal Fast Breeder (LMFBR) Conceptual Design Study (CDS) with the objective of maintaining a viable breeder option. The project is scheduled to be completed in FY-1981 but decisions regarding plant construction will be delayed until at least 1985. This report provides a review of the potential environmental and safety engineering concerns for the CDS and recommends specific action for the Environmental and Safety Engineering Division of DOE.

816 (PNL--3562) OVERVIEW OF ENVIRONMENTAL CONTROL ASPECTS FOR THE GAS-COOLED FAST REACTOR. Nolan, A.M. (Battelle Pacific Northwest Labs., Richland, WA (USA)). May 1981. Contract AC06-76RL01830. 29p. NTIS, PC A03/MF A01.
Environmental control aspects relating to release of radionuclides have been analyzed for the Gas-Cooled Fast Reactor (GCFR). Information on environmental control systems was obtained for the most recent GCFR designs, and was used to evaluate the adequacy of these systems. The GCFR has been designed by the General Atomic Company as an alternative to other fast breeder reactor designs, such as the Liquid Metal Fast Breeder Reactor (LMFBR). The GCFR design includes mixed oxide fuel and helium coolant. The environmental impact of expected radionuclide releases from normal operation of the GCFR was evaluated using estimated collective dose equivalent commitments resulting from 1 year of plant operation. The results were compared to equivalent estimates for the Light Water Reactor (LWR) and High-Temperature Gas-Cooled Reactor (HTGR). A discussion of uncertainties in system performances, tritium production rates, and radiation quality factors for tritium is included.

817 (PNL--3584) IMPACTS OF URANIUM-UTILIZATION IMPROVEMENTS ON LIGHT-WATER-REACTOR RADIONUCLIDE RELEASES. Aaberg, R.L. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Aug 1981. Contract AC06-76RL01830. 26p. NTIS, PC A03/MF A01. Order Number DE81028363.
This report discusses potential changes to radionuclide releases as a result of uranium-saving plant modifications and altered operating practices. Only releases to the environment from routine operation are considered; releases resulting from abnormal conditions outside the technical specifications covering plant operation are not considered.

818 (PNL--3700(Pt.5), pp 59) HANDBOOKS ON EFFLUENT AND ENVIRONMENTAL MONITORING. Feb 1981. NTIS, PC A05/MF A01.
Pacific Northwest Laboratory annual report for 1980 to the DOE Assistant Secretary for Environment. Part 5. Environmental assessment, control, health and safety.
An updating of the Guide for Environmental Radiological Monitoring at DOE Installations was submitted to the Department of Energy (DOE) Environmental Protection Branch for comment. Issue of the draft Guide for Effluent Monitoring at DOE Installations was again delayed because of further regulatory requirements. An analysis and evaluation of CY 1978 annual environmental surveillance reports from DOE nuclear sites was submitted to the Environmental Protection Branch. A draft Executive Summary of 1978 environmental impacts from all DOE nuclear sites was 90% completed.

819 (PNL--3700(Pt.5), pp 45-48) ANALYSIS OF FUSION FUEL CYCLES. Feb 1981. NTIS, PC A05/MF A01.
Pacific Northwest Laboratory annual report for 1980 to the DOE Assistant Secretary for Environment. Part 5. Environmental assessment, control, health and safety.
The objective of this task is to undertake an independent review and analysis of current fusion facilities that have or will generate tritium wastes and effluents to the environment. Consideration is given to those fusion facilities that are in operation or under construction. These include the magnetic fusion facilities: Tokamak Fusion Test Reactor (TFTR), the Tritium Systems Test Assembly (TSTA), and the Fusion Materials Irradiation Test Facility (FMIT). Facilities in the Inertial Confinement Fusion area will also be reviewed and analyses made of the tritium control problems. Based on these analyses, experimental data needs to support the environmental control technology of fusion facilities will be evaluated.

820 (PNL--3700(Pt.5), pp 65-68) HEALTH PHYSICS LEAD LABORATORY. Feb 1981. NTIS, PC A05/MF A01.
Pacific Northwest Laboratory annual report for 1980 to the DOE Assistant Secretary for Environment. Part 5. Environmental assessment, control, health and safety.
Pacific Northwest Laboratory functions as the lead laboratory providing health physics support and assistance to the Division of Operational and Environmental Safety, US DOE, on special studies principally associated with the analysis of impact of standards, regulations, and engineering and administrative actions on occupational and environmental exposure. Included in this support are data base management for personnel dosimetry; assessment and analysis of personnel neutron dosimetry; field testing, calibration, and standardization techniques for neutron spectrometry; preparation of a guide for reducing radiation exposures; and radiation surveys of nuclear facilites and characterization of DOE facilities emergency preparedness. (DLS)

821 (PNL--3725) SURVEY OF TRITIUM WASTES AND EFFLUENTS IN NEAR-TERM FUSION-RESEARCH FACILITIES. Bickford, W.E.; Dingee, D.A.; Willingham, C.E. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Aug 1981. Contract AC06-76RL01830. 84p. NTIS, PC A05/MF A01. Order Number DE81029613.
The use of tritium control technology in near-term research facilities has been studied for both the magnetic and inertial confinement fusion programs. This study focused on routine generation of tritium wastes and effluents, with little referene to accidents or facility decommissioning. This report serves as an independent review of the effectiveness of planned control technology and radiological hazards associated with operation. The facilities examined for the magnetic fusion program included Fusion Materials Irradiation Testing Facility (FMIT), Tritium Systems Test Assembly (TSTA), and Tokamak Fusion Test Reactor (TFTR) in the magnetic fusion program, while NOVA and Antares facilities were examined for the inertial confinement program.

822 (PNL-SA--7381) IMPACT OF LWR DECONTAMINATION ON RADWASTE SYSTEMS. Perrigo, L.D.; Divine, J.R. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 1979.

Contract EY-76-C-06-1830. 13p. (CONF-790313--5). Dep. NTIS, PC A02/MF A01.

From NACE/Corrosion 1979 meeting; Atlanta, GA, USA (12 Mar 1979).

Increased radiation levels around certain reactors in the United States and accompanying increases in personnel exposures are causing a reexamination of options available to utilities to continue operation. One of the options is decontamination of the primary system to reduce radiation levels. The Battelle-Northwest study of decontamination and its impact on radwaste systems has been directed towards existing reactors and allied systems as they are employed during their operational lifetimes. Decommissioning and cleanup during such work are not within the scope of this project although certain processes and waste systems might be similar. Rupture debris cleanup represents a special situation that requires different design features and concepts and it is not a part of this study.

823 (PNL-SA--8143) DEMONSTRATION OF ALTERNATIVE DECONTAMINATION TECHNIQUES AT THREE MILE ISLAND. Arrowsmith, H.W.; Allen, R.P. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Nov 1979. Contract AC06-76RL01830. 41p. (CONF-7911104--1). NTIS, PC A03/MF A01.

From Facility decontamination technology workshop; Hershey, PA, USA (27 Nov 1979).

This paper discusses the following decontamination techniques: immersion electropolishing, in-situ electropolishing, barrel electropolishing, vibratory finishing, high-pressure freon spray, centrifugation, acid adsorption, and solidification. (DLC)

824 (PNL-SA--9051) TRAN-STAT STATISTICS FOR ENVIRONMENTAL STUDIES, NO. 13. Gilbert, R.O. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Nov 1980. Contract AC06-76RL01830. 23p. NTIS, PC A02/MF A01.

This issue deals with evaluating the spatial distribution and amounts of spatial phenomena. The emphasis here is on radionuclides, an application to such concerns as (a) evaluating risk to man of potential adverse health effects due to the dispersion of radionuclides in the environment (air, soil, water), (b) evaluating the need to conduct decommissioning and decontamination operations before releasing formerly used nuclear facilities for unrestricted public use, (c) estimating the amounts of contaminated soil for removal to achieve specified residual levels, and (d) estimating patterns of radon concentrations in the vicinity of mill tailings piles. However, the potential for application of Kriging techniques to DOE programs goes far beyond radionuclides. Some examples include (a) estimating spatial distribution of particulates emitted from the stacks of coal-fired power plants, (b) evaluating the geologic structure and chemical composition of potential radionuclide waste repositories, (c) estimating the extent to which chemical and toxic wastes have spread through the environment at disposal sites, (d) estimating uranium ore reserve deposits and, (e) prospection and exploitation of oil fields and coal reserves.

825 (PNL-SA--9211) IS ENTOMBMENT A SUITABLE ALTERNATIVE FOR DECOMMISSIONING A COMMERCIAL LWR. Holter, G.M. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jun 1981. Contract AC06-76RL01830. 21p. (CONF-8106144--9). NTIS (US Sales Only), PC A02/MF A01. Order Number DE82003845.

From Health Physics Society annual meeting; Louisville, KY, USA (21 Jun 1981).

This paper examines the suitability of entombment as one of the possible alternatives for decommissioning commercial LWRs. A clear definition of entombment is presented to provide a basis for examining the technical and regulatory constraints and restrictions. Entombment is compared with the other two decommissioning alternatives (immediate dismantlement and safe storage with deferred dismantlement) in terms of radiation dose considerations, waste disposal requirements, costs, and other consequences. Under current conditions, entombment is found to be less suitable than the other decommissioning alternatives, with several definite disadvantages. Furthermore, the principle advantage of entombment over immediate dismantlement, reduced occupational radiation dose, can be achieved more reasonably by safe storage with deferred dismantlement. Changing conditions (e.g., new regulations or lack of offsite disposal capacity) could impact the suitability of entombment, but the presence of significant quantities of long-lived radionuclides makes entombment of commercial LWRs unlikely except in very rare circumstances.

826 (PNL-TR--414) RADIOACTIVITY RELEASES FROM FUEL ELEMENTS STORED IN THE KRB-A STORAGE POOL OVER PROLONGED PERIODS OF TIME. Eickelpasch, N.; Wilstermann, H.; Fettel, W.; Huebscher, D. Translated from ATW, Atomwirtschaft, Atomtechnik ; 26: No. 4, 242-246(Apr 1981). Contract AC06-76RL01830. 17p. NTIS, PC A02/MF A01. Order Number DE81024395.

The whole core of the KRB-A nuclear power plant was unloaded in 1977 into the fuel element storage pool as a function of time and temperature was determined with an extensive monitoring program, in one case with a disconnected clean up plant and in the second case with a disconnected coolant in addition. Aerosol measurements were undertaken above the middle of the storage pool. The principal carriers of radioactivity in the pool water were the radionuclides Cs-134, Cs-137 and Co-60. The monitoring period is the longest yet to be devoted to this question. The measured values found give information on whether a longer storage of fuel elements, also with defective fuel rods, is possible without overtaxing the existing cleanup system.

827 USNG SOLUBLE NEUTRON ABSORBERS FOR CRITICALITY SAFETY. Lloyd, R.C.; Bierman, S.R.; Clayton, E.D.; Durst, B.M. (Battelle Northwest Labs., Richland, WA). Transactions of the American Nuclear Society ; 34: 325-327(1980). (CONF-800607--).

From American Nuclear Society annual meeting; Las Vegas, NV, USA (8 Jun 1980).

RADIOACTIVE WASTE STORAGE;SPENT FUEL STORAGE; CRITICALITY;NEUTRON ABSORBERS;SPECIFICATIONS

828 FUSION FUEL CYCLE SOLID RADIOACTIVE WASTES. Gore, B.F. (Battelle Pacific Northwest Labs., Richland, WA); Kabele, T.J.; Kaser, J.D. pp 193-213 of Management of low-level radioactive waste. Volume I. Carter, M.W.; Moghissi, A.A.; Kahn, B. (eds.). Elmsford, NY; Pergamon Press (1979).

From Symposium on management of low level radioactive waste; Atlanta, GA, USA (23 May 1977).

Eight conceptual deuterium--tritium fueled fusion power plant designs have been analyzed to identify waste sources, materials and quantities. All designs included the entire D--

T fuel cycle within each plant. Wastes identified include radiation-damaged structural, moderating, and fertile materials; getter materials for removing corrosion products and other impurities from coolants; absorbents for removing tritium from ventilation air; getter materials for tritium recovery from fertile materials; vacuum pump oil and mercury sludge; failed equipment; decontamination wastes; and laundry waste. Radioactivity in these materials results primarily from neutron activation and from tritium contamination. Activation product isotope production is subject to control by plant designers through materials selection and location. For the designs analyzed (which were prepared in part to identify needed research) annual radwaste volume was estimated to be 150 to 600 m^3/GWe. This may be compared to 500 to 1300 m^3/GWe estimated for the LMFBR fuel cycle. Major sources are replaced reactor structures and decontamination waste. Research to improve the economic attractiveness of fusion power by reducing reactor materials requirements and by increasing lifetimes of structural materials may reduce estimated volumes of replaced structures significantly. Estimates of waste volumes from decontamination are extremely uncertain. More complete design information is required to improve them.

829 METHODOLOGY FOR DETERMINING ACCEPTABLE RESIDUAL RADIOACTIVE CONTAMINATION LEVELS AT DECOMMISSIONED NUCLEAR FACILITIES/SITES. Watson, E.C.; Kennedy, W.E. Jr.; Hoenes, G.R.; Waite, D.A. (Battelle Pacific Northwest Labs., Richland, WA (USA)). pp 263-282 of Decommissioning of nuclear facilities. Vienna, Austria; IAEA (1979).

From International symposium on the decommissioning of nuclear facilities; Vienna, Austria (13 Nov 1978).

The ultimate disposition of decommissioned nuclear facilities and their surrrounding sites depends upon the degree and type of residual contamination. Examination of existing guidelines and regulations has led to the conclusion that there is a need for a general method to derive residual radioactive contamination levels that are acceptable for public use of any decommissioned nuclear facility or site. This paper describes a methodology for determining acceptable residual radioactive contamination levels based on the concept of limiting the annual dose to members of the public. It is not the purpose of this paper to recommend or even propose dose limits for the exposure of the public to residual radioactive contamination left at decommissioned nuclear facilities or sites.

830 PRACTICAL GUIDE FOR RADIOLOGICAL SURVEILLANCE OF THE ENVIRONMENT AT FEDERALLY-OWNED NUCLEAR SITES IN THE USA. Corley, J.P.; Waite, D.A. (Battelle Pacific Northwest Labs., Richland, Wash. (USA)); Elle, D.R. (Energy Research and Development Administration, Washington, D.C. (USA). Div. of Safety, Standards and Compliance). pp 953 of Proceedings of the 4. International congress of the International Radiation Protection Association. Vol. 3. Fontenay-aux-Roses, France; Association Internationale de Protection contre les Rayonnements (1977).

From 4. congress of the International Radiation Protection Agency; Paris, France (24 Apr 1977).

ENVIRONMENT;NUCLEAR POWER PLANTS;RADIATION MONITORING;REACTOR SITES;RADIOACTIVE EFFLUENTS; US ERDA

831 (PNL--3570) Understanding Radioactive Waste. Murray, R.L. (North Carolina State Univ., Raleigh, NC (USA)). Dec. 1981. Contract AC06-76RL01830. 124 p. NTIS Order Number DE82007628.

The material was written by an educator as a source of facts and concepts for use by interested citizens and as a supplementary science reference in high schools and junior colleges. Among the topics covered are basic atomic and nuclear physics, radioactivity, radiation and its biological effects, radiation protection standards, the origin of defense waste, uses of isotopes and radiation waste classification, sources of commercial radioactive waste, the nuclear fuel cycle, mill tailings, low-level waste, transportation, geologic disposal and other methods of isolation of high-level waste, and legal and societal aspects.

PERSONAL AUTHOR INDEX

Authors' surnames are indexed in the form appearing in the abstracted publication; given names are reduced to initials. Each author entry gives the publication title and the citation number. Report numbers, if applicable, are given in parentheses at the end of the entry. For publications with multiple authors, an author entry is provided for each. Entries for the second and succeeding authors give references to the first author.

Because of problems involved in the mechanical preparation of this index, all accent marks are omitted. Changes in spelling introduced as a result of accent omission follow standard conventions.

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The subject index is based on the use of subject descriptors selected from a controlled thesaurus of terms. Subject descriptors and qualifiers (subheadings) are selected and presented in the following format:

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The title may be supplemented with additional words, or a phrase, if it appears additional information would be helpful. In cases for which the title contains little or no information related to the subject entry, it may be replaced entirely by the supplementary information. A qualifier is not always required, and in such cases the title will follow the unqualified subject descriptor.

The descriptors selected for use as subject terms are generally the names of specific materials, things, or processes. To the extent possible, a qualifier is selected to describe the properties of, or processes applied to, the subject term.

Index entries are selected to indicate the important ideas and concepts presented in the report, rather than words that may appear in the text. Within the available thesaurus terms, the most probable or logical place to look for typical information is selected. "See references" are included to guide users from synonymous terms or phrases to the descriptor selected as a subject heading for the concept. (e.g. Pipeline Quality Gas see HIGH BTU GAS). "See also references" are used to indicate where to find references to subject concepts that are narrower, broader, or related to a particular subject heading. to complete an exhaustive search of a given subject, all such headings should be reviewed.

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46	514	See PNL-SA-8712
47	523	See PNL-SA-9054
CONF-801155-		(3. annual symposium on uranium mill tailings management, Ft Collins, CO, USA, 24-25 Nov 1980)
1	518	See PNL-SA-8861
2	731	See PNL-SA-9018
4	115	See PNL-SA-8829
5	517	See PNL-SA-8854
CONF-801206-		(ORNL conference on the leachability of radioactive solids, Gatlinburg, TN, USA, 9-12 Dec 1980)
10	287	See PNL-SA-8931
11	637	See PNL-SA-8935
CONF-801209-		(2. annual National Waste Terminal Storage (NWTS) information meeting, Columbus, OH, USA, 9-11 Dec 1980)
10	519	See PNL-SA-8927
CONF-801217-		(Symposium on modeling and low-level waste management, Denver, CO, USA, 1-4 Dec 1980)
4	524	See PNL-SA-9077
CONF-810217-		(ANS waste management conference, Tucson, AZ, USA, 23-26 Feb 1981)
8	116	See PNL-SA-8872
10	32	See PNL-SA-9247
16	117	See PNL-SA-9231
19	672	See PNL-SA-8881
CONF-810372-		(Symposium on uncertainties associated with the regulation of the geologic disposal of high-level radioactive waste, Gatlinburg, TN, USA, 9-15 Mar 1981)
1	525	See PNL-SA-9092
2	777	See PNL-SA-9096
11	530	See PNL-SA-9111
12	526	See PNL-SA-9094
16	529	See PNL-SA-9110
17	531	See PNL-SA-9114
CONF-810409-		(IAEA international symposium on methods of low-level counting and spectrometry, Berlin, F.R. Germany, 6-10 Apr 1981)
6	638	See PNL-SA-8992
CONF-810528-		(83. symposium on nucleation and crystallization in glasses, Washington, DC, USA, 3-6 May 1981)
4	528	See PNL-SA-9102
7	191	See HEDL-SA-2235
11	527	See PNL-SA-9101
CONF-810644-		(5. international workshop on rare earth-cobalt magnets and their application, Dayton, OH, USA, 6-7 Jun 1981)
1	713	See PNL-SA-9318
CONF-810650-		(International seminar on chemistry and process engineering for high-level waste solidification, Julich, F.R. Germany, 1-5 Jun 1981)
3	289	See PNL-SA-9440
CONF-810675-		(15. rare earth research conference, Rolla, MO, USA, 15-18 Jun 1981)
4	288	See PNL-SA-9377

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2247	375	Dep. NTIS, PC A03/MF A01
2253	579	Dep. NTIS, PC A10/MF A01
2265-4	192	Dep. NTIS, PC A03/MF A01
2278	376	Dep. NTIS, PC A03/MF A01
2286	589	Dep. NTIS, PC A28/MF A01
2286(App.)	804	Dep. NTIS, PC A11/MF A01
2287	590	Dep. NTIS, PC A09/MF A01
2378-3	78	Dep. NTIS, PC A05/MF A01
2383	737	Dep. NTIS, PC A03/MF A01
2400	79	Dep. NTIS, PC A06/MF A01
2419	689	See NUREG/CR-0523
2423	748	Dep. NTIS, PC A07/MF A01
2429	805	Dep. NTIS, PC A07/MF A01
2432	690	Dep. NTIS, PC A03/MF A01
2439	691	Dep. NTIS, PC A07/MF A01
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2551	751	See NUREG/CR-0576
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2614	693	Dep. NTIS, PC A04/MF A01
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2658	386	Dep. NTIS, PC A04/MF A01
2664	197	Dep. NTIS, PC A06/MF A01
2666	683	Dep. NTIS, PC A04/MF A01
2668-1	198	Dep. NTIS, PC A06/MF A01
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2670	694	Dep. NTIS, PC A03/MF A01
2672	82	Dep. NTIS, PC A03/MF A01
2677	695	Dep. NTIS, PC A04/MF A01
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2749	807	Dep. NTIS, PC A06/MF A01
2751	84	Dep. NTIS, PC A03/MF A01
2759	388	Dep. NTIS, PC A03/MF A01
2764	85	Dep. NTIS, PC A05/MF A01
2774	592	Dep. NTIS, PC A05/MF A01
2776	389	Dep. NTIS, PC A05/MF A01
2782	390	Dep. NTIS, PC A05/MF A01
2786	202	Dep. NTIS, PC A04/MF A01
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2801	696	Dep. NTIS, PC A03/MF A01
2808	392	Dep. NTIS, PC A04/MF A01
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2811	797	See NUREG/CR-0572
2821	808	Dep. NTIS, PC A11/MF A01
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2830	809	Dep. NTIS, PC A06/MF A01
2847	783	NTIS, PC A20/MF A01
2851	393	Dep. NTIS, PC A04/MF A01
2851	394	Dep. NTIS, PC A09/MF A01
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2851	780	Dep. NTIS, PC A09/MF A01
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2851	786	Dep. NTIS, PC A09/MF A01
2854	396	Dep. NTIS, PC A04/MF A01
2858	397	NTIS, PC A03/MF A01
2859	398	NTIS, PC A03/MF A01
2862	399	NTIS, PC A06/MF A01
2863	400	Dep. NTIS, PC A04/MF A01
2864	784	NTIS, PC A03/MF A01
2874	401	NTIS, PC A15/MF A01
2875	402	Dep. NTIS, PC A03/MF A01
2879	403	Dep. NTIS, PC A03/MF A01
2882	782	NTIS, PC A03/MF A01
2892	659	NTIS, PC A09/MF A01
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2941	87	Dep. NTIS, PC A05/MF A01
2949	774	Dep. NTIS, PC A08/MF A01
2954	409	Dep. NTIS, PC A03/MF A01
2970	595	Dep. NTIS, PC A15/MF A01
2985	4	Dep. NTIS, PC A04/MF A01
2990	410	NTIS, PC A06/MF A01
3027	206	NTIS, PC A03/MF A01
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3038	208	Dep. NTIS, PC A05/MF A01
3050-1	209	Dep. NTIS, PC A03/MF A01
3060	210	Dep. NTIS, PC A06/MF A01
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3070	411	Dep. NTIS, PC A04/MF A01
3078	89	Dep. NTIS, PC A05/MF A01
3090	211	NTIS, PC A04/MF A01
3094	212	Dep. NTIS, PC A03/MF A01
3109	213	Dep. NTIS, PC A06/MF A01
3127	766	NTIS, PC A08/MF A01
3128	697	NTIS, PC A03/MF A01
3129	596	Dep. NTIS, PC A05/MF A01
3137	214	Dep. NTIS, PC A03/MF A01
3139	412	Dep. NTIS, PC A04/MF A01
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3146	413	NTIS, PC A04/MF A01
3152	90	Dep. NTIS, PC A06/MF A01
3153	5	Dep. NTIS, PC A09/MF A01
3159	811	NTIS, PC A03/MF A01
3160-1	755	Dep. NTIS, PC A03/MF A01
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3170	216	NTIS, PC A03/MF A01
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3180	660	NTIS, PC A10/MF A01
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3198	775	Dep. NTIS, PC A06/MF A01
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3227	425	Dep. NTIS, PC A05/MF A01
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3300(Pt.2)	93	Dep. NTIS, PC A11/MF A01
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3300(Pt.5)	598	Dep. NTIS, PC A07/MF A01
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3302	599	NTIS, PC A07/MF A01
3304	600	Dep. NTIS, PC A04/MF A01
3315	740	NTIS, PC A04/MF A01
3317-1	94	Dep. NTIS, PC A04/MF A01
3317-2	41	NTIS, PC A04/MF A01
3318	426	Dep. NTIS, PC A06/MF A01
3326	427	NTIS, PC A03/MF A01
3333	428	Dep. NTIS, PC A06/MF A01
3336	221	NTIS, PC A04/MF A01
3340	661	NTIS, PC A03/MF A01
3343	222	NTIS, PC A03/MF A01
3346	756	NTIS, PC A06/MF A01
3349	429	NTIS, PC A04/MF A01
3356	430	NTIS, PC A12/MF A01. Order Number DE81027620, Distribution Category MN -70
3363	601	NTIS, PC A10/MF A01
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3432	433	NTIS, PC A04/MF A01
3447	434	NTIS, PC A05/MF A01
3465	228	NTIS, PC A05/MF A01
3473	603	NTIS, PC A04/MF A01
3475	741	NTIS, PC A08/MF A01
3477	229	NTIS, PC A10/MF A01
3479	230	NTIS, PC A03/MF A01
3484	435	NTIS, PC A06/MF A01
3493	231	NTIS, PC A06/MF A01
3495	232	NTIS, PC A99/MF A01
3496	436	NTIS, PC A03/MF A01
3498	727	NTIS, PC A03/MF A01
3505	233	NTIS, PC A13/MF A01
3509(Pt.1)	742	NTIS, PC A03/MF A01
3511	815	NTIS, PC A06/MF A01. Order Number DE81028260, Distribution Category MN -11
3512	234	NTIS, PC A04/MF A01. Order Number DE82000708, Distribution Category MN -70
3514	662	NTIS, PC A05/MF A01
3516	235	NTIS, PC A04/MF A01
3518	604	NTIS, PC A03/MF A01
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3524	704	NTIS, PC A08/MF A01
3528	605	NTIS, PC A05/MF A01
3542	437	NTIS, PC A99/MF A01. Order Number DE81027687, Distribution Category MN -70
3548	728	NTIS, PC A07/MF A01
3550	236	NTIS, PC A06/MF A01. Order Number DE81030408, Distribution Category MN -70
3552	237	NTIS, PC A04/MF A01
3554	238	NTIS, PC A03/MF A01
3556	438	NTIS, PC A03/MF A01. Order Number DE81026322, Distribution Category MN -70
3559	439	NTIS, PC A05/MF A01
3562	816	NTIS, PC A03/MF A01
3564	99	NTIS, PC A02/MF A01
3566	26	NTIS, PC A06/MF A01
3569	440	NTIS, PC A04/MF A01. Order Number DE81026840, Distribution Category MN -70
3570	831	NTIS
3574	606	NTIS, PC A04/MF A01. Order Number DE81028969, Distribution Category MN -70
3584	817	NTIS, PC A03/MF A01. Order Number DE81028363, Distribution Category MN -11
3588	100	NTIS, PC A05/MF A01
3601	663	NTIS, PC A04/MF A01
3608-1	239	NTIS, PC A14/MF A01. Order Number DE82000936, Distribution Category MN -70
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3614	607	NTIS, PC A04/MF A01. Order Number DE81026562, Distribution Category MN -70
3616	729	NTIS, PC A03/MF A01. Order Number DE81029397, Distribution Category MN -70
3617	240	NTIS, PC A02/MF A01
3639	705	NTIS, PC A03/MF A01
3647	730	NTIS, PC A02/MF A01
3678	441	NTIS, PC A03/MF A01. Order Number DE81024016, Distribution Category MN -70
3679	368	See DOE/UMT-0200
3680	608	NTIS, PC A03/MF A01
3683	664	NTIS, PC A04/MF A01
3698	442	NTIS, PC A02/MF A01. Order Number DE82010839, Distribution Category MN -70
3700(Pt.2)	101	NTIS, PC A09/MF A01
3700(Pt.2)	102	NTIS, PC A09/MF A01
3700(Pt.2)	443	NTIS, PC A09/MF A01
3700(Pt.2)	444	NTIS, PC A09/MF A01
3700(Pt.2)	445	NTIS, PC A09/MF A01
3700(Pt.2)	609	NTIS, PC A09/MF A01
3700(Pt.2)	721	NTIS, PC A09/MF A01
3700(Pt.5)	610	NTIS, PC A05/MF A01
3700(Pt.5)	818	NTIS, PC A05/MF A01
3700(Pt.5)	819	NTIS, PC A05/MF A01
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3702	241	NTIS, PC A04/MF A01
3720	446	NTIS, PC A04/MF A01
3725	821	NTIS, PC A05/MF A01. Order Number DE81029613, Distribution Category MN -20e
3728	706	NTIS, PC A05/MF A01
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3729	743	NTIS, PC A04/MF A01. Order Number DE81028358, Distribution Category MN -41
3741	242	NTIS, PC A02/MF A01. Order Number DE81028405, Distribution Category MN -70
3742	243	NTIS, PC A11/MF A01. Order Number DE81024477, Distribution Category MN -70
3750	244	NTIS, PC A04/MF A01
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3774	103	NTIS, PC A08/MF A01. Order Number DE81028208, Distribution Category MN -70
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3791	665	NTIS, PC A10/MF A01. Order Number DE81029275, Distribution Category MN -70
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3797	104	NTIS, PC A03/MF A01. Order Number DE81027835, Distribution Category MN -70
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3802	246	NTIS, PC A10/MF A01
3903	105	NTIS, PC A03/MF A01
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3817	612	NTIS, PC A04/MF A01
3818	247	NTIS, PC A05/MF A01. Order Number DE81029616, Distribution Category MN -70
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3873	451	NTIS, PC A06/MF A01. Order Number DE82000937, Distribution Category MN -70
3877	613	NTIS, PC A06/MF A01. Order Number DE82000239, Distribution Category MN -70
3881	106	NTIS, PC A05/MF A01. Order Number DE81026978, Distribution Category MN -70
3898	452	NTIS, PC A03/MF A01. Order Number DE91030231, Distribution Category MN -70
3902	250	NTIS, PC A03/MF A01. Order Number DE81028403, Distribution Category MN -70
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3915	454	NTIS, PC A04/MF A01. Order Number DE82003233, Distribution Category MN -70
3927	614	NTIS, PC A03/MF A01. Order Number DE81030409, Distribution Category MN -70
3943	251	NTIS, PC A03/MF A01. Order Number DE82001868, Distribution Category MN -70
3948	252	NTIS, PC A04/MF A01. Order Number DE81030621, Distribution Category MN -70
3950	667	NTIS, PC A04/MF A01. Order Number DE82008473, Distribution Category MN -41
3954	455	NTIS, PC A05/MF A01. Order Number DE82000697, Distribution Category MN -70
3957	253	NTIS, PC A03/MF A01. Order Number DE82001050, Distribution Category MN -70
3959	254	NTIS, PC A09/MF A01. Order Number DE82000957, Distribution Category MN -70
3964	255	NTIS, PC A04/MF A01. Order Number DE82004325, Distribution Category MN -70
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4015	107	NTIS, PC A04/MF A01. Order Number DE82002649, Distribution Category MN -70
4022	456	NTIS, PC A04/MF A01. Order Number DE82005595, Distribution Category MN -70
4045	257	NTIS, PC A04/MF A01. Order Number DE82007872, Distribution Category MN -70
4050	108	NTIS, PC A04/MF A01. Order Number DE82002525, Distribution Category MN -70
4052	258	NTIS, PC A02/MF A01. Order Number DE82003954, Distribution Category MN -70
4072	668	NTIS, PC A06/MF A01. Order Number DE82005594, Distribution Category MN -70
4098	259	NTIS, PC A06/MF A01. Order Number DE82008774, Distribution Category MN -70
4099	669	NTIS, PC A04/MF A01. Order Number DE82008470, Distribution Category MN -41
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4100-Pt.2	460	NTIS, PC A08/MF A01
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4120	708	NTIS, PC A03/MF A01. Order Number DE82005722, Distribution Category MN -41
4178	618	NTIS, PC A03/MF A01. Order Number DE82008986, Distribution Category MN -70
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6416	28	NTIS, PC A02/MF A01
6526	29	Dep. NTIS, PC A02/MF A01
6698	461	Dep. NTIS, PC A03/MF A01
6707	109	Dep. NTIS, MF A01
6719	260	Dep. NTIS, PC A03/MF A01
6729	110	Dep. NTIS, PC A02/MF A01
6768	261	Dep. NTIS, PC A02/MF A01
6771	709	Dep. NTIS, PC A02/MF A01
6832	710	Dep. NTIS, PC A02/MF A01
6839	111	Dep. NTIS, PC A02/MF A01
6955	262	Dep. NTIS, PC A02/MF A01
6957	462	Dep. NTIS, PC A99/MF A01
7072	112	Dep. NTIS, PC A03/MF A01
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7142	621	NTIS, PC A03/MF A01. Order Number DE82007090, Distribution Category MN -70
7184	466	Dep. NTIS, PC A02/MF A01
7197	467	Dep. NTIS, PC A03/MF A01
7243	468	Dep. NTIS, PC A02/MF A01
7245	622	Dep. NTIS, PC A02/MF A01
7292	263	Dep. NTIS, PC A02/MF A01
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