DNFSB 94-2 Implementation Plan
Section IX, Research and Development
Task 1: DOE R&D Activities Assessment

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SUMMARY

The purpose of this report is to identify existing research and development (R&D) activities for the Department of Energy (DOE) that relate to the management and disposal of LLW, and technology development. For the purpose of this report, technology development is defined as the development of a tool; this does not include the application of existing tools, refinement of existing models, performing more detailed analyses, or collecting data. This report explains the approach used to assess R&D activities, describes the sources evaluated, and lists the results of those evaluations.

The approach to DOE R&D Activities Assessment was to identify sources of R&D activities, evaluate each activity contained within the sources against two criteria, and prepare a letter report containing the results of the evaluation.

The sources of R&D activities used in this assessment included the (1) R&D Activities Catalog, (2) DOE R&D Tracking Database, and (3) Treatability Study Database.

A total of 23,869 records were evaluated against two criteria: management and disposal of LLW, and technology development. Of the records evaluated, 166 records received a Yes determination (met both criteria), 658 received an Unknown determination (one or more criterion were not met), and 23,045 received a No determination (insufficient information to determine if either criterion were met), and are not included in this report.

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INTRODUCTION

On September 8, 1994, the Defense Nuclear Facilities Safety Board (DNFSB or the Board) issued Recommendation 94-2, "Conformance with Safety Standards at Department of Energy Low-Level Nuclear Waste and Disposal Sites." Part of Recommendation 94-2 focused on the need for the Department of Energy (DOE) to address certain research and development (R&D) needs and background studies that are critical to the effective management of low-level radioactive waste (LLW) within DOE.

On October 28, 1994, DOE accepted Recommendation 94-2 and in response created the 94-2 Implementation Plan. DOE recognized that although its Environmental Management Program has continuing research programs, there is currently no coordinated program specific to LLW that can (1) identify, implement, and guide LLW R&D, and (2) ensure that R&D needs are met. Consequently, as part of the Implementation Plan, DOE committed to establish an R&D Task Team (RDTT) to identify where LLW management R&D needs exist, and to define a strategy for addressing these needs. Because the RDTT no longer exists, the responsibility for the remaining deliverables in Section IX has been delegated to the Center of Excellence for Low-Level and Mixed Low-Level Waste under the direction of DOE-ID.

The R&D section of the Implementation Plan identifies five areas of concern for improving the management of LLW that can be addressed through research and development: (1) improving modeling and predictive capabilities of radionuclide migration, (2) enhancing the stability of buried waste forms, (3) enhancing the deterrence of intrusion, (4) inhibiting the migration of radionuclides, and (5) reducing the volume of waste to be disposed. The following task initiatives were proposed in the Implementation Plan that would identify strategies to address areas of concern:

Task 1. Catalog past, current, and planned LLW management program R&D initiatives
Task 2. Coordinate the identification of LLW management program R&D needs
Task 3. Determine outstanding LLW R&D needs
Task 4. Develop and recommend a strategy for LLW R&D needs.

To implement Task 1, Pacific Northwest National Laboratory (PNNL) compiled a database of LLW management R&D and technology development activities. This was completed on June 27, 1995. Since that time, other resources have become available, and the responsibility for the R&D initiative has shifted from the RDTT to the INEEL. The project is now focused on assessing sources to identify those R&D activities that apply to both management and disposal of LLW, and technology development.

PURPOSE

The purpose of this report is to identify existing R&D activities that relate to the management and disposal of LLW, and technology development. The work plan accepted by DOE states that the purpose of Task 1 is to identify work that is being done or may have been completed that solves a technical need relative to the management and disposal of LLW. As used in this report, technical need and technology development are equal terms defined as the development of a tool. Technology development does not include the application of existing tools, refinement of existing models, performing more detailed analyses, or collecting data. This report explains the approach used to assess selected R&D activities, and describes the sources evaluated and the results of those evaluations.

Page IX-4 of the April 1996 Implementation Plan states ". . .2) the context of any further cataloging will be defined by the needs statement definition under task IX.B.2, Identification of LLW Management R&D Needs. The collection of additional information about existing R&D activities/work will occur as an ongoing exercise under tasks IX.B.2 and IX.B.3." This report, prepared as an extension of the original work done under Task IX.B.1 of the March 1995 Implementation Plan, was completed on June 27, 1995.

APPROACH

The approach to DOE R&D Activities Assessment was to identify sources of R&D activities, evaluate each activity contained within the sources against two criteria, and prepare a letter report containing the results of the evaluation.

The sources of R&D activities used in this assessment included the (1) R&D Activities Catalog, (2) DOE R&D Tracking Database, and (3) Treatability Study Database. TechCon and the Vendor Information System for Innovative Treatment Technologies (VISITT) 3.0 were included in the previous R&D Activities Catalog so they were not reevaluated in this assessment. These sources were chosen because they were available and appeared to be current and comprehensive. The resources available to complete this task did not allow a more comprehensive search for other R&D activities. From these three sources of R&D activities, a total of 23,869 records were evaluated against the following criteria:

- Does the R&D activity relate to the management or disposal of LLW?
- Does the R&D activity represent technology development? Note that technology development is defined for this report as the development of a tool. It does not include the application of existing tools, refinement of existing models, performing more detailed analyses, or collecting data.

From this evaluation, R&D activities were given one of three determinations: Yes, Unknown, or No. If both evaluation criteria were met, the activity received a Yes determination. If either evaluation criterion was not met, it received a No determination. If the information provided did not contain sufficient detail to make a determination, it received an Unknown determination.

SOURCE MATERIAL

R&D Activities Catalog

DNFSB Recommendation 94-2 Implementation Plan Preliminary LLW Management R&D Activities Catalog, Version 1.0, June 27, 1995, was completed and provided to the DNFSB as the deliverable corresponding to Research and Development Task IX.B.1 of the current *DNFSB 94-2 Implementation Plan* (Rev 1, April 1996). The Catalog was developed in two parts: a hard-copy description of the catalog and how it was developed, and a database of DOE and non-DOE LLW management R&D activities compiled through a database and hard copy literature search. Table 1 of the hard-copy description identifies the sources used to capture the R&D activities contained in the database. The Catalog was prepared by Pacific Northwest National Laboratory under contract to Sandia National Laboratories.

Although the Catalog was completed and delivered to the DNFSB, it was also reviewed as part of this task. Reviewers found that most of the activities listed in the Catalog did not constitute LLW technology development. A total of 5,262 records were evaluated; 13 received a Yes determination (see Table A-1 in Appendix A), 222 received an Unknown determination (see Table A-2 in Appendix A).

DOE R&D Tracking Database

The DOE R&D Tracking Database was developed to (1) respond to the annual Office of Science and Technology Policy (OSTP)/Critical Technologies Institute data call, (2) make it simpler for the Department to track how it is spending taxpayer dollars, and (3) reduce the time currently spent in responding to ad hoc data calls.

This database is managed by the Office of the Chief Financial Officer to ensure conformity with budget information. The Office of Scientific and Technical Information (OSTI), Oak Ridge, Tennessee, provides technical support and maintains the central database. The Deputy Under Secretary for R&D Management works with the R&D Council to help ensure implementation of this database. Each of the Program Secretarial Offices has designated a person responsible for implementation.

The Department-wide DOE R&D Tracking Database Team defined the core data to be included in the database for 1996, as well as optional data fields that may be included in future years. The core data fields satisfy the annual OSTP data call as well as a number of internal data needs. The database information submitted by sites includes a project identification (ID) that consists of a site identifier and number.

As of January/February 1997, the database contained 18,534 records. This number can fluctuate due to ongoing quality assurance activities. The database was initially populated with FY 1995 information and has been updated with FY 1996 information. The project ID, project title, project description, and point of contact name and phone number were extracted for each record.

A total of 18,534 records were evaluated; 148 received a Yes determination (see Table B-1 in Appendix B), 418 received an Unknown determination (see Table B-2 in Appendix B).

An attempt was made to validate a small portion of the Yes determinations by discussing the activity with the point of contact. The follow-up was intended to verify that the title and description were correctly interpreted and that technology development was a product. In all cases, the point of contact had limited information on the details of the project in question and was unable to provide additional contact information.

Treatability Study Database

The Treatability Study Database was developed by the Mixed Waste Focus Area (MWFA) as a tool to track the number, technical scope, and research goals of treatability studies targeting mixed waste treatment at DOE facilities. The information is intended as a reference for establishing and maintaining continuity with MWFA research projects.

The technology portion of the database includes fields for the technology name, type, and description. The treatability study portion of the database includes fields for study, name, start and end dates, scale, status, goals, results, and conclusions. The administrative portion of the database includes contact information, study location, and comments. The waste stream portion included fields for waste stream name, description, DOE radiological waste type, radiological handling, alpha contamination, physical/chemical waste matrix, PCB contamination, and weight.

The information in the database was collected directly from DOE sites conducting the studies. The research principal investigator or site treatability study coordinators provided the information.

The information related to LLW in this database is limited because the focus of the data collection was mixed waste. Future activity in this area should result in a larger inventory of mixed waste treatability studies as well as information on LLW treatability studies.

A total of 73 records were evaluated; 5 received a Yes determination (see Tables C-1 in Appendix C), 18 received an Unknown determination (see Table C-2 in Appendix C).

CONCLUSIONS

A total of 23,869 records were evaluated in this assessment; 166 received a Yes determination, and 658 received an Unknown determination. The remaining 23,045 received a No determination and are not included in this report. In closing, the reader should note that the results of this DOE R&D Activities Assessment will be combined with the results of the DOE R&D Needs Assessment to determine if there are technology development needs related to the management or disposal of LLW that are not being met by current activities within the Complex.

This study was limited by a lack of a complete data set, poor data quality, and the reliance on the use of best engineering judgment to determine the inclusion of data entries in a set that represents R&D activities. These limitations are detailed as follows:

- DOE recognizes that although it has continuing R&D projects, there is no coordinated program specific to LLW to identify, implement, and guide LLW research and development, and ensure that the R&D needs are met. It is likely that the databases used in this study do not include many activities currently being performed in the Complex. However, the large number of entries suggests that a majority of R&D activities are probably captured in these databases. It did not appear prudent to expend resources to obtain information on projects not included.

- In most cases, it was difficult to assess whether the activity listed in a database represented an R&D effort related to technology because the information in the data fields was often vague and appeared to be worded as a funding request rather than a clear definition of work.

- The large number of data entries (over 23,000 activities are included in the R&D Catalog, DOE R&D Tracking Database, and the Treatability Study Database), limited time to perform the analysis, and limited human resources did not allow for discussion with project managers to make these determinations. Best engineering judgment and experience were used to categorize activities as R&D efforts related to technology and the development of new tools to manage LLW. It is possible that individuals with more experience concerning specific projects or a different point of view as to what constitutes an R&D activity would arrive at different conclusions.

REFERENCES

Department of Energy, April 1996, *Implementation Plan, Defense Nuclear Facilities Safety Board Recommendation 94-2, Conformance with Safety Standards at Department of Energy Low-Level Waste and Disposal Sites,* Revision 1.

June 27, 1995, *Defense Nuclear Facilities Safety Board (DNFSB) Recommendation 94-2, Preliminary LLW Management R&D Activities Catalog,* Version 1.0.

Pacific Northwest National Laboratory (PNNL) Database, Version 1.0, Available from the Idaho National Engineering and Environmental Laboratory, Waste Operations Department, R.L. Devries at: rdx@inel.gov.

Department of Energy, *DOE R&D Tracking Database*, available via the Internet at: http://www.doe.gov/rd/. Contact Mike Frame at: Mike.Frame@ccmail.osti.gov.

U.S. EPA's Technology Innovation Office, *Vendor Information System for Innovative Treatment Technologies (VISITT) 3.0*, (latest version 5.0 is available on the Internet at: http://www.prcemi.com:80/visitt/).

APPENDIX A

R&D Activities Catalog Analysis

Table A-1. R&D Activities with Yes Determination[1]

ID No.	Title	Abstract
1473	Removing method for carbon isotope in condensed liquid waste	Carbon isotopes in condensed liquid wastes formed by reducing the volume of radioactive liquid wastes discharged from a nuclear power plant are efficiently removed in a relatively simply device.^That is, the bottom liquid of a condenser is vaporized by heating at about 100degC to conduct condensation.^With such a constitution, several % of carbon 14 in the bottom liquid is continuously transferred to the condensed water.^Accordingly, when purified water not containing carbon 14 is supplied to the bottom liquid and concentration is applied, the concentration of carbon 14 in the bottom liquid can be reduced remarkably.^Further, the removing efficiency of the carbon isotopes can be further improved by blowing air into the condensed liquid wastes, at the same time with supply of purified water for evaporating condensation.^(I.S.).
1490	Clean-Cut Removal System for concrete decontamination	This report addresses the Clean-Cut Removal System, CCRS, which can accurately scrape the contaminated concrete surface by the rotation of special cutters and simultaneously collect the scraped debris in a drum.^The techniques up to this time for removing contaminated concrete surfaces have some problems, such as irregular depth of removal, difficulty of collecting the debris, and so on.^As a solution to these problems, this new method with sophisticated cutters and capable vacuum collecting systems has been developed.^The cutters used in this system can scrape the concrete surface to a few millimeters' depth accurately in one pass.^The debris is a uniform powder and is collected almost 100% by a vacuum collecting system.^This method has many advantages, such as radioactive waste reduction, prevention of internal exposure of workers, recontamination prevention, and easy measurement of residual radioactivity after decontamination.^In this report, it is shown that this system is very useful and effective for removing contaminated concrete in the nuclear facilities.^(author).
1499	Measurement of trace uranium-235 and plutonium-239, 240 in waste tank material at the Savannah River Site	At the Savannah River Site (SRS), large quantities of radioactive liquid waste are evaporated to reduce volume before eventual processing through the In-Tank Precipitation process.^Actinides in the liquid waste are only slightly soluble in the highly alkaline waste solution.^Since some of the actinide isotopes are fissionable, the quantities being processed through the evaporator system are of interest.^To better quantify the concentration and mass of fissionable material entering the evaporator system and eventually deposited as salt, analysis of the actinide elements were necessary.^The predominant fissionable actinide isotopes of interest are U{sup 235} and Pu{sup 239}.^To enable the reliable measurement of these radionuclides, the Central Laboratory has developed high speed separation techniques to measure U{sup 235} content by Isotope Dilution Mass Spectrometry and Pu{sup 239,240} by alpha spectrometry.^Due to the high radioactivity levels in the samples all separations are performed in shielded analytical cells.^Uranium is purified and concentrated using a high speed extraction chromatography technique that employs applied vacuum and columns containing tri (2-ethylene) phosphate solvent coated on a small particle inert support.^The uranium method enables measurement of U{sup 235} concentrations to 1 {times} 10{sup {minus}4} g/L.^Plutonium is purified and concentrated using a high speed anion exchange technique.^The Pu method enables measurements of Pu{sup 239,240} to 2 {times} 10{sup {minus}6} g/L.

[1]This information was extracted from the PNNL database. No effort was made to correct the information or improve its appearance.

Table A-1. R&D Activities with Yes Determination (continued)

ID No.	Title	Abstract
1568	Cold-crucible melting of hulls and structural materials	The method currently implemented at the La Hague UP3 reprocessing plant for conditioning of PWR zircaloy hulls is cement embedding.^Another promising method, mainly for reducing the waste volume and the available exchange surface area, is melting.^A cold-crucible melting process has therefore been developed by the CEA at Marcoule (France) over the last decade.^Development work first concentrated on cladding hulls from fast breeder reactors, then from pressurized water reactors.^The process can be used for both types of cladding wastes.^Subassembly head and foot end-caps are sheared off and should be suitable for surface storage after {alpha} decontamination by successive rinsing.^If necessary because of their {alpha} activity, they could be melted in a larger furnace.
1608	Advanced methods for the removal of actinides from low and intermediate level waste effluents	Recent effluent treatment studies at Harwell Laboratory have led to the development of a new type of process, seeded ultrafiltration.^This offers a very versatile method for the decontamination of waste effluents, with significantly reduced volumes of secondary waste arising compared to conventional precipitation and ion exchange processes.^Experimental results are presented which show the effectiveness of seeded ultrafiltration for the removal of plutonium and americium to very low levels (down to 2.1 {times} 10{sup {minus}5} Bq/m{ell}Pu and 4.1 {times} 10{sup {minus}5} Bq/m{ell}Am) from a simulated low level effluent.^Further studies, using laundry effluent from a nuclear site, indicate that significant decontamination could still be achieved in the presence of complexants known to interfere in many conventional treatment processes.
1660	Recovery of tritium from water	The pilot-scale Combined Electrolysis Catalytic Exchange (CECE) system developed at the U.S.^Department of Energy's Mound facility has evolved into a fully operational tritium recovery system.^This has resulted from the evaluation of recent developments in AECL/CRNL hydrophobic exchange catalyst in the CECE system.^Data obtained during recent tests led to the design and installation of an aqueous tritium recovery facility.^Operation of the Tritium Aqueous Waste Recovery System makes possible the recovery of tritium from low-level tritiated aqueous waste streams.
1750	Recovery of tritium from water	The pilot-scale Combined Electrolysis Catalytic Exchange (CECE) system developed at the US Department of Energy's Mound facility has evolved into a fully operational tritium recovery system.^This has resulted from the evaluation of recent developments in AECL/ORNL hydrophobic exchange catalyst in the CECE system.^Data obtained during recent tests led to the design and installation of an aqueous tritium recovery facility.^Operation of the Tritium Aqueous Waste Recovery System makes possible the recovery of tritium from low-level tritiated aqueous waste streams.^14 refs., 4 figs.

Table A-1. R&D Activities with Yes Determination (continued)

ID No.	Title	Abstract
1857	Method of disposing radioactive waste water	Purpose: To efficiently remove radioactive materials to such an extent as causing no trouble to the subsequent use or disposal, thereby greatly reduce the volume of radioactive wastes upon disposal of radioactive waste water.^Method: Waste water containing radioactive materials is filtered by using easily combustible ion exchange paper filters prepared by mixing synthetic fibers made of polyethylene or the like with ion exchange fibers to which sulfonic or like other functional groups are chemically bonded, cellulose and a small amount of synthetic fibers to make paper and then applying hot pressing.^Then, ion exchange paper filters are burnt out together with the radioactive materials captured by filtration.^By making paper while adding 30 - 50 parts of cellulose to 50 parts of fibers, the burnability can greatly be improved, the combustion is rendered extremely moderate and complete incineration can be attained with no generation of soots and sagging.^Accordingly, since the volume of ashes is reduced, significant volume-reduction of radioactive wastes can be attained.^(Takahashi, M.).
2244	Improvements in inductively coupled plasma - mass spectrometry for elemental and isotopic analysis	A new type of ICP-MS device that contains two separate quadrupole mass analyzers.^The objective is to improve the precision and sensitivity with which either stable or radioactive elements may be determined in any type of sample.^Representative applications pertinent to waste remediation will be pursued.
2255	SEAMIST soil sampling for tritiated water: First year's results	SEAMIST is a recently developed sampling system that enables one to measure various soil parameters by means of an inverted, removable, impermeable membrane tube inserted in a borehole.^This membrane tube can have various measuring devices installed on it, such as gas ports, adsorbent pads, and electrical sensors.^These membrane tubes are made of a laminated polymer.^The Lawrence Livermore National Laboratory in Livermore, California, has installed two of these systems to monitor tritium in soil resulting from a leak in an underground storage tank.^One tube is equipped with gas ports to sample soil vapor and the other with adsorbent pads to sample soil moisture.^Borehole stability was maintained using either sand-filled or air-inflated tubes.^Both system implementations yielded concentrations or activities that compared well with the measured concentrations of tritium in the soil taken dozing borehole construction.^In addition, an analysis of the data suggests that both systems prevented the vertical migration of tritium in the boreholes.^Also, a neutron probe was successfully used in a blank membrane inserted in one of the boreholes to monitor the moisture in the soil without exposing the probe to the tritium.^The neutron log showed excellent agreement with the soil moisture content measured in soil samples taken during borehole construction.^This paper describes the two SEAMIST systems used and presents sampling results and comparisons.

Table A-1. R&D Activities with Yes Determination (continued)

ID No.	Title	Abstract
2506	States of the art: advanced technology instrumentation from Amdel	This article focuses on recent innovative equipment introduced by Amdel Ltd., South Australia.^Products covered are: the High Performance Density Gauge; the Thickener Interface Gauge; and the Dry Stream Analyzer.^The density gauge for on-pipe measurement of liquids, slurries and bulk solids is faster, more accurate and stable.^Key features are: ease of calibration; a self diagnostic utility and fast response.^The Thickener Interface Gauge (TIG) measures mud bed level in thickener or washer tanks.^Via a probe inserted in the tank, the TIG measures the natural radioactivity using a vertical assay of up to 12 detectors.^The relatively high level of the natural gamma radiation of the solids and the low gamma emission of the liquid (or vice versa) enable accurate determination of the solids/liquid interface.^The TIG allows greater control and efficiency of thickener or washer operation.^The Dry Stream Analyser measures the elemental concentrations in dry mineral sand streams.^Mounted in a special sample presentation system, the solid state probe provides simultaneous measurement of up to 8 elements.^Applications include: feed from dredge; and rutile, zircon and ilmenite product.
3438	Ba(OH)/sub 2/.8H/sub2/O process for the removal and immobilization of carbon-14. Final report	The airborne release of /sup 14/C from various nuclear facilities has been identified as a potential biohazard due to the long half-life of /sup 14/C (5730 years) and the ease with which it may be assimilated into the biosphere.^At ORNL, technology has been developed for the removal and immobilization of this radionuclide.^Prior studies have indicated that /sup 14/C will likely exist in the oxidized form as CO/sub 2/ and will contribute slightly to the bulk CO/sub 2/ concentration of the gas stream, which is air-like in nature (approx.300 ppM/sub v/ CO/sub 2/).^The technology that has been developed utilizes the CO/sub 2/-Ba(OH)/sub 2/.8H/sub 2/O gas-solid reaction with the mode of gas-solid contacting being a fixed bed.^The product, BaCO/sub 3/, possesses excellent thermal and chemical stability, prerequisites for the long-term disposal of nuclear wastes.^For optimal process operation, studies have indicated that an operating window of adequate size does exist.^When operating within the window, high CO/sub 2/ removal efficiency (effluent concentrations <100 ppB/sub v/), high reactant utilization (>99%), and an acceptable pressure drop across the bed (3 kPa/m at a superficial velocity of 13 cm/s) are possible.^Three areas of experimental investigation are reported: (1) microscale studies on 150-mg samples to provide information concerning surface properties, kinetics, and equilibrium vapor pressures; (2) macroscale studies on large fixed beds (4.2 kg of reactant) to determine the effects of humidity, temperature, and gas flow rate upon bed pressure drop and CO/sub 2/ breakthrough; and (3) design, construction, and operation of a pilot unit capable of continuously processing a 34-m/sup 3//h (20-ft/sup 3//min) air-based gas stream.
4264	Determination of Th-230 in water	On of the current goals of the Radiochemistry Division of CDTN is to acquire capability in determining contaminants of high radiotoxicity in different matrixes.^The method described here was developed in order to determine Th-230 in water, one of the decay products of the uranium series, belongs to the most restrictive class of radionuclides, on account of its alpha emission and half-life of 8 x 10{sup 4} years.^The method consists of two radiochemical steps, one electrodeposition step and alpha spectrometry with the use of a surface-barrier detector.^Water samples with internal Th-230 standards were analysed and the results showed good reproducibility with errors around 10% and alpha detection efficiency of 12%.^The lower detection limit is around 0,4 Bq/l.^(author).

Table A-2. R&D Activities with Unknown Determination[1]

ID No.	Title	Objective/Abstract
367	APOLLO: A computer program for the calculation of chemical equilibrium and reaction kinetics of chemical systems	Several of the technologies being evaluated for the treatment of waste material involve chemical reactions.^Our example is the in situ vitrification (ISV) process where electrical energy is used to melt soil and waste into a glass like" material that immobilizes and encapsulates any residual waste.^During the ISV process, various chemical reactions may occur that produce significant amounts of products which must be contained and treated.^The APOLLO program was developed to assist in predicting the composition of the gases that are formed.^Although the development of this program was directed toward ISV applications, it should be applicable to other technologies where chemical reactions are of interest.^This document presents the mathematical methodology of the APOLLO computer code.^APOLLO is a computer code that calculates the products of both equilibrium and kinetic chemical reactions.^The current version, written in FORTRAN, is readily adaptable to existing transport programs designed for the analysis of chemically reacting flow systems.^Separate subroutines EQREACT and KIREACT for equilibrium ad kinetic chemistry respectively have been developed.^A full detailed description of the numerical techniques used, which include both Lagrange multiplies and a third-order integrating scheme is presented.^Sample test problems are presented and the results are in excellent agreement with those reported in the literature.
634	The separation of radionuclide migration by solution and particle transport in LLRW repository buffer material	Laboratory-scale lysimeter experiments were performed with simulated waste forms placed in candidate buffer materials which have been chosen for a low-level radioactive waste repository.^Radionuclide releases into the effluent water and radionuclide capture by the buffer material were determined.^The results could not be explained by traditional solution transport mechanisms, and transport by particles released from the waste form and/or transport by buffer particles were suspected as the dominant mechanism for radionuclide release from the lysimeters.^To elucidate the relative contribution of particle and solution transport, the waste forms were replaced by a wafer of neutron-activated buffer soaked with selected soluble isotopes.^Particle transport was determined by the movement of gamma-emitting neutron-activation products through the lysimeter.^Solution transport was quantified by comparing the migration of soluble radionuclides relative to the transport of neutron activation products.^The new approach for monitoring radionuclide migration in soil is presented.^It facilitates the determination of most of the fundamental coefficients required to model the transport process.
828	Development of technology for the long-term stabilization and closure of shallow land burial sites in semiarid environments	The eight-year field research program involving the development of technology for the closure and stabilization of shallow land burial (SLB) sites is described.^Results of field testing of biointrusion barriers at active waste disposal areas at Los Alamos and at the Los Alamos Experimental Engineered Test Facility (EETF) are reported.^Field experiments performed in 6-m-deep caissons with a diameter of 3 m at the EETF are described in which the performance of migration and capillary barriers were tested.^Finally, the results of model verification and validation efforts associated with hydrologic and chemical transport processes in the field experiments are described.^75 refs., 23 figs., 9 tabs.

[1]This information was extracted from the PNNL database. No effort was made to correct the information or improve its appearance.

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
863	Technology needs for selecting and evaluating high-level waste repository sites in crystalline rock	This report describes properties and processes that govern the performance of the geological barrier in a nuclear waste isolation system in crystalline rock and the state-of-the-art in the understanding of these properties and processes.^Areas and topics that require further research and development as well as technology needs for investigating and selecting repository sites are presented.^Experiences from the Swedish site selection program are discussed, and a general investigation strategy is presented for an area characterization phase of an exploratory program in crystalline rocks.^255 refs., 65 figs., 10 tabs.
993	Measurement apparatus of water flow rate in aquifer zone and its performance	As an apparatus attached to the simulation apparatus for environmental radionuclide migration, a measurement apparatus of water flow rate equipped with a heater and two sensors was made for the measurement of the distribution of water flow rate.^In the aquifer zone vessel of 900 mmW x 2700 mmL x 450 mmH, the measurement was carried out by heat pulse method.^In the range of water flow rate of 0 - 1.25 mm/min, the measured values in water flow rate had some scatters at 45 measured points.^The mean value of the distribution observed was close to the setting value and maximum error of 12 % had been obtained in the measurement.^The sufficient performance as measurement apparatus of water flow rate in aquifer zone was ascertained here.
1165	Method and instrument needs for detailed site characterization for a repository for spent nuclear fuel in Sweden	Site investigations in Sweden for a repository for spent nuclear fuel started in 1977.^Around 1990 two or three out of ten or more possible sites will be selected for more detailed studies, and around 2000 one site will be selected for a licence application.^When the detailed characterization stage begins, an extended investigation programme with new and improved methods and instruments will be required.^To obtain data on the geometry of fracture zones and the distribution of rock types, cross-hole radar and seismic methods are expected to be extensively utilized.^Cross-hole interference tests will be performed for determination of hydrogeological conditions.^Measurements of rock stress and strength will be important in evaluation of hydraulic anisotropy, geological and tectonic history and repository design.^The geochemical system has to be studied in detail by means of in situ measurements in boreholes.^Methods have to be established to validate and verify conceptual models and model calculations of average groundwater flow and nuclide migration.^These methods include comparisons with the surface water budget, the groundwater head distribution and the natural groundwater flow.^At the end of the detailed characterization programme, a shaft will be sunk in at least one of the investigated sites.^The results will be used to further improve the conceptual model and to verify the calculated groundwater flow and nuclide transportation.^ 8 refs, 3 figs.

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
1277	Research needs related to element migration and fixation	Regulatory requirements (NRC 40 CFR 191) will necessitate a scientific investigation of the possible migration of radionuclides released from repositories into the nearest aquifers.^Projections of radionuclide migration in groundwater must extend at least 10,000 years into the future and perhaps as long as 100,000 years.^Unfortunately, transport equations based only on available hydrologic data gathered during the past few decades cannot be extended beyond a few hundred years without requiring great speculation.^To meet these requirements, the author recommends that research should include analytical improvements for the determination of Cl/sup 36/, Kr/sup 81/, and I/sup 129/.^Basic analytical developments are also needed for determining small natural concentrations of Tc/sup 99/ and Ca/sup 41/.^Field studies are needed to further confirm the usefulness of the foregoing radionuclides.^There is also a great need for integrated hydrodynamic-environmental isotope models.^These models will be far superior to existing hydrodynamic-based models for long-term projections of contaminant transport.
1278	Thermodynamic properties of actinides: basic research needs	Computer models have shown that actinides, U through Am, are major contributors of waste radioactivity for storage times of 1000 years or longer.^In the event that the canisters of waste forms fail to contain the radioactive waste materials, water is expected to provide the main mechanism by which these elements could be transported to the accessible environment.^The released actinides can react with various components of the system, e.g. components of groundwater, dissolved waste form, backfill, and host rock, to form insoluble phases and solution species that will control their solution concentrations and migration rates.^The identities and solubilities of solid phases and the nature and concentrations of solution species depends on the oxidation states of the actinides, the nature and concentrations of precipitating and complexing ions in the modified groundwater, temperature, and ionic strength.^The author proposes research needs to address these issues.
1279	Research needs in water-rock interactions: effects on permeability and subsurface fluid flow	Quantitative assessment of the near- and far-field effects of the disposal of nuclear waste in deep repositories requires the development of a capability to analyze coupled mass transport with chemical reactions in multi-dimensional space.^Crucial to this goal is experimental work to describe the reaction kinetics at the pore level, with further parallel development in model equation formulation at the continuum level.^Much more research in numerical methods is needed to solve the coupled and nonlinear set of transport equations efficiently.^Further testing is also warranted on the effects of water-rock interactions that can cause permeability changes in geologic media.^And understanding condition role of chemical reactions in mass transport processes is important.

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
1348	Annihilation of transuranic elements	In order to make nuclear power generation into stable energy source, the establishment of nuclear fuel cycle is indispensable.^As one of its subjects, there is the annihilation treatment of long half-life nuclides contained in high level waste, of which the research and development as the innovative technology for nuclear energy have been advanced.^If the long half-life nuclides are separated from high level waste liquid, and can be burnt in nuclear reactors to change them to short half-life nuclides, the treatment and disposal of high level waste become easy, and it can contribute to obtaining the social consensus on the development of nuclear energy.^The wet process separation of TRU and its annihilation in reactors, the annihilation of TRU by the collision with high energy protons, the reduction of secondary waste accompanying the separation and annihilation and so on have been researched.^The dry process separation technology being developed by Central Research Institute of Electric Power Industry and the annihilation treatment technology using metallic fuel FBRs also being developed by CRIEPI are explained.^(K.I.).
1350	Reduction of nuclear waste with ALMRS	The Advanced Liquid Metal Reactor (ALMR) can operate on LWR discharged material.^In the calculation of the reduction of this material in the ALMR the inventory of the core should be taken into account.^A high reduction can only be obtained if this inventory is reduced during operation of ALMRs.^Then, it is possible to achieve a high reduction upto a factor 100 within a few hundred years.^(orig.)
1362	Laboratory development of sludge washing and alkaline leaching processes: Test plan for FY 1994	The US Department of Energy plans to vitrify (as borosilicate glass) the large volumes of high-level radioactive wastes at the Hanford site.^To reduce costs, pretreatment processes will be used to reduce the volume of borosilicate glass required for disposal.^Several options are being considered for the pretreatment processes: (1) sludge washing with water or dilute hydroxide: designed to remove most of the Na from the sludge, thus significantly reducing the volume of waste to be vitrified; (2) sludge washing plus caustic leaching and/or metathesis (alkaline sludge leaching): designed to dissolve large quantities of certain nonradioactive elements, such as Al, Cr and P, thus reducing the volume of waste even more; (3) sludge washing, sludge dissolution, and separation of radionuclides from the dissolved sludge solutions (advanced processing): designed to remove all radionuclides for concentration into a minimum waste volume.^This report describes a test plan for work that will be performed in FY 1994 under the Sludge Washing and Caustic Leaching Studies Task (WBS 0402) of the Tank Waste Remediation System (TWRS) Pretreatment Project.^The objectives of the work described here are to determine the effects of sludge washing and alkaline leaching on sludge composition and the physical properties of the washed sludge and to evaluate alkaline leaching methods for their impact on the volume of borosilicate glass required to dispose of certain Hanford tank sludges.

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
1369	Project Management Plan for the Idaho National Engineering Laboratory Waste Isolation Pilot Plant Experimental Test Program	EG G Idaho, Inc. and Argonne National Laboratory-West (ANL-W) are participating in the Idaho National Engineering Laboratory's (INEL's) Waste Isolation Pilot Plant (WIPP) Experimental Test Program (WETP).^The purpose of the INEL WET is to provide chemical, physical, and radiochemical data on transuranic (TRU) waste to be stored at WIPP.^The waste characterization data collected will be used to support the WIPP Performance Assessment (PA), development of the disposal No-Migration Variance Petition (NMVP), and to support the WIPP disposal decision.^The PA is an analysis required by the Code of Federal Regulations (CFR), Title 40, Part 191 (40 CFR 191), which identifies the processes and events that may affect the disposal system (WIPP) and examines the effects of those processes and events on the performance of WIPP.^A NMVP is required for the WIPP by 40 CFR 268 in order to dispose of land disposal restriction (LDR) mixed TRU waste in WIPP.^It is anticipated that the detailed Resource Conservation and Recovery Act (RCRA) waste characterization data of all INEL retrievably-stored TRU waste to be stored in WIPP will be required for the NMVP.^Waste characterization requirements for PA and RCRA may not necessarily be identical.^Waste characterization requirements for the PA will be defined by Sandia National Laboratories.^The requirements for RCRA are defined in 40 CFR 268, WIPP RCRA Part B Application Waste Analysis Plan (WAP), and WIPP Waste Characterization Program Plan (WWCP).^This Project Management Plan (PMP) addresses only the characterization of the contact handled (CH) TRU waste at the INEL.^This document will address all work in which EG G Idaho is responsible concerning the INEL WETP.^Even though EG G Idaho has no responsibility for the work that ANL-W is performing, EG G Idaho will keep a current status and provide a project coordination effort with ANL-W to ensure that the INEL, as a whole, is effectively and efficiently completing the requirements for WETP.
1385	Engineering scale electrostatic enclosure demonstration	This report presents results from an engineering scale electrostatic enclosure demonstration test.^The electrostatic enclosure is part of an overall in-depth contamination control strategy for transuranic (TRU) waste recovery operations.^TRU contaminants include small particles of plutonium compounds associated with defense-related waste recovery operations.^Demonstration test items consisted of an outer Perma-con enclosure, an inner tent enclosure, and a ventilation system test section for testing electrostatic curtain devices.^Three interchangeable test fixtures that could remove plutonium from the contaminated dust were tested in the test section.^These were an electret filter, a CRT as an electrostatic field source, and an electrically charged parallel plate separator.^Enclosure materials tested included polyethylene, anti-static construction fabric, and stainless steel.^The soil size distribution was determined using an eight stage cascade impactor.^Photographs of particles containing plutonium were obtained with a scanning electron microscope (SEM).^The SEM also provided a second method of getting the size distribution.^The amount of plutonium removed from the aerosol by the electrostatic devices was determined by radiochemistry from input and output aerosol samplers.^The inner and outer enclosures performed adequately for plutonium handling operations and could be used for full scale operations.

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
1395	Denitrification of reprocessing concentrates of middle activity	In order to reduce the releases from the Marcoule reprocessing plant, the treatment of liquid waste of low and medium level activity by chemical precipitation has been replaced by evaporation.^Due to the high nitrate content of liquid waste, encapsulation in bitumen of the concentrate leads to considerable volumes of waste to be stored in geological formation.^For safety reasons and so as to reduce the volume of waste, the elimination of the nitrates is essential: there exist various means: electrodialysis, biological denitration, chemical denitration and incineration.^In view of the very high sodium nitrate content of the concentrate, electrodialysis and biological denitration were discarded.^Preliminary experiments carried out at Cadarache led us to choose calcination in a fluidized bed rather than chemical denitration using a mixture of formic and phosphoric acids.^Tests on a low temperature mock-up have determined the choice of an injection system that operates with liquid under pressure with the nozzle situated inside the fluidized layer.^So as to avoid the vaporization of the liquid within, the injection piping also requires a cooling system using air, with a double casing.^Under these conditions, liquid can be injected into the reactor without encountering any special difficulty: no plugging of the nozzle, a regular flow and liquid, stable temperature and pressure levels from top to bottom of the reactor.^Differential thermogravimetric and heat analyses have led to the following conclusions: - at temperatures below 500 deg C, the nitric acid, then the aluminium nitrate decompose and produce alumina.^- between approximately 570 deg C and 630 deg C, the sodium nitrate in turn decomposes and reacts with the alumina to produce a sodium aluminate.^-finally, these tests enabled a reaction kinetics low of sodium nitrate decomposition in the temperature range of 500 deg C to 1000 deg C to be established.^(author).
1400	Continuous Oxidation/Reduction System (CORS)	A Continuous Oxidation-Reduction System (CORS) is being developed at Lawrence Livermore National Laboratory (LLNL).^The CORS combines reduction and oxidation reactions within a single reactor.^The materials being processed are fed continuously while the products are continuously removed.^The system is being developed for the continuous oxidation and/or reduction of reactive metals, salts, oxides, and other compounds.^Recent feasibility tests reducing cerium oxide were successful.^Two applications of this system currently being examined are: actinide oxide reduction and the separation of radionuclides from wastes and residues.^In comparison to traditional processes such as fluorination/bomb reduction and Direct Oxide Reduction (DOR), CORS reduces operator radiation exposure, waste, processing time and equipment requirements.^A patent has been filed for this process.

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
1419	Efforts for volume reduction of low-level waste in Korea/development of borate waste drying system and nonboiling forced draft evaporator	R D programs on high volume reduction technologies have been carried out at KAERI to decrease required disposal capacity.^Two of them are described in the paper.^The process of borate waste consists of drying and solidification units.^The main keys of two unit process are lime added to waste to prevent adherence at heating surface and dilution degree of unsaturated polyester with styrene to improve workability.^The results of this study show that the volume reduction factor is 6 tunes higher than that of the conventional processes.^On the other hand, nonboiling forced draft evaporation system (NBFE) had been developed to treat very low level liquid radwaste (VLAW) generated at KAERI from zero release concept.^Evaporation rate, decontamination factor and environmental impact assessment were studied under various weather conditions such as relative humidity, temperature, wind velocity and so on.^The experimental results indicate that the evaporation rate is 0.028 V[sup 0.62](Psm - Pa) (kg/m[sup 2]hr).^Radioactivity in discharged air is about 10[sup [minus]3] Bq/m[sup 3] air under the most conservative conditions.^Consequently, no environmental impact is detected when this technology is used for the treatment of VLAW.^Based on the results, a commercial facility has been designed and constructed.
1439	Recent advances in in situ vitrification	In Situ Vitrification (ISV) is an innovative mobile remediation technology for soils and other underground contamination: Developed by the US Department of Energy's Pacific Northwest Laboratory (PNL), ISV has advanced during the past decade from a laboratory concept to a remediation technology commercially available for contaminated soils.^ISV technology is currently being developed for remediation of DOE waste sites at Hanford, Oak Ridge National Laboratory (ORNL) Idaho National Laboratory (INEL), and other sites.^The incentives for application of ISV can convert contaminated sites to a solid, highly durable block similar to naturally occurring obsidian.^The ISV product has been shown capable of passing US Environmental Protection Agency (EPA) tests such as the Toxic Characteristic Leach Procedure (TCLP).^Retrieval, handling and transport of untreated hazardous material would normally not be required after application of ISV.^Therefore, costs, exposure to personnel, risk of releases to the environment, and generation of secondary wastes are greatly reduced compared with remove-and-treat technologies.
1440	Current technology for radioactive waste treatment and decontamination	Various activities conducted continuously by Japanese electric power companies and suppliers have brought about significant improvements in reducing the volume of radioactive waste generated, minimizing the amount of radioactive material discharged from nuclear power stations, and decreasing employees' exposure to radiation.^Toshiba is now focusing on modifying the design of the latest radioactive waste systems for future use and the future operation of boiling water reactors (BWRs), taking into consideration the recent trends of (1) the construction of burial sites for low-level waste, (2) environmental protection, and (3) reduced waste generation after effective countermeasures.^Toshiba will also manage large-scale decontamination work by combining several cleaning techniques.^These steps will result in even greater reductions in the radiation dose rate.^(author).

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
1443	Decontamination of alpha-bearing solid wastes and plutonium recovery	Nuclear activities in the Radiochemistry building of Fontenay-aux-Roses Nuclear Research Center concern principally the study of fuel reprocessing and the production of transuranium isotopes.^During these activities solid wastes are produced.^In order to improve the management of these wastes, it has been decided to build new facilities: a group of three glove-boxes named ELISE for the treatment of [alpha] active solid waste and a hot-cell, PROLIXE, for the treatment of solid wastes.^Leaching processes were developed in order to: decontaminate these wastes and recover actinide elements, particularly the highly valuable plutonium, from the leachates.^The processes developed are sufficiently flexible to be able to accommodate solid wastes produced in other facilities.^Laboratory studies were conducted to develop the leaching process based on the use of electrogenerated Ag(II) species which is particularly suitable to provoke the dissolution of PuO[sub 2].^Successful exhaustive Pu decontaminations with DF(Pu) higher than 10[sup 4] were achieved for the first time during the treatment of stainless steel PuO[sub 2] cans (future MELOX plant) by electrogenerated Ag (II) in nitric acid medium.
1446	Field test of ELOMIX radioactive waste treatment process for decontamination solutions	The objective of the Electrochemical Ion Exchange process is to reduce the volume of waste arising from LOMI decontamination operations.^This is achieved by using the conventional ion exchange resin as an intermediate, rather than a final waste form.^Radioactive and metallic constituents removed during the decontamination are converted by the process to a metallic form.^Laboratory work on the concept of Electrochemical Ion Exchange has been taking place since May 1989.^In October 1990 a small pilot scale ELOMIX cell, was operated successfully on a side stream of actual decontamination solution at Commonwealth Edisons Dresden Unit 2 Nuclear Plant.^This was reported in EPRI report NP-7277 (May 1991).^Following the Dresden demonstration a second unit was constructed to undertake a further demonstration test on a larger scale prior to construction of commercial scale equipment.^The objectives of the test were to demonstrate operation in a pressurized system with a multiple (as opposed to single) compartment cell, to develop process control capability, to address waste handling issues and demonstrate clean-up and transportability.^It was agreed that the unit would be operated at an on-site demonstration at Gulf States Utilities River Bend Plant.^This demonstration took place in May 1992, and is described in this report.^The test successfully achieved the objectives.^In particular the demonstration has given confidence that an ELOMIX cell can be operated and then cleaned up to radiation levels low enough to permit transportation to another site, and that the metallic waste can be transferred hydraulically in much the same way as ion exchange resin.^Thisreport discusses the implications of the test and covers the recommendations for future work to cover the remaining R D issues.^A program of work is now under way to cover these issues before the construction and operation of commercial scale equipment.

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
1468	Development of chemical decontamination process with sulfuric acid-cerium(IV) for decommissioning	Chemical decontamination techniques for decommissioning are divided into two groups : a predismantling system decontamination for a man-rem saving and a decontamination of dismantled components for reduction of radioactive wastes volume.^At JAERI, efforts have been made to develop both the decontamination processes with H[sub 2]SO[sub 4]-Ce[sup 4+] (SC) and H[sub 2]SO[sub 4] solutions.^In the chemical decontamination processes, it is required to reduce the amounts of secondary wastes as well as to achieve a high decontamination factor (DF).^For this purpose, it is very important to select an optimum waste-solution treatment process from the point of characteristics and volume of waste solutions.^This report describes outline of the SC decontamination processes, laboratory testing results for various waste treatments of the SC spent solutions such as electrodialysis, diffusion dialysis, ion-exchange resins.^Finally, an optimum waste treatment process is proposed and discussed from the reduction of the amounts of final wastes.^(author).
1469	Decontamination method for radioactively contaminated metal	A decontaminating liquid having a carbonic acid gas dissolved therein is filled as an electrolyte in a decontamination vessel.^If DC current is supplied between a metal to be decontaminated and a counter electrode, a corrosion membrane formed on the inner surface of the metal to be decontaminated is peeled off in a flaky manner.^When the electrolysis step is completed, the decontaminating liquid in the decontamination vessel is sent to a carbonic acid gas degassing vessel.^When the carbonic acid gas is discharged out of the decontaminating liquid by the injection of the degassing gas, and pH is in an neutral region, iron ions dissolved out of the substrate of the metal to be decontaminated and deposited as hydroxides.^Radiation in the decontaminating solution having carbonic acid gas, etc. degassed in the degassing vessel is recovered as a solid material by a filter or a strainer in a radioactivity recovering step of the decontaminating liquid.^With such procedures, processing cost is decreased and volume of secondary radioactive wastes is reduced.^(I.N.).
1478	Treatment of solid waste highly containimated by alpha emitters	In the recent years, efforts have been made in order to reduce the amount of alpha emitters essentially plutonium isotopes present in the solid wastes produced either during research experiments on fuel reprocessing, done in the Radiochemistry building in the centre d'etudes nucleaires de FONTENAY-AUX-ROSES (CEA, FRANCE), or in the MARCOULE reprocessing plant (COGEMA, FRANCE).^The goals defined for the treatments of these different wastes were: to reduce their [alpha] and [beta], Y contamination levels; to recover the plutonium, a highly valuable material, and to minimize its quantity to be discharged with the wastes.^To achieve these goals leaching processes using electrogenerated Ag (II(a very aggressive agent for PuO[sub 2])) in nitric acid solutions, were developed and several facilities were designed and built to operate the processes.^A brief description of the process and of the different facilities will be presented in this paper; the main results obtained in ELISE and PROLIXE are also summarized.

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
1501	Cryofracture as a tool for preprocessing retrieved buried and stored transuranic waste	This paper summarizes important features of an experimental demonstration of applying the Cryofracture process to size-reduce retrieved buried and stored transuranic-contaminated wastes.^By size reducing retrieved buried and stored waste, treatment technologies such as thermal treatment can be expedited.^Additionally, size reduction of the waste can decrease the amount of storage space required by reducing the volume requirements of storage containers.^A demonstration program was performed at the Cryofracture facility by Nuclear Remedial Technologies for the Idaho National Engineering Laboratory.^Cryofracture is a size-reducing process whereby objects are frozen to liquid nitrogen temperatures and crushed in a large hydraulic press.^Material s at cryogenic temperatures have low ductility and are easily size-reduced by fracturing.^Six 55-gallon drums and six 2 {times} 2 {times} 8 ft boxes containing simulated waste with tracers were subjected to the Cryofracture process.^Data was obtained on (a) cool-down time, (b) yield strength of the containers, (c) size distribution of the waste before and after the Cryofracture process, (d) volume reduction of the waste, and (e) sampling of air and surface dusts for spread of tracers to evaluate potential contamination spread.^The Cryofracture process was compared to conventional shredders and detailed cost estimates were established for construction of a Cryofracture facility at the Idaho National Engineering Laboratory.
1508	Electrocell method and apparatus	This invention consists of an apparatus including Kingdomelectrocell and, preferably, also a diaphragm pump for processing waste materials and recovering metals therefrom.^The electrocell comprises a set of coaxial cylinders with end housings to maintain spacing.^The innermost cylinder is a cathode.^Surrounding the cathode is a porous barrier, then an anode and finally an outer shell.^Interior to the cathode is a cathode coolant passage.^Between the cathode and the porous barrier is a passage for a catholyte.^Between the porous barrier and the anode is a passage for an anolyte.^Exterior to the anode but inside an outer shell is a passage for the anode coolant.^The passage for the anolyte is dimensioned, based upon the flow rate of the anolyte, to produce fully turbulent flow, preferably with a Reynolds number in excess of 4000.^The diaphragm pump surges anolyte into the anolyte passage to further increase turbulence, thereby increasing the efficiency of the apparatus.

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
1510	Slow demolition of thick wall using hydrostatic tube ^Example of dismantling RC structures in radioactive facilities	In the dismantling of reinforced concrete (RC) structures of radioactive facilities such as nuclear power plants, methods that suppress sound, restrict vibrations to a minimum, prevent entry of dust and contamination and reduce waste volume resulting from dismantling work to a minimum, need to be adopted.^Recently, a demolition method using hydrostatic tube developed by Kajima Corporation was used in the dismantling of thick RC radiation barrier wall built in a nuclear power plant building and excellent results have been attained.^This method utilizes a diamond disk cutter for cutting reinforced bars and a diamond core drill for drilling the hydrostatic loading holes.^For pressurization fracture, a hydrostatic tube was used, cracks were generated in the thick walls (within the scope of the demolition plan), and concrete breakers, were utilized for secondary break-up, then the concrete pieces and debris were collected.^The adoption of this method greatly simplified concrete breaking work, reduced the working time and at the same time, contributed considerably to the prevention of dust propagation.^Vinyl sheets were used to protect the cooling water for cutter and drill bits during operation.^The vinyl sheets helped to prevent water splashes, simplified clean-up operations and assisted in complete recovery of the cooling water.^The RC demolition conditions, basic characteristics of hydrostatic tube, outline of equipment, work methods and work results based on work execution examples, are reported in this paper.^(author).
1511	Chromatographic separations of zirconium isotopes with reduced waste liquor	This patent describes a continuous chromatographic process for the simultaneous separation of each of the isotopes of zirconium in a sample containing a mixture of zirconium isotopes to produce substantially pure fractions of each of the separated isotopes.^It comprises loading a substantially vertical chromatographic separation column having an effective column height sufficient to resolve each the zirconium isotope into a distinct product fraction having a purity greater than 90% with a cation exchange resin including pentavalent phosphorus-derived active groups having a strong affinity for zirconium cations and complexes thereof; preparing an aqueous feed solution of ionic zirconium from the sample having a pH of between about 3 and 4 from a solubility limit of about 90 g/l; preparing an eluant capable of displacing zirconium ions from the exchange resin, wherein the eluant is an aqueous solution of a strong mineral acid having a normality greater than one; feeding the aqueous zirconium feed solution to the top of the located chromatographic separation column so that the feed solution begins to travel down the column.
1513	Separation of Cs-127, Sr-90, and Th-232 in aqueous solution by using a multistage countercurrent batch contactor ion-exchange system	The radioisotopes Cs-137, Sr-90, and Th-232, in simulated high level nuclear liquid waste, were recovered and sequentially separated by using an 8-stage flask simulation of a multistage countercurrent batch contractor (MCBC) ion-exchange system.^The solution normality ratio used for the three ions Cs:Sr:Th was 1:1:1.^The solution was contacted for 2 h with resin Dowex HCR-W2 initially in the H{sup +} form.^The obtained recovery efficiency of the MCBC and the purity of the separated ions for the three ions were 77 and 60% for Cs{sup +}, 64 and 93% for Sr{sup 2+}, and 99 and 65% for Th{sup 4+}, respectively.

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
1550	Method of removing radioactive waste	A paste prepared by mixing a mixed acid containing HF and at least one of HCl and HNO{sub 3} with a paste aid is coated at the surface of radioactive wastes, to dissolve the surface thereof.^Water is jetted to remove the dissolved radioactive contaminants and the pastes from the surface of the radioactive wastes.^Since the pastes are thus used, the amount of liquid wastes can be remarkably reduced compared with that in a conventional electrolysis method.^Further, if it is confirmed that dose rate of the radioactive wastes after decontamination is lower than a predetermined level by adding a step of measuring the extent of contamination of the wastes before and after the steps, they can be handled hereinafter being regarded as ordinary wastes.^(T.M.).
1553	Method of processing chloride waste	In a method of applying molten salt electrolysis to chloride wastes discharged from a electrolytic refining step of a dry reprocessing step for spent fuels, and removed with transuranium elements of long half-decaying time, metals capable of alloying with alkali and alkaline earth metals under melting by electrolysis are used as a cathode material, and an electrolytic temperature is made higher than the melting point of salts in a molten salt electrolysis bath, to recover Li, Ca and Na as alloys with the cathode material in a first electrolysis step.^Then, the electrolytic temperature is made higher than the melting point of the chloride salts remained in the bath after the electrolysis step described above by using the cathode material, to recover Ba, Rb, Sr and Cs of nuclear fission products also as alloys with the cathode material in a second electrolysis step.^Accordingly, the amount of wastes formed can be reduced, and the wastes contain no heat generating nuclear fission elements.^(T.M.).
1563	Solidification processing method for radioactive waste	The pressure in a vessel containing radioactive wastes is previously reduced and cement mortar prepared by kneading cement, sand and kneading agent with water is poured under shaking substantially to the upper end of the vessel.^After the lowering of the mortar level due to the deforming has been terminated, the pressure is increased gradually.^Then, the cement mortar is further poured substantially to the upper end of the vessel again.^With such a two step pouring method, spaces other than the radioactive wastes in the vessel can be filled substantially completely with the cement mortar.^Accordingly, it is possible to avoid the problem in view of the strength due to the formation of gaps at the inside of the vessel, or leaching of radioactive materials due to the intrusion of water into the gaps.^Further, if washing water is reutilized as water for kneading WASTEwashing after the precipitation of the solid contents, the amount of the secondary wastes generated can be reduced.^(T.M.).

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
1570	Electric melting furnace for waste solidification	To avoid electric troubles or reduction of waste processing performance even when platinum group elements are contained in wastes to be applied with glass solidification.^For this purpose, a side electrode is disposed to the side wall of a melting vessel and a central electrode serving as a counter electrode is disposed about at the center inside the melting vessel.^With such a constitution, if conductive materials are deposited at the bottom of the furnace or the bottom of the melting vessel, heating currents flow selectively between the side electrode and the central electrode.^Accordingly, no electric currents flow through the conductive deposits thereby enabling to prevent abnormal heating in the bottom of the furnace.^Further, heat generated by electric supply between the side electrode and the central electrode is supplied efficiently to raw material on the surface of the molten glass liquid to improve the processing performance.^Further, disposition of the bottom electrode at the bottom of the furnace enables current supply between the central electrode and the bottom electrode to facilitate the temperature control for the molten glass in the furnace than in the conventional structure.^(I.S.).
1577	The PHOENIX Concept	A proposed means of transmuting key long-lived radioactive isotopes, primarily the so-called minor actinides (Np, Am, Cm), using a hybrid proton-accelerator-sub-critical lattice, is described.^It is argued that by partitioning the components of the light water reactor (LWR) spent fuel and by transmuting key elements, such as the plutonium, the minor actinides, and a few of the long-lived fission products, that some of the most significant challenges in building a waste repository can be substantially reduced.^If spent fuel partitioning and transmutation were fully implemented, the time required to reduce the waste stream toxicity below that of uranium ore would be reduced from more than 10,000 years to approximately 30 years.^The proposed machine, based on the described PHOENIX Concept, would transmute the minor actinides and much of the iodine produced by 75 LWRs, and would generate usable electricity (beyond that required to run the large accelerator) of 850 MW{sub e}.^14 refs., 29 figs.
1594	Processing system for decontaminating liquid waste	Ce{sup 4+} is recovered by means of an anion exchange resin method from radioactive metal ions dissolved in decontaminated liquid wastes, iron ions, nickel ions and chromium ions leached out from decontaminated products and Ce{sup 3+} and Ce{sup 4+} as a decontaminating agent.^Subsequently, a neutralizing agent for stabilizing only free sulfuric acid is applied to redox decontaminating liquid wastes after recovering Ce{sup +4}, is which iron ions, chromium ions and nickel ions are dissolved.^The decontaminating liquid wastes are dehydrated under microwave heating to form sulfates.^The sulfates are powderized and kept in the state as they are, or solidified separately depending on their radiation level.^When Ce{sup +4} in the decontaminating liquid wastes is thus recovered and re-utilized, the amount of the decontaminating liquid wastes formed can be decreased, and the amount of secondary wastes formed can be suppressed by forming sulfates by dehydration under heating.^(T.M.).
1617	Microwave vitrification of Rocky Flats TRU sludge	The aqueous wastes from the plutonium recovery areas at the Rocky Flats Plant (RFP) are treated in a hydroxide precipitation process to remove heavy metallic elements.^The wet sludge alone does not meet the Waste Isolation Pilot Plant --- Waste Acceptance Criteria (WIPP-WAC) prohibiting the presence of free liquids.^In the present operation a Portland cement/diatomite mixture is added to the waste container to absorb free liquids.^The TRU waste forms presently produced at RFP, with the absorbants, meet the criteria established by the WIPP-WAC.^Bench scale microwave vitrification tests and pilot scale tests using TRU radioactive waste has begun.

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
1645	A method for heavy metal removal from and deactivation of radioactive liquid wastes	A method is suggested for removing heavy metals from and deactivating liquid radioactive wastes, such as waste waters, solutions and suspensions in the uranium industry, during their purification and neutralization.^Liquid wastes are conventionally treated with lime which, however, is not always available in the required amount and quality.^It is proposed that lime can be conveniently replaced by carbide lime.^Its assets include a higher reactivity, more favorable granulometric composition, a higher quality with respect to the CaO nad MgO contents, and the presence of soot and coke as adsorbents for radionuclides.^The radionuclides are thus removed by adsorption, and the heavy metal contents of the waters and solutions are reduced due to the presence of small amounts of sulfides, arsine, and phosphine.^Owing to the weakly reducing nature of carbide lime, iron separates in the divalent form during the precipitation and neutralization of waste waters, solutions and suspensions from the uranium industry, which contributes to a higher quality of the waste waters and solutions, reduction in the consumption of CaO and generation of lower volumes of sludges with higher dry matter contents.^(P.A.).
1708	Process for cesium decontamination and immobilization	Cesium can be selectively recovered from a nuclear waste solution containing cesium together with other metal ions by contact with a modified phlogopite which is a hydrated, sodium phlogopite mica.^Once the cesium has entered the modified phlogopite it is fixed and can be safely stored for long periods of time.^6 figs., 2 tabs.
1709	Decontamination of systems and components for decommissioning	For the decontamination of complete systems the CORD process is applied.^ This process is a "soft", low chemical concentration (/similar to/ 2000 ppm/cycle), multicycle decontamination process.^ It can be performed with a mobile external system or, for complete primary loop decontamination, with the systems of the nuclear power plant itself.^The amount of waste can be reduced considerably if the decontamination solution is evaporated.^ Actual decontamination tasks resulted in activity level reductions of 6 - 8 in the compartments of the systems, associated with local decontamination factors (DFs) of /similar to/ 20 or more.^ A severe problem is the decontamination for unrestricted release of dismantled components and material.^ A demonstration in decontamination was carried out with recirculation piping sections, which were replaced in 1985 in a BWR.^ About 15 tons of piping and valves were decontaminated from an initial contact dose rate of /similar to/ 30000 ..mu..Sv/h to 1 - 2 ..mu..Sv/h in a combined chemical and electrochemical treatment by CORD and ELPO decontamination processes.^This enabled the requirements of ICRP for unrestricted release to be easily met.
1718	Method of separating useful radioactive nuclide in radioactive liquid waste	Purpose: To separate useful radioactive nuclides from radioactive liquid wastes for reducing the amount of radioactive secondary wastes generated upon disposal of radioactive liquid wastes.^Method: Nitric acid is added to radioactive liquid wastes containing radioactive metal ions, iron ions, nickel ion, chromium ions and oxidative tetravalent cerium ions dissolved therein, to convert tetravalent cerium ions into complex ions.^The liquid wastes are circulated through an ion exchange resin column.^This enables to efficiently recover tetravalent cerium ions which are useful oxidative nuclides thereby enabling the reuse of cerium.^Further, since the oxidative nature of the radioactive liquid wastes is eliminated, there is no requirement of adding a reducing agent and it is possible for drying treatment and solidification processing such as plastic solidification.^(Takahashi, M.).

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
1719	Sorting method for radioactive waste	This paper describes a method for detecting radioactive components in dry active waste, comprising the steps of: providing a substantially airtight housing, withdrawing air from the housing, reducing the waste to pieces of substantially uniform size, providing a first conveyor in the housing, the first conveyor having a receiving portion and a discharge portion, discharging the pieces of reduced waste onto the first conveyor, flattening the pieces of reduced waste, detecting radiation emanating from the pieces of reduced waste from a position closely overlying the first conveyor, after the pieces are flattened, removing from the first conveyor the pieces of reduced waste from which radioactive radiation above a determined level is detected, providing a second conveyor in the housing, the second conveyor having a receiving portion and a discharge portion, disposing the second conveyor so that its receiving portion is below and spaced from the discharge portion of the first conveyor, discharging the pieces of reduced waste from the discharge portion of the first conveyor so that they fall onto the receiving portion of the second conveyor; the space between the last named discharge portion and the last named receiving portion being sufficiently great so that the pieces of reduced waste are substantially overturned and dispersed as they fall to the last named receiving portion.
1726	Solidification or radioactive waste using cement glass	As for the treatment of radioactive wastes generated in nuclear power stations, by placing emphasis on the fundamental policy of the further reduction of released radioactivity, the improvement of the reliability of the facilities, the reduction of waste generation in generating sources, the reduction of generated waste volume and the stabilizing treatment of wastes, and on the improvement of economical efficiency, the development of various new technologies and improvement have been carried out, and took root as the technologies developed in Japan.^Recently, the focus of waste treatment is the development of the solidifying treatment technology for wastes, which is excellent in volume reduction capability and long term stability corresponding contractor)the facilities for burying wastes scheduled to begin operation in 1991.^Hitachi Ltd. paid attention to the endurance of inorganic construction such as rocks, and has advanced the development of the inorganic solidifying technology for wastes, in this way, recently the prospect of putting cement glass solidifying method in practical use was obtained.^This is the first solidifying technology developed in Japan.^In this parer, the results of the development such as the verifying test using an actual scale pilot plant and the physical property test on solidified bodies are reported.^(Kako, I.).
1743	Method of solidifying radioactive waste	Purpose: To process metal oxide-containing radioactive wastes into volume-reduced solidification products at a low processing cost.^Method: The present invention concerns a method of processing medium - low level radioactive wastes generated from radioactive material handling facilities such as BWR type nuclear power plants.^Into radioactive wastes containing metal oxides, for example, FeO/sub 3/, the same amount of powdery aluminum is mixed and stirred.^Then, the mixture is set to a reaction furnace and the mixture is ignited by an ignition device.^The mixture is set to a reactor, ignited by an ignition device and brought to thermite reaction, thereby melting the metal oxide in the mixture under reduction.^According to this method, since solidification treatment is applied by utilizing the heat of the thermite reaction, the processing cost is reduced.^In addition, since the radioactive wastes are treated substantially at the same volume ratio with powdery aluminum, they can be solidified at 1/2 - 1/3 volume reduction ratio as compared with the use of conventional solidifying agent.^(Kamimura, M.).

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
1746	Using solvent extraction to process nitrate anion exchange column effluents	At the Los Alamos National Laboratory (LANL) plutonium facility, nitrate anion exchange is routinely used to purify impure plutonium scrap; however, the anion exchange process generates large volumes of low-activity aqueous waste.^Unfortunately, the evaporation/cementation process has several disadvantages.^First, valuable materials, like plutonium, cannot be recovered and have to be placed in burial sites.^Since the cost of producing plutonium is increasing and environmental restrictions are tighter, it is desirable to minimize the amount of transuranic (TRU) materials sent to burial.^Also, the waste treatment capacity is undersized to handle the anion exchange capacity currently available.^However, with the development of octyl(phenyl)-N,N-diisobutylcarbamoyl-methylphosphine oxide (CMPO) and a new centrifugal mixer-settler, it is possible to use solvent extraction as a means of recovering /sup +/III, /sup +/IV, and /sup +/VI actinides from anion exchange waste streams.^Extracting the actinides from the column waste streams reduces the activity of the solution to the level that allows it to be discharged directly to the waste handling facility, reducing the load on the evaporation/cementation processes and decreasing the volume of TRU waste generated.^The objective of this study was to evaluate CMPO and the centrifugal mixer-settlers and then design a flow sheet and a set of centrifugal contacting equipment that would remove actinides from a concentrated nitric acid waste stream.
1755	Conditioning liquid waste from I-131 production by natural TeO/sub 2/ neutron irradiation	The treatment techniques assayed to decontamination and volume reduction of the waste from I-131 production, are described.^The assayed techniques were: (NH/sub 4/)/sub 2/ TeO/sub 4/ precipitation in alcoholic medium and chemical reduction using Sn/sup 2 +/ in acid medium and too Fe/sup 2 +/ in alkaline medium.^Some considerations about products features and final results, are presented.
1760	Hanford transportable grout facility: technology and design	Grouting of selected low-level liquid waste (LLW) is a key part of the overall management strategy developed for disposal of some Hanford site wastes.^Cement-based grouts will be utilized to immobilize LLW generated from a variety of sources.^It is expected that up to 415,000 m/sup 3/ (10/sup 8/ gal) of grout may be produced over the next 25 yr.^The mixing of LLW with cementitious materials to form grout will make valuable double-shell tank storage space available for other wastes, reduce the construction of new double-shell tanks, and dispose of waste in an environmentally safe manner.^Specific grout formulations are being developed by Oak Ridge National Laboratory, tailored for each category of Hanford LLW.^Bulk powdered materials will be blended to produce a homogeneous dry mix.^Grout formulations are developed to maximize the waste loading using commercially available, low-cost raw materials.^The grout production facilities will consist of several major systems the Dry Materials Receiving and Handling Facility (DMRHF), a one-million-gal liquid feed tank, the transportable grout equipment (TGE), and the disposal system.^These facilities can be operated safely and within US Dept. of Energy orders and standards.

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
1765	Metal melting for volume reduction and recycle	This paper summarizes the experiences with melting contaminated steel materials for volume reduction and melting uranium-contaminated copper and aluminum for possible recycle.^In the past there has not been an economic incentive to reduce the volume of low-level beta-gamma contaminated metallic scrap materials in the United States.^With the rising cost of transportation and burial facility fees new interest in volume reduction is being generated.^This new interest has been primarily focused at the Idaho National Engineering Laboratory (INEL) where the Waste Experimental Reduction Facility (WERF) was established to demonstrate both metal melting and incineration of combustible material for volume reduction.^Other demonstration programs involving melting for volume reduction and recycle of aluminum and copper, as well as ferrous scrap, were related to the Cascade Improvement and Cascade Upgrade Programs (CIP/CUP) at the Paducah, Kentucky facility.^While the melting demonstrations for the CIP/CUP material were not primarily based on economic incentives, several observations recorded during the programs are of interest with regard to melting of copper and aluminum.^(4 refs., 8 tabs.)
1766	The University of Missouri Research Reactor facility can melter system	At the University of Missouri Research Reactor (MURR), a waste compacting system for reducing the volume of radioactive aluminum cans has been designed, built and put into operation.^In MURR's programs of producing radioisotopes and transmutation doping of silicon, a large volume of radioactive aluminum cans is generated.^The Can Melter System (CMS) consists of a sorting station, a can masher, an electric furnace and a gas fired furnace.^This system reduces the cans and other radioactive metal into barrels of solid metal close to theoretical density.^The CMS has been in operation at the MURR now for over two years.^Twelve hundred cu ft of cans and other metals have been reduced into 150 cu ft of shipable waste.^The construction cost of the CMS was $4950.84 plus 1680 man hours of labor, and the operating cost of the CMS is $18/lb.^The radiation exposure to the operator is 8.6 mR/cu ft.^The yearly operating savings is $30,000.^20 figs., 10 tabs.
1770	Treatment of solid waste highly contaminated by alpha emitters: Low-temperature impact crushing, leaching and incineration	Reprocessing plants, hot laboratories and fuel fabrication plants produce solid wastes containing residual amounts of plutonium and uranium in nitrate and oxide form at concentrations up to several tens of grams per m/sup 3/.^ Dismantling of nuclear facilities having handled these radioelements also generates large volumes of solid wastes highly contaminated with alpha emitters.^ It is desirable to process these alpha wastes to recover valuable fissile materials and/or permit surface storage.^ Solid waste treatment by low-temperature impact crushing and then leaching, after minimal sorting and classifying at the sites of production, meets the corresponding ally quirements for high volume reduction plus fissile material recovery or waste decontamination.^ Additional volume reduction of crushed wastes containing mainly combustible materials can be obtained by incineration.^ This is facilitated by the low fissile material content after low-temperature impact crushing and leaching.^ Sorted wastes can also be leached or incinerated directly after, in most cases, crushing by more conventional techniques.

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
1775	Radionuclide separation process (RASEP)	Liquid radwaste generated from nuclear power plants or other nuclear facilities consists of a small amount of radioactive nuclides and a large amount of non-radioactive matter.^ By separating radioactive and non-radioactive matter, original liquid radwaste can be further reduced in volume and a large portion of it can be released as non-radioactive waste.^ In addition, by fixing the separated radionuclides according contractor)their nature, it will be possible to effectively and efficiently meet the waste disposal requirements.^With this approach, JGC developed a radionuclide separation process called RASEP in which radionuclides are selectively separated from liquid waste and fixed in an inorganic adsorbent.^ As a result, maximum reduction of waste via a simple and economical method, plus safe discharge of the treated (decontamination) liquid waste to the environment can be achieved.
1778	Development of a plutonium-239 recovery incinerator	A Plutonium-239 Recovery Incinerator is being developed for the Savannah River Plant (SRP) to reduce the volume of solid glovebox waste and to allow recovery of Pu-239 from the waste.^The process will also allow treatment of some waste materials that are not certifiable for disposal at the Waste Isolation Pilot Plant (WIPP).^It will consist of two electrically heated combustion chambers (furnace and afterburner) and a dry filtration off-gas system.^A unique feature of the process is that it uses pyrohydrolysis to produce an ash that is amenable to Pu recovery through nitric acid/HF dissolution.^A series of thermogravimetric (TGA) analyses have been performed to characterize potential incinerator feed materials.^A functioning furnace mockup was built and operated to demonstrate electrically-heated pyrohydrolysis operation.^8 refs., 4 figs.
1779	Significant reduction of liquid radwaste release using innovative techniques	The Liquid Effluent Activity Reduction (LEAR) task force was formed in January 1986 with a goal of liquid effluent activity reduction (in terms of curies) by a factor of 2 relative to 1985 discharges.^This was a very challenging goal in that the San Onofre Nuclear Generating Site (SONGS) Unit 1 radwaste system design is archaic.^The radwaste system for Units 2 and 3 is undersized and has many major design deficiencies; these units have a large number of leaking fuel elements and the time constraint was strict.^The LEAR task force, under the leadership of C. Chiu and composed of representatives from all divisions directly involved in plant operation, has achieved the stated goal.^The liquid effluent activity released in 1986 totaled 1.46 Ci, a factor of 13 reduction relative to the 19 Ci released in 1985.^The task force has accomplished this remarkable deed through major innovations and has successfully translated these innovations into a 92.3% liquid waste reduction (by activity) in a short time period at a minimal capital expenditure.
1783	Geode process achieves high volume reduction (of power plant wastes)	The Geode process is a two-step crystallisation and cementation process for power plant liquid wastes.^ The process is carried out in two steps.^ The first step provides volume reduction of the liquid waste stream by use of a Mobile Crystalliser Unit; the second step immobilises and stabilises the concentrated waste solution in disposable liners by use of a Mobile Cement Solidification Unit.^

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
1790	Incineration facility for combustible solid and liquid radioactive wastes in IPEN-CNEN - Sao Paulo	A system for incinerating the combustible solid and liquid radioactive wastes was developed in order to achieve higher mass and volume reduction of the wastes generated at IPEN-CNEN/SP or received from other institutions.^The radioactive wastes for incineration are: animal carcasses, ion-exchange resins, contaminated lubricant oils, cellulosic materials, plastics, etc. The optimization of the process was achieved by considering the following factors: selection of better construction and insulating material; dimensions; modular design of combustion chambers to increase burning capacity in future; applicability for various types of wastes; choise of gas cleaning system.^The off-gas system utilizes dry treatment.^The operation is designed to function with a negative pressure.
1794	Volume reduction of low-level radiation waste by incineration	An incinerator in which the volume of low-level radiation waste is reduced by combustion, is described including: a burner housing opening downwardly, a first conduit into the burner housing through which low-level radiation waste is flowed to the interior of the housing, a second conduit into the housing through which supplemental conventional fuel is flowed to the interior of the housing, a third conduit into the housing through which primary combustion air is flowed to the interior of the housing at a substantially stoichiometric rate, means within the housing for directing the primary combustion air into a cyclonic swirl which mixes the air and waste and fuel as they are ignited, a furnace cavity mounted below the burner housing with its top entry aligned with the burner housing exit to receive the combusting mixture of waste and fuel and air, a fourth conduit connected to the burner housing to introduce secondary combustion air into the mixture in quantities providing a total air in excess of stoichiometric combustion, a vertical downward flow path extended within the furnace cavity from the connection with the burner housing in which the waste is burned in suspension, a refractory lining for the burner housing and furnace cavity, an induction fan mounted at the exit of the furnace cavity to maintain the burner housing and furnace cavity under negative pressure, means for sensing the temperature of the products of combustion which exit the furnace cavity, and means for connecting the temperature sensing means to the supplemental conventional fuel conduit to regulate the rate of fuel flow under the control of the exit temperature.

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
1801	Electrical processes for the treatment of medium active liquid wastes. Final report January 1983-April 1985	Cross-flow electrokinetic dewatering has been developed on a lab-scale into an effective process for the treatment of such wastes as gravity-settled flocs, or sludges arising from fuel storage.^ The product may be concentrated to 25-42% solids while still remaining fluid, prior to immobilization - e.g. by addition of cement powder.^ Complete retention of activity in the concentrate was observed during the treatment of Harwell low-level waste sludges due to the high solids separation factor (> 10/sup 4/).^ It is a low pressure, low temperature process - consuming only 0.03-0.13 kWh/L at permeation rates of 0.3-1.5 m/h (depending on the stream), corresponding to /sup 1//67 - /sup 1//15 that needed for evaporation.^ An advanced electrochemical ion-exchange system has been developed in which ionic material can be electrically adsorbed and eluted by polarity reversal > 1000 times, without any change in performance.^ Decontamination factors of about 2000 were achieved for Cs removal, up to 75% loading of the exchanger at flow rates of 8 bed volumes/h.^ Elution into water can give concentrates of >= 0.25 M - with consequent high volume reduction factors.^ Inorganic ion-exchangers have also demonstrated system selectivity for the removal of specific cations.^ Overall energy consumption is < 5 kWh/m/sup 3/ (/sup 1//400 evaporation).^Significant cost savings over conventional ion-exchange may accrue from the improved performance under electrical control, and the reduced volumes of waste requiring disposal.^
1812	Selective dissolution and recovery of depleted uranium from armor plate. Final report, 26 June 1986-5 May 1987	The impacted armor targets used in testing high density armor-piercing ammunition containing depleted uranium (DU) are subject to disposal as low-level radioactive waste.^Because of the costs associated with disposal of the entire armor plate and the limited use of secured commercial sites in the future, the U.S.^Army is seeking to identify and evaluate new technologies for decontaminating these armor plates.^The objectives of this Phase I SBIR program are two-fold, namely: to develop a selective solvent that can decontaminate impacted armor targets containing DU for disposal or recycle, and to identify and characterize technologies that can remove depleted uranium from the solvent for solvent recycle and uranium recovery for easier hazardous-waste disposal.
1816	Method of preparing water glass-hardened products	Purpose: To obtain optimum hardening conditions and hardening method for water glass-hardened products effective as solidifying material for dried powder of radioactive liquid waste.^Method: A method of drying to powderize and solidifying liquid waste concentrates is known as a method of volume-reducing radioactive wastes.^While various solidifying materials have been proposed, it is proposed here a solidifying method with water glass having less interaction with radioactive powderous wastes.^That is, in the method of preparating water glass-hardened products, the addition amount of a phosphate type hardener added upon hardening is set to from 0.8 to 1.2 molar ratio between phosphate hardening agent/water glass.^This enables to produce radioactive solidified wastes at high strength by using inexpensive water glass, phosphate or cement.^(Takahashi, M.).

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
1817	Method of decontaminating liquid wastes	Purpose: To regenerate water from boron-containing decontaminating liquid wastes at such a quality capable of re-using as processing water and remarkably decrease the generation amount of wastes.^Methods: Silica in boron-containing liquid wastes can not be removed by a reversed osmotic process.^Then, chemical materials other than silica in the boron-containing decontaminating liquid wastes are removed by the reversed osmotic process and silica is removed with inorganic adsorbents.^The reversed osmotic process and the inorganic adsorbent process may be conducted in any optional order.^Inorganic oxide type material is used as the inorganic adsorbent and the reverse osmotic membranes usable in the reversed osmosis can include, for example, cellulose acetate membrane and high molecular compound membrane.^The thus treated decontaminating liquid wastes can be re-used as processing water, while on the other hand, separated liquid concentrates are solidified to thereby greatly reduce the amount of wastes.^(Takahashi, M.).
1818	Evaluation of a modified Zirflex process to minimize high-level waste generation at the Idaho Chemical Processing Plant	Extensive laboratory experimentation was conducted to develop a dissolvent suitable for Zircaloy based fuels having a surface oxide coating.^Other laboratory experimentation was conducted on the precipitation and solids separation steps of the process.^Computer simulation was used to determine the stability and uranium extractability of the output stream, and potential waste volume reduction.^From these studies a conceptual flowsheet was developed which could potentially reduce HLW volumes by about 30%.^Other process alternatives being investigated achieve equal HLW volume reduction and potentially improve safety of operation.^Therefore, the Modified Zirflex process is not presently being considered for further development.^22 refs., 21 figs., 3 tabs.
1822	Method of processing radioactive liquid wastes	Purpose: To efficiently recover metal ions from radioactive wastes in an extremely small electrolytic cell.^Constitution: Radioactive liquid wastes containing sulfate ions formed by the decomposition of sulfur-containing ingredients, metal salts of catalyst or metal ions absorbed through ion exchange prepared by oxidative decomposition of ion exchange resins or like other radioactive wastes using hydrogen peroxide under the presence of metal salt catalysts are caused to flow into an electrolytic cell having a cathode made of electroconductive powder such as of carbon, copper or platinum.^The metal ions in the liquid wastes are deposited on the powder and separated from the liquid wastes.^It is also possible to remove sulfate ions in the liquid wastes reduced into the form of sulfur dioxide by heating to concentrate the liquid wastes separated from the powder while being in contact with the powder on which metal copper is deposited.^The powder from which the metal copper has been exhausted, can be re-used by returning into the electrolytic cell again.^(Sekiya, K.).

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
1838	Device for processing regenerative wastes of ion exchange resin	The operation and maintenance of a processing device is facilitated by dividing radioactive wastes produced in the regenerative process of the ion exchange resin into a regenerated usable recovery liquid and wastes.^Sulfuric acid is recovered by a diffusion dialysis method from wastes containing sulfuric acid that are generated in the regenerative process of cation-exchange resin and also caustic soda is recovered by the diffusion dialysis method from wastes containing caustic soda that are generated in the regenerative process of anion-exchange resin.^The sulfuric acid and caustic soda thus recovered are used for the regeneration of ion-exchange resin.^A concentrator is provided for concentrating the sulfuric acid and caustic soda water solution to concentration suitable for the regeneration of these ion-exchange resins.^Also provided is a recovery device for recovering water generated from the concentrator.^This device is of so simple a constitution that its operation and maintenance can be performed very easily, thereby greatly reducing the quantity of waste liquid required to be stored in drums.^(Takahashi, M.).
1843	Development and active demonstration of acid digestion of plutonium-bearing waste	The aim of the work was to develop the process of wet ashing and the component required for this for waste containing a high level of plutonium and to prove their feasibility by an active demonstration plant.^The active demonstration was provided in the context of a joint by KfK and Eurochemie Mol/Belgium.^The wet ashing of about 800 kg of waste and the recovery of 6.3 kg of plutonium in a half scale plant proved the suitability of the process and the plant components for the treatment of combustible waste containing a high level of plutonium.^High conversion rates of waste and plutonium were achieved in the reactor specially designed for this process.
1844	Advanced radioactive waste treatment system (Slim-rad)	Since the introduction of nuclear power stations into Japan, as for the radioactive wastes generated there, various improvements and the development of new techniques have been carried out based on the basic policy of the reduction of release, the heightening of reliability, the reduction of quantity generated, the decrease of volume and stabilization, and those have taken root as the Japanese own techniques.^The recent foci in waste treatment are the needs of improving the economy of nuclear power generation and the measures of the final disposal of solid wastes.^Responding contractor)these, the optimization of the design condition for waste treatment facilities based on the operational results including condition reduction of waste generation and radioactivity by the improvement of the upstream facilities in plants which are the source of generating wastes, the large scale slimming of the treatment facilities by the application of the new techniques of waste treatment, the reduction of quantity and the long term stabilization treatment of solid wastes have been carried out.^In this report, waste treatment techniques centering around the Slim-rad waste treatment system are described.^The techniques applied, features and effects of the Slim-rad system are explained.

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
1846	Electrical processes for the treatment of medium-active liquid wastes	Cross-flow electrokinetic dewatering has been developed on a lab-scale into an effective process for the treatment of such wastes as gravity-settled flocs, or sludges arising from fuel storage.^The product may be concentrated to 25-42% solids while still remaining fluid, prior to immobilization - e.g. by addition of cement powder.^Complete retention of activity in the concentrate was observed during the treatment of Harwell low-level waste sludges due to the high solids separation factor (>10/sup 4/).^It is a low pressure, low temperature process - consuming only 0.03-0.13 kWh/L at permeation rates of 0.3-1.5 m/h (depending on the stream), corresponding to 1/67 - 1/15 of that needed for evaporation.^An advanced electrochemical ion-exchange system has been developed in which ionic material can be electrically absorbed and eluted by polarity reversal > 1000 times, without any change in performance.^Decontamination factors of about 2000 were achieved for Cs removal, up to 75% loading of the exchanger at flow rates of 8 bed volumes/h.^Elution into water can give concentrates of > 0.25 M - with consequent high volume reduction factors.^Inorganic ion-exchangers have also demonstrated system selectivity for the removal of specific cations.^Overall energy consumption is < 5 kWh/m/sup 3/ (1/400 evaporation).^Significant cost savings over conventional ion-exchange may accrue from the improved performance under electrical control, and the reduced volumes of waste requiring disposal.^25 refs, 28 tabs, 114 figs.
1849	Process for preparing radioactive and/or radioactively contaminated solid wastes and evaporator concentrates for final storage in final storage containers	The solid wastes containing water are introduced into the inlet chamber of a final storage container, are heated there and dried under vacuum, where these solid wastes are reduced in volume.^The evaporator concentrate is then introduced in liquid form into the inlet chamber, which is still at sub-pressure.^The final storage container is then heated again and the concentrate water is removed.
1873	A chemical decontamination process for decontaminating and decommissioning nuclear reactors	Five chemical decontamination processes have been developed for nuclear reactor applications.^One of these processes is the cerium decontamination process (CDP).^This method uses a cerium acid reagent to rapidly decontaminate surfaces, obtaining decontamination factors in excess of 300 in 6 h on pressurized water reactor specimens.^Sound volume reduction and waste management techniques have been demonstrated, and solidified waste volume fractions as low as 9% experimentally obtained.^The CDP method represents the hybrid decontamination technique often sought for component replacement and decommissioning operations: high effectiveness, rapid kinetics, simple waste treatment, and a low solidified waste volume.
1882	Induction melting for volume reduction of metallic TRU wastes	Volume reduction of metallic transuranic wastes offers economic and safety incentives for treatment of wastes generated at a hypothetical commercial fuel reprocessing facility.^Induction melting has been identified as the preferred process for volume reduction of spent fuel hulls, fuel assembly hardware, and failed equipment from a reprocessing plant.^Bench-scale melting of Zircaloy and stainless steel mixtures has been successfully conducted in a graphite crucible inside a large vacuum chamber.^A low-melting-temperature alloy forms that has demonstrated excellent leach resistance.^The alloy can be used to encapsulate other metallic wastes that cannot be melted using the existing equipment design.

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
1889	Implementation of a non-contaminated waste segregation and environmental control program at a large Canadian utility	A study of the monitoring of non-radioactive solid waste in bags recommended changes in monitoring methods and practices.^A program has been carried out over an 8-month period using a large volume gamma monitor.^The monitoring was carried out in a central area which also considered the requirements for handling and shipping of the active and inactive wastes.^The results indicate that residual quantities of radioactivity in the inactive waste bags, were not detectable with conventional portable monitoring instruments, that the average specific radioactivity of the inactive waste stream was reduced, and that 40% of the nominally active waste could be classified as inactive.^Use of the monitor is considered to improve environmental control for non-radioactive solid wastes and provide economic benefits.
1901	Method of continuously regenerating decontaminating electrolytic solution	The purpose of this patent is to continuously recover radioactive metal ions from the electrolytic solution used for the electrolytic decontamination of radioactive equipment and increased with the radioactive dose, as well as regenerate the electrolytic solution to a high concentration acid.^A liquid in an auxiliary tank is recycled to a cathode chamber containing water of an electrodepositing regeneration tank to render pH = 2 by way of a pH controller and a pH electrode.^The electrolytic solution in an electrolytic decontaminating tank is introduced by way of an injection pump to an auxiliary tank and, interlocking therewith, a regenerating solution is introduced from a regenerating solution extracting pump by way of a extraction pipeway to an electrolytic decontaminating tank.^Meanwhile, electric current is supplied to the electrode to deposit radioactive metal ions dissolved in the cathode chamber on the capturing electrode.^While on the other hand, anions are transferred by way of a partition wall to an anode chamber to regenerate the electrolytic solution to high concentration acid solution.^While on the other hand, water is supplied by way of an electromagnetic valve interlocking with the level meter to maintain the level meter constant.^This can decrease the generation of the liquid wastes and also reduce the amount of the radioactive secondary wastes.^(Horiuchi, T.).
1902	Radioactive metal sodium processing device	A method is claimed to burn metal sodium in air into powdery sodium carbonate without containing unreacted sodium and with no generation of hydrogen.^Metal sodium to be treated is fluidized by a heater at the outer periphery of a supply tank and contained by way of a pipe into a metal vessel.^Metal sodium is burnt in the oxidizing reaction tank within the vessel while supplying air.^Then, the burning products are transferred to a geseous carbon dioxide reaction tank in the identical metal vessel.^Gaseous carbon dioxide is blown to the combustion product to form sodium carbonate.^The sodium carbonate is caused to fall by a scraper into a receiver vessel.^Smoke resulting from the combustion is released externally through a filter by way of a blower.^Since no water is used, hydrogen is not produced to eliminate the explosive danger and the protection countermeasure can be simplified.^In addition, since the product is powdery, the amount of wastes is reduced.^(Ikeda, J.).

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
1903	Method of processing radioactive metal wastes	A method is claimed to reduce the amount of wastes as the final processed products.^Contaminated equipment and devices belonging to categories 2 and 4 except for category 1 at a high radioactivity level according contractor)the IAEA classification are decomposed and decontaminated into contaminated metals at low radioactivity level.^Then, after pulverizing the radioactive material-contaminated metals and radiation material-contaminated metals reduced to such a low radioactivity level, they are blended so as to obtain an appropriate casting property and mechanical strength.^They are melted and poured into a casting mold and cast as shielding bodies.^They are utilized as shielding material to be disposed to the inside of a container such as a drum can used for the processing of radioactive metal wastes.^(Sekiya, K.).
1904	Vitrification system of low-level radioactive wastes	The concept of a vitrification system for low-level radioactive wastes is introduced.^This system aims to treat almost all sorts of radioactive wastes from nuclear power plants, including MATHEMATICAL hes, powders from condensed liquid wastes, and non-burnable solides.^After some pre-treatment, these wastes are melted in a high-frequency radio heater (1,000 - 3,000 Hz) with a ceramic melter and then vitrified and filled in canisters.^A pilot plant (100 kW, 25 - 30 kg/hr) with a ceramic melter of 45 l was built and cold tests were performed to see the effects of vitrification conditions and the characteristics of the vitrified forms.^Results are summarized in a table.^Hot tests were also performed with a ceramic melter of 4 l (30 kW, 3 - 5 kg/hr), to see the decontamination factors and the leaching characteristics of the vitrified forms.^Estimated volume reductions of final wastes relative to the conventional waste treatment systems are 1/5 for PWRs and 1/4.2 for BWRs, respectively.^(Aoki, K.).
1918	Method of processing liquid radioactive wastes from nuclear power plants	Silicon and/or phosphorus compounds are added to liquid radioactive wastes with a high content of sodium or boron salts to such an amount that the concentrations of their oxides in the final product should be 30-70 w.%.^Apart from the said vitrification substances, intermediate elements are added in form of compounds of aluminium and/or iron and/or zinc to the amount which corresponds to the concentration of oxides of these elements of 0-30 w.%.^The mixture is heated to a temperature of 150-1200 deg C producing silicate, phosphate or phosphosilicate glass.^The advantage of this solidification method is higher volume reduction of wastes, higher mechanical strength and the same or lower solubility of the end product of fixation.
1919	Volume reduction of combustible radioactive waste by cementation under pressure	Cementation under pressure for the conditioning of radioactive ashes and slags results in improved volume reduction with good physical and chemical product properties.^The cementation mixture prepared in the crushing can according to a given formula is compacted in a high-pressure press into pellets capable of being stacked in drums.^Even in the scanning electron microscope, the hardened concrete matrix exhibits a closed structure and is suitable for ultimate storage.

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
1923	Induction melting for volume reduction of metallic TRU wastes	Volume reduction of metallic transuranic wastes offers economic and safety incentives for treatment of wastes generated at a hypothetical commercial fuel reprocessing facility.^Induction melting has been identified as the preferred process for volume reduction of spent fuel hulls, fuel assembly hardware, and failed equipment from a reprocessing plant.^Bench-scale melting of Zircaloy and stainless steel mixtures has been successfully conducted in a graphite crucible inside a large vacuum chamber.^A low-melting-temperature alloy forms that has demonstrated excellent leach resistance.^The alloy can be used to encapsulate other metallic wastes that cannot be melted using the existing equipment design.^18 refs., 4 figs., 3 tabs.
1939	Solving sampling problems	The implementation of 10CFR61 and the adoption of the International Atomic Energy Agency's "Regulation for the Transport of Radioactive Materials" by the DOT require that extensive analysis be performed on the waste products that are generated by facilities which produce or utilize by-product materials.^Specifically, the waste products must be analyzed to determine the presence and concentration of by-product materials.^Representative samples of the waste products are necessary to perform this analysis.^Most waste streams are sampled with some difficulty; however, obtaining representative samples from volume reduced waste product streams is even more difficult.^Mathematical correlation models can be used to analyze these waste products, but representative sampling is still needed periodically to verify the correlations.^The purpose of this paper is to share how Duke Power Company has solved the problems encountered in obtaining representative samples from items such as cartridge filters, resin slurries, and volume reduced dry product.^Some of the problems we have encountered include minimizing personnel exposure, obtaining representative samples from heterogenous waste forms, transporting samples from high-rad areas to low-rad areas for retrieval and analysis, controlling the spread of contamination, adaptability to variations in process parameters and sampling new waste forms such as dry product from volume reduction equipment.^Solving the problems of sampling volume reduction (VR) dry products was very challenging.^Since it is basically a new waste stream, we were unable to locate a sampling device specifically designed for VR dry product.^Consequently, we developed one.^Our goal was to come up with a simple design that had few moving parts, required low maintenance, solved the problems listed above and was inexpensive.^The details of this device are discussed.
1943	Critique of the National Academy of Sciences study of the isolation system for geologic disposal of radioactive waste	The 1983 report of the Waste Isolation Systems Panel of the National Academy of Sciences (referred to as NAS-83) introduces a solubility limited dissolution (SLD) theory to estimate release rates from high-level radioactive waste packages.^It is pointed out that this theory, as presented, should apply equally well to each grain of average rock, but that when applied to that problem, it overpredicts the observed dissolution rate of SiO/sub 2/ by seven orders of magnitude.^The SLD theory also predicts that cesium and other trace elements are leached out of rock grains orders of magnitude more rapidly than the SiO/sub 2/; it is shown that this is clearly contrary to the experimentally observed situation.^Other shortcomings of the NAS-83 treatment are pointed out.^Modifications to the theory that avoid these large discrepancies are suggested; when applied to the waste problem, they pose some very important questions that should be answered before proceeding with waste management problems.^For example, they suggest that reprocessing may reduce the hazards from waste by a factor of 10 million, and that synroc may be millions of times more secure against leaching than waste glass.

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
1967	Pretreatment of sludge and floc wastes by evaporation, drying and calcination	The Magnox silo sludge and alumino-ferric hydroxide flocs represent two major volume waste streams from the reprocessing of Magnox fuel.^ Thermal pretreatments by evaporation, drying and possibly calcination prior to encapsulation in cement offer the potential for significant reductions in their final waste volumes and improvements in waste product quality.^ This paper describes the results of such pretreatments using simulated wastes in a vertical wiped film evaporator and a rotary kiln.
1969	Real-time aqueous tritium monitor using liquid scintillation counting	The ability to continuously monitor low-level tritium releases in aqueous effluents is of particular interest to heavy water facilities such as those at the Savannah River Site (SRS) and Canadian CANDU reactors.^SRS developed a real-time monitoring system based on flow-through liquid scintillation (LS) counting.^Sensitivities of 16 pCi/ml and 1 pCi/ml result from five minute and daily averages of counting data respectively.^This sensitivity is about 200 times better than similar methods using solid scintillants.^The LS system features uncomplicated sample pretreatment, precise proportioning of the cocktail and sample water, on-line quench corrections, cocktail consumption as low as 0.15 ml/min, and response to changes in environmental tritium in less than 30 min.^Field tests demonstrate that conditions necessary for stable analytical results are achieved.^((orig.))
1970	Stack gas activity release monitoring systems for NPP	The sensitivity, accuracy and stability of the measurement of stack gas activity release monitoring system for nuclear power plants are greatly enhanced by the computing facilities and functionalities provided by microcomputer based systems.^The developed stack monitoring systems, while offering all the advantages of conventional analogue system, results in lower minimum detectable level of various radioactivities.^It can be used for monitoring either gross beta, gamma radioactivity or activity release of specific isotope by employing multichannel analyser technique.^The system provides wide dynamic range of measurement, monitoring normal and abnormal releases of gaseous activities confirming to both regulatory compliance and emergency releases.^Stack gas activity release monitoring systems for KAPP have been designed by adopting standard, single height VME back plane architecture and developing modular plug in cards, the details of which are described in the paper.^(author).^1 fig.
1979	Analysis of radioactive waste samples by ion chromatography-ICP/MS	A comprehensive ion chromatography (IC) with beta-counting (beta) and inductively coupled plasma mass spectrometry (ICP/MS) detection approach has been developed to separate and detect 20 radionuclides in a Hanford waste tank sample.^The IC separation was performed using a multi-functional group (anion/cation) resin and eluents of oxalic acid, diglycolic acid, and hydrochloric acid.^Shorter-lived radionuclides were detected by a solid-state beta scintillation counter on-line with the IC separation.^Mass spectrometry detection using an efficient and robust plasma ionization source provides isotopic discernability for both stable isotopes and long-lived radioactive species.^Effective separation of over 47 elements and 160 isotopes was obtained from a single-elution scheme lasting 70 min.^Automated IC separations provide the potential for rapid isotopic and radionuclide analysis of complex radioactive waste, using minimal sample and reagent volumes and reducing personnel exposures.

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
1985	Chinese experience in the removal of actinides from highly active waste by trialkylphosphine-oxide extraction	Mixed trialkylphosphine oxide (TRPO) (alkyl is C[sub 6]-C[sub 8]) was chosen as the extractant for the removal of uranium, neptunium, plutonium, and americium from highly active waste (HAW) in China.^Composition and properties of the extractant and process chemistry are based on 30 vol% TRPO-kerosene as solvent.^Hexa- and tetravalent actinides are highly extractable in 30 vol% TRPO extraction from acidic HAW, and trivalent americium (curium) can be extracted effectively from HAW with a nitric acid concentration of [approximately]1 mol/[ell].^Actinides extracted can be stripped successively by 5.5 mol/[ell] HNO[sub 3], 0.6 mol/[ell] H[sub 2]C[sub 2]O[sub 4], and 5% Na[sub 2]CO[sub 3] into americium + rare earth, neptunium + plutonium, and uranium fractions, respectively.^The loading capacity of TRPO solvent is higher than that of bifunctional organophosphorus extractants, and the radiolytic stability of TRPO is higher than that of tributyl phosphate (TBP) and bis(2-ethyl hexyl)phosphoric acid.^The extraction and stripping rate of TRPO is high enough to be compatible with the centrifugal contactors.^Optimized process parameters of multistage countercurrent extraction and stripping and results of experimental verification are established.^In both a batch experiment with simulated nuclear power plant (NPP) spent-fuel Purex HAW and a continuous experiment with real NPP spent-fuel Purex HAW, 99.9% recovery of actinides was achieved.^The modification of the solvent system with TBP to fit the conditions in the chemical pretreatment of defense HAW is considered.
1993	Light Duty Utility Arm System applications for tank waste remediation	The Light Duty Utility Arm (LDUA) System is being developed by the US Department of Energy's (DOE's) Office of Technology Development (OTD, EM-50) to obtain information about the conditions and contents of the DOE's underground storage tanks.^Many of these tanks are deteriorating and contain hazardous, radioactive waste generated over the past 50 years as a result of defense materials production at a member of DOE sites.^Stabilization and remediation of these waste tanks is a high priority for the DOE's environmental restoration program.^The LDUA System will provide the capability to obtain vital data needed to develop safe and cost-effective tank remediation plans, to respond to ongoing questions about tank integrity and leakage, and to quickly investigate tank events that raise safety concerns.^In-tank demonstrations of the LDUA System are planned for three DOE sites in 1996 and 1997: Hanford, Idaho National Engineering Laboratory (INEL), and Oak Ridge National Laboratory (ORNL).^This paper provides a general description of the system design and discusses a number of planned applications of this technology to support the DOE's environmental restoration program, as well as potential applications in other areas.^Supporting papers by other authors provide additional in-depth technical information on specific areas of the system design.
2002	Making glass marbles	Significant increases in commercial Low Level Radioactive Waste (LLRW) burial fees, delays in US compact burial site development schedules and the tightening of NRC and EPA regulation limits have created the need for an improved LLRW stabilisation process.^Vectra Technologies is working to commercialise its EnviroGlass process - which stabilises LLRW and low level mixed wastes in a glass matrix for disposal - to serve this need.^(author).

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
2003	Graphite electrode DC arc technology program for buried waste treatment	The goal of the program is to apply EPI's Arc Furnace to the processing of Subsurface Disposal Area (SDA) waste from Idaho National Engineering Laboratory.^This is being facilitated through the Department of Energy's Buried Waste Integrated Demonstration (BWID) program.^A second objective is to apply the diagnostics capability of MIT's Plasma Fusion Center to the understanding of the high temperature processes taking place in the furnace.^This diagnostics technology has promise for being applicable in other thermal treatment processes.^The program has two parts, a test series in an engineering-scale DC arc furnace which was conducted in an EPI furnace installed at the Plasma Fusion Center and a pilot-scale unit which is under construction at MIT.^This pilot-scale furnace will be capable of operating in a continuous feed and continuous tap mode.^Included in this work is the development and implementation of diagnostics to evaluate high temperature processes such as DC arc technology.^This technology can be used as an effective stabilization process for Superfund wastes.
2015	Low-temperature-setting phosphate ceramics for low-level mixed waste stabilization	Chemically bonded phosphate ceramics (CBCs) were investigated for low-temperature stabilization and solidification of DOE mixed wastes where conventional high-temperature treatments cannot be used due to presence of volatiles and pyrophorics in the wastes.^This article deals with stabilization of chemical contaminants.^Phosphate ceramics of Mg, Mg-Na and Zr are being investigated as candidate materials.^The authors discuss the basic properties of the phosphate waste forms made with surrogates of typical DOE mixed wastes with an emphasis on ash waste stream.^The performance of the final waste forms, including leachability of the contaminants durability of the final waste forms in aqueous environment, and strength of the waste forms are discussed in detail.^Based on the results, the authors present possible economic applications of these materials.
2028	Ion-exchange performance of crystalline silico-titanates for cesium removal from Hanford tank waste simulants	A new class of inorganic ion exchangers, called crystalline silicotitanates (CSTs), has been prepared at Sandia National Laboratories and Texas A M University.^CSTs have high selectivity for the adsorption of Cs, Sr, and other radionuclides from highly alkaline, high-sodium supernate such as those found at the Hanford Site.^The applicability of CSTs to remove Cs and other radionuclides have been assessed for treating Hanford tank wastes using continuous-flow, ion-exchange columns.^This paper presents tests results which address chemical, physical, and radiological properties which are expected to be relevant for Hanford radwaste processing.^Results indicate that CSTs have a large distribution coefficient (K[sub d] >2000 mL/g in NCAW simulants) for adsorbing ppm concentrations of Cs.^CSTs exhibit very high K[sub d] values (>20,000 mL/g) for Cs in neutral pH solutions and K[sub d] values of >2000 mL/g in solutions containing 2M HNO[sub 3].^Results are presented from initial experimental efforts that describe the potential performance of the CSTs in laboratory-scale ion-exchange columns.^Included are results showing the stability of the CST material in basic solutions and in radiation doses up to 10[sup 9] rads (Si).

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
2029	Method for treating materials for solidification	According contractor)its major aspects and broadly stated, the present invention is a method for treating materials, including hazardous wastes, for solidification in a solid, substantially nonleachable product.^Addition of amorphous reactive silica to the material results in enhanced stabilization and retention of hazardous species in either glass or cement.^The process includes adjustment of the composition of the waste to take advantage of common glass-forming or cement-forming constituents already present therein.^The waste is analyzed to determine its composition, including the concentration of common glass-forming and cement-forming compounds, in order to assess the amount and type of additives needed to obtain an end product with the desired constituents.^The waste may be a liquid, a solid, or a sludge, and may contain chemical waste, radioactive waste, mixed chemical and radioactive, heavy metals, or other wastes.^The amount of each constituent needed is added to the waste, and the resulting mixture assayed to verify that its combined constituents are within the correct operational range.^The mixture is processed to form a solid, stable product.
2030	Treatment of low-level radioactive cesium-137 and technetium-99 liquid wastes by inorganic ion-exchangers	This study has been directed towards the testing and evaluation of selected inorganic ion-exchangers that are inexpensive and suitable for the conditioning and disposal of radioactive wastes.^The experiments were based on sorption characteristics of inorganic ion-exchange materials and their ability to retain radionuclides.^The sorption efficiency of cesium-137 and technetium-99 on exchangers was tested under various conditions including the effect of pH, equilibrium time, temperature and concentration for bentonite, kaolinite, sand and sandy soil, which are classified as natural inorganic exchangers.^Titanium dioxide, zeolite, antimony pentioxide and hydrated antimony pentoxide (HAP) were used as synthetic inorganic exchangers.^This report also includes a basic study of the cementation process along with the measurement of several important waste from properties such as physical stability, compressive strength and leachability.^The percentage sorption efficiency of cesium-137 was found to be 99, 98, 88, 87, 86 and 85 respectively for zeolite, kaolinite, antimony pentoxide, sandy soil, sand and HAP, at 25-50[sup 0]C, pH range 3-9 and 10-20 minutes contact time.^The sorption of technetium-99 on antimony pentoxide was 80-90, at 25[sup 0] C, pH range 1-9 and 5 days contact time.^The physical stability tests indicated that all specimens had good homogeneity.^Percentages of weight-loss after 28 day cure times at ambient room temperature were in the range of 7-15, 2-6, 6-9, 3-10 and 4-5 for bentonite, kaolinite, zeolite, titanium dioxide and sand respectively.^The compressive strength of conditioned waste form consisting of various simulated waste compositions are reported.^For the purpose of this study, an arbitrary compressive strength value of 150 kgm/cm[sup 2] was established.^The proper percentage composition of cemented wastes were 19, 24, 25, 45 and 54 for bentonite, kaolinite, zeolite, titanium dioxide and sand respectively.^(Abstract Truncated)

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
2042	Accelerator transmutation of nuclear waste: Towards the elimination of long-lived radioactive waste	Researchers at Los Alamos have been developing transmutation concepts involving accelerator-driven nuclear systems.^A medium energy, high current proton beam strikes a heavy metal target, producing a high flux of spallation neutrons.^These neutrons are moderated to near-thermal energies in a blanket surrounding the target.^Materials to be transmuted flow through the blanket region where they are fissioned or transmuted to stable nuclides.^Stable or short-lived nuclides are separated while the long-lived radioactive species are returned to the blanket.^For most applications the fission energy produced is much greater than that required to power the accelerator and can be directed to the commercial power grid.^A number of possible applications are envisioned for accelerator-driven nuclear systems.^These include destruction of surplus weapons-grade plutonium, production of tritium, transmutation of commercial spent fuel, and even commercial power generation in next-generation nuclear power plants.^Some of these applications will be discussed with particular emphasis on the required chemical separations for such systems.
2044	Summary technical report on the electrochemical treatment of alkaline nuclear wastes	This report summarizes the laboratory studies investigating the electrolytic treatment of alkaline solutions carried out under the direction of the Savannah River Technology Center from 1985-1992.^Electrolytic treatment has been demonstrated at the laboratory scale to be feasible for the destruction of nitrate and nitrite and the removal of radioactive species such as [sup 99]Tc and [sup 106]Ru from Savannah River Site (SRS) decontaminated salt solution and other alkaline wastes.^The reaction rate and current efficiency for the removal of these species are dependent on cell configuration, electrode material, nature of electrode surface, waste composition, current density, and temperature.^Nitrogen, ammonia, and nitrous oxide have been identified as the nitrogen-containing reaction products from the electrochemical reduction of nitrate and nitrite under alkaline conditions.^The reaction mechanism for the reduction is very complex.^Voltammetric studies indicated that the electrode reactions involve surface phenomena and are not necessarily mass transfer controlled.^In an undivided cell, results suggest an electrocatalytic role for oxygen via the generation of the superoxide anion.^In general, more efficient reduction of nitrite and nitrate occurs at cathode materials with higher overpotentials for hydrogen evolution.^Nitrate and nitrite destruction has also been demonstrated in engineering-scale flow reactors.^In flow reactors, the nitrate/nitrite destruction efficiency is improved with an increase in the current density, temperature, and when the cell is operated in a divided cell configuration.^Nafion[reg sign] cation exchange membranes have exhibited good stability and consistent performance as separators in the divided-cell tests.^The membranes were also shown to be unaffected by radiation at doses approximating four years of cell operation in treating decontaminated salt solution.

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
2059	Hydrochemical characterization techniques in deep boreholes	The characterisation of the hydrogeochemical parameters controlling the radionuclide migration towards the biosphere implicates the development of a methodology including instrumentation and equipment for working in deep boreholes, according with the data in which we are more interested.^This methodology consists of a systematic and detailed study of the boreholes (studies on the cores, geophysical parameters, hydraulic tests...) that allows to select zones for water sampling and characterisation.^The selected areas are adequately isolated with a double packer system and coupled instrumentation for pumping water (pumps or gas lift systems).^The isolated zone is cleaned by pumping, and the tracer concentration, tritium measures or the control of the stabilization of physico-chemical parameters, are good indicators of the representativity of the sample.^Water, gases, particles, colloids and microorganisms sampling is made in glove boxes and the analysis are made in a mobile laboratory in situ for determining the more sensible parameters, avoiding the sample alterations with the transport.^(Author) 18 refs.
2066	Off-gassing induced tracer release from molten basalt pools	Two in situ vitrification (ISV) field tests were conducted at the Idaho National Engineering Laboratory (INEL) during the summer of 1990 to assess ISV suitability for long-term stabilization of buried waste that contains transuranic and other radionuclide contaminants.^The ISV process uses electrical resistance heating to melt buried waste and soil in place, which upon cooldown and resolidification fixes the waste into a vitrified (glass-like) form.^In these two ISV field tests, small quantities of rare-earth oxides (tracers DY[sub 2]O[sub 3], Yb[sub 2]O[sub 3], and Tb[sub 4]O[sub 7]) were placed in the test pits to simulate the presence of plutonium oxides and assess plutonium retention/release behavior.^The analysis presented in this report indicates that dissolution of tracer oxides into basaltic melts can be expected with subsequent tracer molecular or microparticle carry-off by escaping gas bubbles, which is similar to adsorptive bubble separation and ion flotation processes employed in the chemical industry to separate dilute heavy species from liquids under gas sparging conditions.^Gaseous bubble escape from the melt surface and associated aerosolization is believed to be responsible for small quantities of tracer ejection from the melt surface to the cover hood and off-gas collection system.^Methods of controlling off-gassing during ISV would be expected to improve the overall retention of such heavy oxide contaminants during melting/vitrification of buried waste.
2067	Stabilization of low-level mixed waste in chemically bonded phosphate ceramics	Mixed waste streams, which contain both chemical and radioactive wastes, are one of the important categories of DOE waste streams needing aterialsabilization for final disposal.^Recent studies have shown that chemically bonded phosphate ceramics may have the potential for stabilizing these waste streams, particularly those containing volatiles and pyrophorics.^Such waste streams cannot be stabilized by conventional thermal treatment methods such as vitrification.^Phosphate ceramics may be fabricated at room temperature into durable, hard and dense materials.^For this reason room-temperature-setting phosphate ceramic waste forms are being developed to stabilize these to problem waste streams."

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
2081	Solidifying sulfate-bearing wastes to make glass composite materials	A promising method of solidifying radioactive wastes is in glasses.^The glass matrix can have high chemical and radiation stability and retain radionuclides reliably while producing minimal environmental effects.^However, vitrification causes a problem in incorporating certain waste components into the glass to obtain homogeneous products.^The wastes may contain not only components that dissolve in molten glass but also certain sparingly soluble ones such as molybdates, chlorides, and sulfates.^For example, a borosilicate glass can accommodate not more than 1% sulfate, although it is considered at present as a most promising glass for radioactive wastes.^When sulfate-bearing radioactive wastes are melted, a sludge phase is produced at the surface consisting mainly of sodium and calcium sulfates and chlorides, whose specific radioactivity is much higher than that of the glass.^It is water-soluble and greatly reduces the resistance in the final product, which complicates vitrifying sulfate-bearing wastes.^Methods of dealing with sulfate-bearing wastes have been examined.^Here the authors show that sulfate-bearing wastes can be vitrified by dispersing the insoluble components in the molten glass and then cooling the mixture to obtain a glass composite.
2088	In situ chemical characterization of waste sludges using FTIR-based fiber optic sensors	The characterization of unknown mixed wastes is a mandatory step in today's climate of strict environmental regulations.^Cleaning up the nuclear and chemical wastes that have accumulated for 50 years at the Hanford Site is the largest single cleanup task in the United States today.^The wastes are stored temporarily in carbon steel single- and double-shell tanks that are buried in tank farms at the Site.^In the 1950s, a process to scavenge radioactive cesium and other soluble radionuclides in the wastes was developed to create additional tank space for waste storage.^This scavenging process involved treatment of the wastes with alkali cyanoferrates and nickel sulfate to precipitate [sup 137]Cs in the presence of nitrate oxidant.^Recent safety issues have focused on the stability of cyanoferrate-bearing wastes with large quantities of nitrates and nitrites.^Nitrate has been partially converted to nitrite as a result of radiolysis during more than 35 years of storage.^The major safety issue is the possibility of the presence of local hot spots enriched in [sup 137]Cs and [sup 90]Sr that under optimum conditions can self-heat causing dry out and a potential runaway reaction of the cyanoferrates with the nitrates/nitrites).^For waste tank safety, accurate data of the concentration and distribution of cyanoferrates in the tanks are needed.^Because of the extensive sampling required and the highly restricted activities allowed in the tank farms, simulated tank wastes are used to provide an initial basis for identifying and quantifying realistic concerns prior to waste remediation.^Fiber optics provide a tool for the remote and in situ characterization of hazardous and toxic materials.^This study is focused on near-infrared (NIR) and mid-infrared (MIR) fiber optic sensors for in situ chemical characterization of Hanford Site waste sludges.
2090	Stabilization void-fill encapsulation high-efficiency particulate filters	This report discusses high-efficiency particulate air (HEPA) filter systems that which are contaminated with radionuclides are part of the nuclear fuel processing systems conducted by the US Department of Energy (DOE) and require replacement and safe and efficient disposal for plant safety.^Two K-3 HEPA filters were removed from service, placed burial boxes, buried, and safely and efficiently stabilized remotely which reduced radiation exposure to personnel and the environment.

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
2113	Synthesis of a SYNROC ceramic from the melt	The possibility of preparing SYNROC ceramic by fusion in crucibles in a laboratory silite electric furnace and in an induction melter in a cold crucible is studied.^Samples are synthesized with SYNROC-A, SYNROC-B, SYNROC-D, and SYNROC-B compositions containing 10 and 15 mass % liquid oxides and solid intermediate-level wastes (ILW).^The properties of the fused materials not containing ILW are analogous to those of materials obtained by hot pressing.^Adding Na-containing ILW to SYNROC-B lowers its chemical stability.^The phase composition of the fused materials is approximately the same as that of the hot pressed ones.^The effectiveness of preparing SYNROC ceramic by fusion in a cold crucible is demonstrated.
2145	The removal and solidification of radioactive iodide ions using a new inorganic anion exchanger	The present paper discusses a method for removing radioactive iodide ions produced in a nuclear reactor by fixing them onto [alpha] -Bi[sub 5]O[sub 7] I by use of a new anion exchanger.^For the immobilization of radioactive iodide, Bi[sub 5]O[sub 7] I is the most promising compound.^Five kinds of bismuth iodide oxides (BiOI, Bi[sub 7]O[sub 9]I[sub 3], Bi[sub 5]O[sub 7]I, Bi[sub 4]O[sub 5]I[sub 2] and [beta] -Bi[sub 5]O[sub 7]I) have been reported.^Among them, Bi[sub 5]O[sub 7]I is the most stable at high temperature[sub 2,4,5] and it is seven orders of magnitude more stable than BiOI and fourteen orders of magnitude more stable than BiI[sub 3] toward hydrolysis.^The present paper discusses a new method to remove and solidify the radioactive iodide from solution by use of a new inorganic anion exchanger, the composition of which is Bi[sub 5]O[sub 7]X (X is a monovalent anion) and the structure of which is isostructural to Bi[sub 5]O[sub 7]I.^If the new material shows ion exchange with iodide ions, it can be utilized as new material for the removal and the solidification of the radioactive iodide.^In this paper, the synthesis of the new material and its ion exchange properties are reported.^(Author).
2162	Performance characteristics of a glove box inductively coupled plasma mass spectrometer for the analysis of nuclear materials	A commercial inductively coupled plasma mass spectrometer (Elan 250) was modified in order to analyse nuclear materials in a glove box.^The nebulizer, plasma torch and sliding interface are situated inside the glove box while the mass spectrometer and associated electronics are outside.^The sensitivity of the modified instrument is slightly reduced compared with the original owing to a flange that separates the mass spectrometer from the vacuum interface.^This has modified the original distance between the skimmer cone and the ion lens system.^The plasma torch is mounted in a fixed position and the load coil is now separated 25 mm from the tip of the sampling cone.^Optimum plasma operating conditions, stability of the signal and isotopic ratios, levels of oxide and hydroxide polyatomic ions were evaluated in the modified instrument for selected fission products and actinides.^The effect of the ion lens settings on sensitivity and mass discrimination were studied in detail.^Interference effects due to heavy matrix elements (U and Pu) were also studied.^(author).

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
2174	A stable ceramic matrix for fixation of fission products	A ceramic matrix based on [alpha]-Al[sub 2]O[sub 3] (corundum) has been developed for fixation of radioactive wastes.^Transuranium elements are encapsulated in a pure alumina matrix whereas fission products that contain cesium require a mixed matrix consisting of 90 wt% Al[sub 2]O[sub 3] and 10 wt% SiO[sub 2].^The production process utilizes a sol-gel technique supported by seeding with [alpha]-Al[sub 2]O[sub 3] submicron particles.^Thus, a good product homogeneity and a relatively low sintering temperature are accomplished.^The production process consists of simple and conventional steps: Mixing of starting materials and waste, extrusion of the gel, drying, calcination and sintering.^During heat treatment only negligible volatilization is observed and no corrosive melt must be handled.^The waste product shows very low leaching rates.^Weight losses of 3x10[sup -7] g cm[sup -2] d[sup -1] have been measured in a Soxhlet apparatus at 97 C. (orig.)
2181	Cadmium-zinc-telluride for radiation monitoring of gaseous effluent	Cadmium-zinc-telluride (CZT) detectors offer room-temperature operation and improved measurement capability over alternative technologies.^The CZT detectors are similar to cadmium-telluride (CdTe) detectors, but with significant advances.^The use of CdTe has been hindered by technological problems that appear to stem from the commonly used method of growth.^The CZT detectors produced by Aurora Technologies Corporation by high-pressure Bridgman growth are free from these problems, exhibiting superior resolution and excellent stability, reliability, and lifetime, in addition to the advantages of room-temperature operation, high counting efficiency, good energy resolution, small size, low bias voltage requirement, direct conversion of gamma event to charge output, and solid-state durability.^The CZT crystals are grown in sizes up to 10 cm in diameter and up to 10 kg with uniformly high resistivity, providing detectors with leakage currents two orders of magnitude lower than for CdTe.^Consequently, the detectors have improved energy resolution that is superior to scintillators from preamplifier noise levels to well over 100 key and can provide useful energy information at temperatures up to at least 100[degrees]C.
2186	Improving iron-enriched basalt with additions of ZrO[sub 2] and TiO[sub 2]	The iron-enriched basalt (IEB) waste form, developed at the Idaho National Engineering Laboratory a decade ago, was modified to IEB4 by adding sufficient ZrO[sub 2] and TiO[sub 2] to develop crystals of zirconolite upon cooling, in addition to the crystals that normally form in a cooling basalt.^Zirconolite (CaZrTi[sub 2]O[sub 7]) is an extremely leach-resistant mineral with a strong affinity for actinides.^Zirconolite crystals containing uranium and thorium have been found that have endured more than 2 billion years of natural processes.^On this basis, zirconolite was considered to be an ideal host crystal for the actinides contained in transuranic (TRU)-contaminated wastes.^Crystals of zirconolite were developed in laboratory melts of IEB4 that contained 5% each of ZrO[sub 2] and TiO[sub 2] and that were slow-cooled in the 1200--1000[degrees]C range.^When actinide surrogates were added to IEB4, these oxides were incorporated into the crystals of zirconolite rather than precipitating in the residual glass phase.^Zirconolite crystals developed in IEB4 should stabilize and immobilize the dilute TRUs in heterogeneous, buried low-level wastes as effectively as this same phase does in the various formulations of Synroc used for the more concentrated TRUs encountered in high-level wastes.^Synroc requires hot-pressing equipment, while IEB4 precipitates zirconolite from a cooling basaltic melt.

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
2192	Adsorption behavior of cesium and strontium on synthetic zeolite P	Adsorption behavior of Cs and Sr on a synthetic zeolite P has been studied.^Natural zeolites, clinoptilolite and mordenite, were converted into the zeolite P having a high crystallinity through hydrothermal treatment with NaOH solutions.^The distribution coefficient (K[sub d]) for Cs[sup +] and Sr[sup 2+] depended on pH in acidic region, while attained nearly constant values around 10[sup 3] cm[sup 3]/g in neutral and alkaline regions.^This value is almost the same as that for Cs[sup +] and about ten times that for Sr[sup 2+] on the original clinoptilolite.^The adsorption obeys Langmuir-type isotherms, and the saturated amounts of Cs and Sr were respectively estimated to be 1.76 and 1.84 mmol/g.^The zeolite P containing these cations was readily converted into a stable solid form consisting of pollucite and Sr-feldspar by calcination above 1,000degC.^(author).
2206	In situ vitrification: Immobilizing radioactive contaminants in place by melting soils into man-made rocks	From 1951 to 1966 over 1 [times] 10[sup 6] Ci of Cs-137, Sr-90, and other radioisotopes in liquid wastes were disposed of in shallow seepage pits at ORNL.^In situ methods to stabilize these sites are being investigated because of radiation exposure risks to personnel during excavation and removal activities.^A field test at ORNL of In Situ Vitrification (ISV) was performed to evaluate its ability to resistance heating through graphite electrodes to melt contaminated soils in place.^The resulting small lava lake cools and solidifies to a rock consisting of glassy and crystalline material.^Volatile products released from the surface of the melt are collected and treated.^The Sr-90 was incorporated into mineral phases and residual glass that form upon solidification.^The Cs-137, however, is incompatible with the mineral structures and is concentrated into the small amount of residual glass that is trapped in the interstices between mineral grains.^Leach tests were performed on samples of sludge, sludge + soil, crushed ISV rock, crushed ISV rock + soil, and low surface area fragments of ISV rock.^First, sequential extractions with 0.1 N CaCl[sub 2] were used.^Then, sequential treatments with 0.1 N HCl were used.^Approximately 10% of the Sr-90 was released from the sludge, with or without soil, after CaCl[sub 2] was applied.^Subsequent treatment with HCl released essentially all the Sr-90.^The Sr-90 in the crushed ISV rock was resistant to cation exchange, with only 0.4% leached after treatment with CaCl[sub 2].^Treatment with HCl released only 4% of the total Sr-90 present in the crushed ISV rock.^The low surface area fragments, more representative of expected field conditions, released 10 [times] less of the Sr-90 than the crushed ISV rock samples.^The Cs-137 was not significantly leached from any of the samples of sludge or ISV rock.

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
2212	Sorption of cesium and strontum by zeolite single crystals	The aspect ratios of crystals of platey clinoptilolite and fibrous mordenite observed in mineral assemblages coating fractures through tuffs at Yucca Mountain, Nevada, influence the sorption properties of these two zeolites.^The crystallographic dependencies of cation exchange reactions have been demonstrated in clinoptilolite by reacting CsCl with oriented single crystals mounted on (100), (010), (001) and (101) faces.^Competing cation exchange reactions involving Cs[sup +], Sr[sup 2+] and Ba[sup 2+], as well as Cs[sup +] in NaCl or NaHCO[sub 3] solutions, were performed on the oriented zeolite crystals.^Reactions were carried out at 60[degrees]C for 1 to 8 weeks in a shaking water bath with dissolved metal chloride solutions ranging in concentrations from 1M to 10[sup [minus]4]M.^Electron microprobe analyses were performed on the surfaces of the reacted zeolite crystals.^In clinoptilolite, cation exchange is initially retarded on (010) faces which are nominal to the one direction (parallel to the b-axis) along which channels do not exist in the clinoptilolite structure.^This orientation effect was particularly severe for Sr, concentrations of which on (010) faces remained 90% lower than values measured on other crystal faces even when reaction times exceeded 2 months.^In competition with Sr and Ba, the uptake of Cs into clinoptilolite was lowered significantly (and vice versa for Ba and Sr), particularly in the presence of Ba.^The addition of 1M NaCl did not significantly affect the relative concentrations of these competing cations in reacted zeolite crystals.^In NaHCO[sub 3] solutions, however, the Cs uptake was lowered significantly.^Although clinoptilolite has a very high selectivity for Cs[sup +] compared to other cations, competition with Sr[sup 2+] and Ba[sup 2+] reduces the concentration of Cs[sup +] exchanged into this zeolite.^31 refs., 11 figs.
2221	Accelerator-driven transmutation technology for energy production and nuclear waste treatment	New concepts recently developed show that the use of intense particle accelerators affords unique opportunities for electrical power generation, from plentiful fuel such as thorium, with little long term waste.^The concept can also effectively transmute existing actinide and fission product wastes.^The new concept uses the accelerator beam to generate intense flux levels of thermal energy neutrons that efficiently transmute fuels or actinide wastes, and also efficiently transmute fission products to stable or short-lived end-products.^Proton cw accelerators in the 800-1600 MeV, 50-250 mA class are required, depending on the desired plant configuration.^Beam dynamics and optimization issues related to insuring low beam loss along the linac are outlined.^(R.P.) 19 refs.; 1 fig.
2241	Construction of Tokai-Vitrification Facility	The Power Reactor and Nuclear Fuel Development Corporation (PNC) has carried out the research and development of the high-level liquid waste vitrification technology since 1975 in accordance with the Japanese policy for the treatment and disposal of high-level liquid waste (HLLW).^The construction of the Tokai Vitrification Facility (TVF), based on the results of R and D, was started in June 1988, and was completed April 1992.^The purpose of the TVF is to immobilize HLLW stored in the Tokai Reprocessing Plant (TRP) into the stable glass form, and is to demonstrate the HLLW vitrification technology on an industrial scale and the remote control operability of equipment and the maintenance technology.^In this present paper, outline and feature of the TVF and the construction are described.^(author).

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
2256	Modulated structures of a new natural representative of the hollandite series	The authors have refined the modulated structure of Ba, Ti hollandite by means of the JANA specialized program system in the four-dimensional space group P(I4:1) to R = 4.28% from all the reflections and R = 4.15% from the main reflections.^The structure was analyzed from the positions of the microdomain concept.^Three types of domains are distinguished; the different variants of their joining along the c axis are analyzed, together with the defect regions arising at the boundary of two different domains and impairing the sequence of alternation of occupied and vacant subcells.^Compounds with the general formula A[sub 2[minus]x]B[sub 8]O[sub 16] including condition mineral investigated in this article, have recently attracted the attention of investigators, since they are chemical and structural analogs of synthetic hollandites preferentially composing SYNROC rocks regarded as absorbents of radioactive waste.^Since hollandites easily accumulate radioactive ions, in particular Cs, which are concentrated in tunnel structures, and display high stability against changes in the thermodynamic parameters, A. Ringwood proposed using them as matrices for burial of radioactive wastes.^The process of immobilization of radioactive elements is based on introducing them as a solid solution in hollandite of the composition BaAl[sub 2]Ti[sub 6]O[sub 16].^The capacity of hollandites to absorb radioactive elements and withstand leaching is due mainly to their structural characteristics.^12 refs., 3 figs., 5 tabs.
2262	Tracer-level radioactive pilot-scale test of in situ vitrification for the stabilization of contaminated soil sites at ORNL	A field demonstration of in situ vitrification (ISV) was completed in May 1991, and produced approximately 12 Mg of melted earthen materials containing 12.7 mCi of radioactivity within 500 g of sludge in amodel of an old seepage trench waste disposal unit.^Past waste disposal operations at Oak Ridge National Laboratory have left several contaminated seepage sites.^In planning for remediation of such sites, ISV technology has been identified as a leading candidate because of the high risks associated with any retrieval option and because of the usual high quality of vitreous waste form.^Major isotopes placed in the test trench were [sup 137]Cs and [sup 90]Sr, with lesser amounts of [sup 6O]Co, [sup 241]Am, and [sup 239,240]Pu.^A total of 29 MWh of electrical power was delivered to the ground over a 5-day period producing a melt depth of 8.5 ft.^During melting, 2.4% of the [sup 137]Cs volatilized from the melt into an off-gas containment hood and was captured quantitatively on a high efficiency particulate air filter.^No volatilization of [sup 90]Sr, [sup 241]Am, or [sup 239,240]Pu was detected and > 99.993% retention of these isotopes in the melt was estimated.^The use of added rare earth tracers (Ce, La, and Nd), as surrogates for transuranic isotopes, led to estimated melt retentions of >99.9995% during the test.^The molten material, composed of the native soil and dolomitic limestone used for filling the test trench, reached a processing temperature of 1500[degrees]C.^Standardized leaching procedures using Product Consistency Testing indicated that the ISV product has excellent characteristics relative to other vitreous nuclear waste forms.
2278	Chemically bonded phosphate ceramics for radioactive and mixed waste solidification and stabilization	Results of an initial investigation of low temperature setting chemically bonded magnesium ammonium phosphate (MAP) ceramics as waste form materials, for solidification and stabilization of radioactive and mixed waste, are reported.^The suitability of MAP for solidifying and encapsulating waste materials was tested by encapsulating zeolites at loadings up to [approximately]50 wt%.^The resulting composites exhibited very good compressive strength characteristics.^Microstructure studies show that zeolite grains remain unreacted in the matrix.^Potential uses for solidifying and stab wastes are discussed.

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
2305	Small angle X-ray scattering by TiO[sub 2]/ZrO[sub 2] mixed oxide particles and a Synroc precursor	This high resolution small angle X-ray scattering study of a concentrated oxide sol, precursor of the SYNROC matrix for the storage of the high level radioactive waste, evidences a locally cylindrical microstructure.^Locally, nanometric cylinders show disordered axis with some concentration dependent connections.^This microstructure explains the paradoxal stability of this oxide dispersions upon the addition of concentrated acidic solutions.^This stability has a steric origin and electrostatic repulsions are not needed.^The addition of aluminium to the initial titanium-zirconium mixture enhances branching on the locally cylindrical microstructure.^Finally, we show that the solid powder obtained after calcination (drying) of the sol has the same specific area ([approx] 1000 m[sub 2]/g) than the sol.^(Author).^23 refs., 7 figs., 1 tab.
2306	Adsorption of radionuclides on oxide sorbents and impregnated porous membranes under high temperature conditions	The adsorption properties of hydrous titanium and zirconium oxides for Co(II) and other corrosion products have been studied under high temperature and pressure condition.^The studies of dependence of distribution coefficients (K[sub d]) on temperature indicate that K[sub d] decreases with increasing temperature.^The more negative enthalpy values for cobalt sorption at high temperature on oxide sorbents are connected with formation of spinel-type compounds like cobalt metatitanates.^The sorption of radionuclides on oxide sorbents in column processes was studied under high temperature and pressure conditions, similar to those existing in the BWR recirculation loop.^The column filled with TiO[sub 2]aq. worked very effectively as both mechanical and ionic filter.^Due to low mechanical stability of oxide sorbents, hydrous titanium oxide was incorporated into porous stainless steel membrane.^The membranes impregnated with Ti0[sub 2]aq. are very efficient materials for sorption of radionuclides from aqueous solution, and can be used for removal of radioactive corrosion products in RWCU.^(author).^11 refs, 6 figs, 10 tabs.
2330	Radiation-stimulated drift and escape of radionuclides from glassy matrices	In evaluating the effectiveness of materials for solidifying radioactive wastes the most important indicator is the rate at which radionuclides escape from the matrix by ground water and leaching is taken into account.^The normalized rate of escape from modern matrices (borosilicates, phosphate glasses, the material SYNROC, etc.), measured under laboratory conditions, equals 10[sup [minus]7]-10[sup [minus]5] g[center dot]cm[sup [minus]2][center dot]days[sup [minus]1].^Storage locations and methods that would reduce to a minimum the possibility of contact between the matrix and groundwater are being sought.^There arises the question of the rate of escape of radionuclides from the matrix under conditions of dry storage.^The paper describes a phenomenon that can lead to the accumulation of radionuclides at the interface between the matrix and the metal casing.^When the matrix breaks down owing to corrosion or for some other reason the radionuclides accumulated on the surface will enter the environment directly from the interface.^The value determined by an equation derived here establishes some lower limit for the rate of escape of radionuclides from the matrix under any storage conditions.^This limit cannot be lowered merely by increasing the chemical stability of the matrix with respect to water.
2333	Microbiological treatment of low level radioactive waste	This report summarises the work of an experimental programme investigating the anaerobic digestion of low-level radioactive wastes.^The project focused on the selection of the optimum bioreactor design to achieve 95% removal or stabilisation of the biodegradable portion of low-level radioactive wastes.^Performance data was obtained for the bioreactors and process scale-up factors for the construction of a full-scale reactor were considered.^(author).

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
2334	Incineration of [sup 90]Sr and [sup 137]Cs by an inertial fusion target	We discuss the inertial confinement fusion system as a transmutator of radioactive waste ([sup 90]Sr and [sup 137]Cs).^An analytic model of the implosion of the target, which is composed of DT fuel and radioactive waste, is used to evaluate its internal energy and the probability of neutron utilization.^From the results of this calculation, we could evaluate the energy that is required to transmute radioactive waste.^(orig.).
2359	Underground tank vitrification: Field-scale experiments and computational analysis	In situ vitrification (ISV) is a thermal waste remediation process developed by researchers at Pacific Northwest Laboratory for stabilization and treatment of soils contaminated with hazardous, radioactive, or mixed wastes.^Many underground tanks containing radioactive and hazardous chemical wastes at U.S.^Department of Energy sites will soon require remediation.^Recent development activities have been pursued to determine if the ISV process is applicable to underground storage tanks.^As envisioned, ISV will convert the tank, tank contents, and associated contaminated soil to a glass and crystalline block.^Development activities include testing and demonstration on three scales and computational modeling and evaluation.^In this paper, the authors describe engineering solutions implemented on the field scale to mitigate unique problems posed by ISV of a confined underground structure along with the associated computational analysis.^The ISV process, as applied to underground storage tanks, is depicted.^The process is similar to ISV of contaminated soils except the tank also melts and forms a metal ingot at the bottom of the melt.
2360	In situ vitrification of a simulated seepage trench: A radioactive field test at ORNL	The pits and trenches used at Oak Ridge National Laboratory (ORNL) from 1951 to 1966 to dispose of over a million curies of radioactive liquid wastes are currently undergoing remedial investigations/feasibility studies to identify potential technologies for cleanup and/or stabilization.^In situ vitrification (ISV) is a leading technology candidate because of the high risks associated with options requiring retrieval and because of the high-quality waste form produced by ISV.^The radioactive field test conducted on a simulated ORNL seepage trench in May 1991 is the second step in evaluating ISV as a remedial action at these sites.^A previous test using nonradioactive tracers for cesium and strontium was completed in 1987.

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
2376	Research and development of waste forms for geological disposal	Ceramics are candidate materials for immobilizing high-level waste (HLW) stemming from the reprocessing of spent fuels.^We are proceeding with R and D on two types of ceramic waste form : a polyphase titanate ceramic named Synroc and three kinds of single-phase zirconium ceramics.^The effect of self-irradiation damage on the long-term integrity of Synroc due to alpha decay was studied under a cooperative program between JAERI and ANSTO.^The hot-pressed polyphase titanate ceramic (10 wt% waste loading) was doped with {sup 244}Cm to accumulate a dose of 1.6 x 10{sup 18} alpha decays/g.^The phase assemblage of the curium-doped titanate ceramic included freudenbergite and loveringite in addition to three main phases: hollandite, perovskite and zirconolite.^Accumulation of alpha decays was accompanied by a gradual decrease in density.^The change in density was -2.7 % after an equivalent age of 45000 years.^The durability of three single-phase zirconium ceramics which contained the appropriate amount of simulated high-level waste elements was examined at 90degC and 150degC in hydrochloric acid or deionized water.^The waste forms examined included 10 mol% Y{sub 2}O{sub 3}-stabilized ZrO{sub 2}, La{sub 2}Zr{sub 2}O{sub 7} with a pyrochlore structure, and CaZrO{sub 3} with a perovskite structure.^La{sub 2}Zr{sub 2}O{sub 7} showed excellent durability, and leach rates of all constituents were less than about 10{sup -4} g{center dot}m{sup -2}{center dot}day{sup -1} at 150degC in deionized water.^This suggests that La{sub 2}Zr{sub 2}O{sub 7} is a promising candidate material for immobilization of waste elements from HLW.^(J.P.N.).
2379	Separation of technetium from ruthenium after the accelerator transmutation of technetium	Both civilian and defense related waste must be processed with a strategy for dealing with Tc.^One solution is to remove the Tc from the waste steam and transmute the Tc to stable Ru in either a reactor or an accelerator.^Before any processing of waste streams can be performed (even if transmutation is not performed) the separations chemistry from the spent fuels or the stored wastes containing Tc must be developed.^This report details some of the separation schemes possible for the separation of Tc and Ru, which include the baseline ion exchange process of Roberts, Smith and Wheelwright, ozonolysis, filtration, magnetic separation, solvent extraction, electrodeposition, fluorination, and pyrolysis.^5 figs, 4 refs.^(DLC)
2384	Remote excavation using the telerobotic small emplacement excavator	For nearly five decades the United State Department of Energy (DOE) and its predecessor agencies have engaged in broad-based research and development activities as well as in the production of nuclear weapons components.^As a by-product of these activities, large quantities of waste materials have been generated.^With the current emphasis on environmental restoration, DOE now plans to either to retrieve much of the legacy of buried waste or to stabilize the waste in place via in situ vitrification or by other means.^Because of the potential hazards of these wastes and uncertainty about waste contents and container integrity, it is highly desirable to excavate these wastes using remotely operated equipment.^This paper describes the development of a teleoperated military tractor called the Small Emplacement Excavator (SEE).^2 refs.

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
2386	Underground tank vitrification: Field scale experiments and computational analysis	In situ vitrification (ISV) is a thermal waste remediation process developed by researchers at Pacific Northwest Laboratory (PNL) for stabilization and treatment of soils contaminated with hazardous, radioactive or mixed wastes.^Many underground tanks containing radioactive and hazardous chemical wastes at US Department of Energy (DOE) sites will soon require remediation.^Recent development activities have been pursued to determine if the ISV process is applicable to underground storage tanks.^As envisioned, ISV will convert the tank, tank contents. and associated contaminated soil to a glass and crystalline block.^Development activities include testing and demonstration on three scales and computational modeling and evaluation.^A description of engineering solutions implemented on the field scale to mitigate unique problems posed by ISV of a confined underground structure, along with the associated computational analysis, is given in the paper.
2388	Sampling methods and non-destructive examination techniques for large radioactive waste packages	Progress is reported on work undertaken to evaluate quality checking methods for radioactive wastes.^A sampling rig was designed, fabricated and used to develop techniques for the destructive sampling of cemented simulant waste using remotely operated equipment.^An engineered system for the containment of cooling water was designed and manufactured and successfully demonstrated with the drum and coring equipment mounted in both vertical and horizontal orientations.^The preferred in-cell orientation was found to be with the drum and coring machinery mounted in a horizontal position.^Small powdered samples can be taken from cemented homogeneous waste cores using a hollow drill/vacuum section technique with the preferred subsampling technique being to discard the outer 10 mm layer to obtain a representative sample of the cement core.^Cement blends can be dissolved using fusion techniques and the resulting solutions are stable to gelling for periods in excess of one year.^Although hydrochloric acid and nitric acid are promising solvents for dissolution of cement blends, the resultant solutions tend to form silicic acid gels.^An estimate of the beta-emitter content of cemented waste packages can be obtained by a combination of non-destructive and destructive techniques.^The errors will probably be in excess of +/-60 % at the 95 % confidence level.^Real-time X-ray video-imaging techniques have been used to analyse drums of uncompressed, hand-compressed, in-drum compacted and high-force compacted (i.e. supercompacted) simulant waste.^The results have confirmed the applicability of this technique for NDT of low-level waste.^8 refs.,,12 figs.; 3 tabs.
2393	High-power proton linac for transmuting the long-lived products in nuclear waste	A high power proton linac is being considered at Los Alamos as a driver for a high-flux spallation neutron source capable of transmuting the troublesome long-lived fission products in defense nuclear waste.^The transmutation scheme under study provides a high flux (> 10{sup 16}/cm{sup 2}-s) of thermal neutrons which efficiently converts fission products to stable or short-lived isotopes.^A transmuter based on a medium-energy proton linac generating 110 MW of beam power could burn the accumulated {sup 99}Tc and {sup 129}I inventory at the DOE's Hanford site within 30 years.^Preliminary concepts for the accelerator are discussed.

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
2400	Operational experience and development of hollow fiber filter for BWRs in Japan	After four years stable operation in 1F-3, it is confirmed that HFF has excellent treatment performance.^Moreover, HFF application to the condensate polishing system contributes to reducing the impurity load to the deep bed condensate demineralizer.^It is effective for longer operation, with no chemical regeneration of the condensate demineralizer, unless there sea water leakage occurs at the main condenser cooling tube.^It is obvious that the HFF condensate filter generates no secondary waste, like spent powdered resin from the precoat filter, so the radwaste system does not need large tanks for the waste storage.^This can realize a more compact and economical radwaste system.^HFF does not need a higher regeneration operation frequency then the conventional precoat filter.^This markedly contributes to reducing the operator's work load.^(author).
2402	Contaminated soil stabilization demonstration	Long-term herbicide control along with a shotcrete cover was constructed at the Hanford Site in May 1991.^The cover system allows for maintenance-free containment of contaminants by preventing wind and water transport of contaminants from the soil surface, preventing plant uptake of contaminants, and minimizing water infiltration through the soil column.^The cover is composed of two parts: a commercial nonwoven geotextile material impregnated with trifluralin, and a >5-centimeter top cover of shotcrete containing polyethylene fibers.^The herbicide-impregnated geotextile functions to prevent plant root growth into contaminated soil if any holes or cracks develop in the shotcrete layer.^The herbicide component, trifluralin, is mixed into polymer nodules that degrade slowly over many years, thus releasing trifluralin slowly over time.^The shotcrete topcover was sprayed using a sludge pump and air compressor to form a hard, impenetrable surface that prevents wind erosion and reduces water infiltration through the contaminated materials underneath.^The benefits of the cover system are expected to last 20 to 30 years.^2 refs., 4 figs.
2409	Predictions of PuO sub 2 and tracer compound release from ISV melts	Two field tests were conducted at the Idaho National Engineering Laboratory (INEL) to assess in situ vitrification (ISV) suitability for long-term stabilization of buried radioactive waste.^Both tests contained rare-earth oxide tracers (DY{sub 2}O{sub 3}, Yb{sub 2}O{sub 3}, and Tb{sub 4}O{sub 7}) to simulate the presence of plutonium in the form of PuO{sub 2}.^In the first test, Intermediate Field Test (IFT)-l, approximately 4-% release of tracer material occurred during soil melting and associated off-gassing, while essentially nil release was observed for the second experiment (IFT-2) for which off-gassing was much reduced.^This report presents an evaluation of the IFT test data in terms of governing release processes.^Prediction of tracer release during ISV melting centered on an assessment of three potential transport mechanisms, (a) tracer diffusion through stagnant pool, (b) tracer transport by convective currents, and (c) tracer carry-off by escaping gas bubbles.^Analysis indicates that tracer release by escaping gas is the dominant release mechanism, which is consistent with video records of gas bubble escape from the ISV melt surface.^Quantitative mass transport predictions were also made for the IFT-I test conditions, indicating similarity between the 4-% release data and calculational results at viscosities of {approx} poise and tracer diffusivities of {approx}10{sub {minus}6} CM{sup 2}/s.^Since PuO{sub 2} has similar chemical and transport (diffusivity) properties as the rare-earth tracers used in the rare earth tracers used in the IFT experiments, release of PuO{sub 2} is predicted for similar off-gassing conditions.^Reduced off-gassing during ISV would thus be expected to improve the overall retention of heavy-oxides within vitrified soil.

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
2411	Durability of glasses from Pacific Northwest Laboratory Composition Variability Study-II (CVS-II)	Pacific Northwest Laboratory (PNL) is developing a borosilicate glass as a solid, stable medium for the disposal of high-level radioactive waste at the Hanford site.^They are optimizing the glass forming region and developing process models to be used in the Hanford Waste Verification Project (HWVP).^Their experimentally-based statistical approach for optimizing the glass composition for HWVP has been designated the Composition Variability Study (CVS).^In Part 1 of the CVS study PNL tested wide ranges of composition developed first-order empirical models, and provided input for planning CVS-2.^In part 2, they are generating glass property data for a number of compositions in order to develop second-order empirical models which will be used to identify the composition region that simultaneously satisfies all quality and processability requirements of HWVP.
2412	Radioactive waste treatment using cement glass solidification technique	A new radioactive waste treatment system has been developed from the viewpoint of minimizing the volume and then solidifying the waste into a stable form.^Emphasis has been placed on long term stability in the final disposal environment by solidification using a new inorganic agent (cement glass).^The primary components of the cement glass are sodium silicate and phosphorous silicate.^The mixing of cement glass powder and water initiates a polymerization reaction by silicates, resulting in a three-dimensional structure of O-Si-O.^The paper describes the solidification characteristics and waste from as well as the system outline.^Application to the waste from a fuel reprocessing plant is also discussed.^(J.P.N.).
2413	Irradiation test of sup 99 Tc and sup 129 I transmutation in the Fast Flux Test Facility	An attractive option for dealing with the problems of nuclear waste disposal includes reprocessing spent light water reactor fuel to recover and recycle the uranium and plutonium, partitioning key long-lived actinides and fission products, and transmuting recovered and purified very long-lived problem isotopes.^Most transmutation studies have dealt with the minor actinides; however, a successful transmutation strategy also must address the long-lived fission products {sup 99}Tc and {sup 129}I.^Destruction of {sup 99}Tc and {sup 129}I is accomplished by a single neutron-capture event, followed by very rapid decay to stable {sup 100}Ru and {sup 130}Xe, respectively.^The probability of a neutron-capture event is significantly higher in a moderated neutron spectrum than in a fast spectrum.^Studies have shown that effective transmutation rates of {sup 99}Tc and {sup 129}I potentially can be achieved in specially designed metal hydride assemblies in fast reactors or advanced accelerator-driven devices.^A successful transmutation experiment for the key long-lived fission products {sup 99}Tc and {sup 129}I was performed using a metal-hydride-moderated environment in the radial reflector region of a sodium-cooled fast reactor with reasonably good agreement between measured and calculated transmutation rates.^The underprediction of both transmutation rates is likely a result of axial gradients in the low-energy neutron flux over the target regions.
2433	Studies of high-level radioactive waste form performance at Japan Atomic Energy Research Institute	The recent studies of high-level radioactive waste form at Japan Atomic Energy Research Institute can be classified into the following three categories; (1) Study on the leaching behavior of the nuclear waste glass placing the focus on the alteration layer and the chemical composition of leachant for the prediction of the long-term corrosion of the waste glass.^(2) Study on the radiation (alpha-radiation) effects which have relation to the long-term stability of the nuclear waste glass.^(3) Study on the long-term self-irradiation damage of a SYNROC waste form using a curium-doped sample.^In the present report, the recent results corresponding to the above categories are described.^(author).

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
2441	Making steady progress with Synroc research	In Australia, much research effort is being put into developing and testing Synroc.^Synroc is an advanced ceramic made up of titanate minerals chosen for their geochemical stability and their collective ability to immobilize radioactive elements present in the high-level waste from fuel reprocessing.^It acts much as natural rock has done to immobilize radioactive elements over millions of years in a wide range of geological settings in the earth.^(author).
2443	Separation of rare gases from radioactive contaminants by means of an ultracentrifuge	A sample of xenon enriched in {sup 136}Xe has been obtained with the ultracentrifuge system of the United States Department of Energy in Oak Ridge, Tennessee.^The same system has also been used for cleanup only, namely to eliminate undesired gas impurities.^Measurements of radioactivity carried out with a multicell proportional chamber show that ultracentrifugation is indeed very effective in the elimination of krypton.^(orig.).
2449	Evaluation of aluminium-silicate ceramics as a matrix for the solidification of radioactive wastes from the reprocessing and fabrication of nuclear fuels	The immobilization of nuclear waste by embedding SOURCESto an aluminum-silicate based ceramic matrix has been studied by using real TRU-wastes, as well as simulated high-level and TRU-waste mixtures from the nuclear fuel cycle.^The aluminum-silicate ceramic matrix is less sensitive to the chemical composition of the waste, than other waste forms such as glass or Synroc.^Therefore, a wide spectrum of waste streams with loadings of > 20 wt.% can be immobilized.^Mixtures of 34 wt.% clay minerals (kaolinite, bentonite) and 66 wt.% reactive corundum were selected as raw materials.^Waste streams investigated were: 1. dissolver residues, 2. ashes from the dry incineration of burnable TRU-wastes, 3. TRU-slurries, precipitated from different TRU-bearing waste solutions, 4. slurries, precipitated from medium level waste solutions, dissolver residues, dry incinerator ashes and denitrated high-level waste.^Fixation of radionuclides is achieved either by inclusion into the microstructure of the aluminum-silicate or oxide matrix phases or by incorporation into the crystal structure of host phases, depending each on the chemical composition of the waste and the matrix raw materials.^Leach tests were performed with TRU-waste ceramics, according to the IAEA-test, by using a MgCl{sub 2}-rich salt brine.^Leach rates were found to range between 5 E-3 and 1 E-4 g/m{sup 2}d at room temperature after a duration of about one year of total leaching time.^The stability of the ceramics against {alpha}-irradiation was studied by doping it with 20 wt.% of Pu(238)O{sub 2}.^After three years an accumulated {alpha}-dose of 8,33 E 9 Gy (1 E 19 {alpha}-decays/g ceramic) was reached, which exceeds the maximum accumulated {alpha}-dose of the ceramics under investigation after 100 000 years.^The lattice constants of the matrix phases mullite and corundum were found to be only slightly enlarged (<2%).^There were no signs of metamictization and no changes in the microstructure.^(orig./BBR).

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
2472	Physical and chemical properties of the products of in situ vitrification engineering tests 5, 6, and 7	In situ vitrification (ISV) is an in situ thermal treatment process that is being investigated by the Idaho National Engineering Laboratory (INEL) for application to buried waste sites.^ISV is a thermal treatment process that converts contaminated soil into a chemically inert and stable glass and crystalline product.^The INEL is evaluating whether the treatment process is a viable one for remediating a buried mixed transuranic waste site at the INEL Subsurface Disposal Area (SDA).^The SDA is a Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) site.^As part of the INEL investigation, a series of tests have been performed that address issues associated with vitrification of buried waste.^Two pilot ISV tests and four tests at laboratory scale, formerly called engineering scale, were performed in 1990 to support the INEL investigation.^The chemical composition and leaching of the produce glass is described.
2481	Tracer-level radioactive pilot-scale test of in situ vitrification technology for the stabilization of contaminated soil sites at ORNL	This plan summarizes the activities to be performed during FY 1990 and FY 1991 for the tracer-level radioactive pilot-scale in situ vitrification (ISV) test.^This test is the second step in evaluating ISV as a remedial action for the pits and trenches at Oak Ridge National Laboratory (ORNL).^A previous test used nonradioactive tracers for cesium and strontium.^This new test will again use a one-half-scale model of trench 7 and the pilot-scale ISV equipment of Pacific Northwest Laboratory (PNL).^A small and precisely known amount of waste from a liquid waste disposal pit will be used for the test.^An actually contaminated waste site cannot be used for this test because of the necessity to use an exactly known inventory of radionuclides so that a precise measurement of the volatilization of various constituents to the off-gas can be determined.
2482	Properties and behavior of the platinum group metals in the glass resulting from the vitrification of simulated nuclear fuel reprocessing waste	Two types of platinum group metal particles were found in borosilicate nuclear waste glasses: needle-shaped RuO{sub 2} particles and spherical PdRh{sub {ital x}}Te{sub {ital y}} alloys.^They form a dense sediment of high electrical conductivity and relatively high viscosity at the bottom of the ceramic melting furnace.^The sludge shows a non-Newtonian flow behavior.^The viscosity and conductivity of the sludge depend not only on the platinum group metal content but also on the texture and morphology of the RuO{sub 2} particles.^RuO{sub 2} forms long, needle-shaped crystals which are caused by alkalimolybdate salt melts that formed in the calcine layer.^The salt melts oxidize the Ru present as small RuO{sub 2} particles after calcination to higher oxidation states.^Ruthenium (VI) compounds are formed, presumably, which are not stable with respect to RuO{sub 2} under the melting conditions.^RuO{sub 2} precipitates and crystallizes into long, needle-like particles.
2489	In situ vitrification of buried waste: Containment issues and suppression systems	Pacific Northwest Laboratory (PNL) and Idaho National Engineering Laboratory (INEL) are developing a remedial action technology for buried waste through the adaptation of the in situ vitrification (ISV) process.^The ISV process is a thermal treatment process originally developed for the US Department of Energy (DOE) to stabilize soils contaminated with transuranic waste.^ISV tests with buried waste forms have demonstrated that the processing of buried waste is more dynamic than the processing of soils.^This paper will focus on the issue of containment of the gases released during the processing of buried waste and on engineered suppression systems to alleviate transient events associated with dynamic off-gassing from the ISV melt.

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
2499	ISV of a simulated seepage trench: A Radioactive Field Test at ORNL	The pits and trenches used at Oak Ridge National Laboratory (ORNL) from 1951 through 1966 to dispose of over a million curies of radioactive liquid wastes are currently undergoing remedial investigations/feasibility studies to identify potential technologies for cleanup and/or stabilization.^In situ vitrification (ISV) is a leading technology candidate because of the high risks associated with options requiring retrieval, and because of the high-quality waste form produced by ISV.^The Radioactive Field Test, conducted on a simulated ORNL seepage trench, in May 1991, is the second step in evaluating ISV as a remedial action at these sites.^This document discusses this field test.
2518	The natural zeolite, laumontite, as a potential material for the treatment of aqueous nuclear wastes	A natural laumontite from the Isle of Skye, Scotland was examined as a candidate material for aqueous nuclear waste treatment, and its fully Ca exchanged form was shown to be Sr selective.^Laumontite has a good pH stability in acid and alkaline media.^The materials used were characterized by wet chemical analysis, XRD and thermal analysis.^The studies include both ion exchange kinetics and equilibrium isotherm studies which tend to confirm simple K{sub d} tests.^(author) 12 refs.; 2 figs.; 3 tabs.
2533	Solidification of high-level radioactive waste	High-level radioactive wastes include concentrated fission product solutions produced by reprocessing nuclear fuel.^In addition to fission products, these solutions also contain long half-life transuranic nuclides requiring stringent surveillance.^For this reason, it is widely accepted today that solidification of these liquids into a stable form presents major advantages from a safety standpoint.^Considerable research has been undertaken in this area since the early 1950s to develop a solid material with suitable properties, principally ease of industrial fabrication, compatibility with site and environmental safety standards, and material behaviour for interim storage and especially long-term disposal.^The long-term material behaviour is assessed by quantifying several phenomena, notably the effects of alpha self-irradiation, structural modifications, leach rates for matrix elements and radionuclides.^Alteration mechanisms affecting the surface of the solid material and liable to occur in a geological repository are also investigated with attention to local parameters including pressure, temperature and the underground water composition.^A wide range of solid materials has been investigated, notably calcinates, ceramics, glasses and composites.^Special attention has been given to tailored ceramics and to "SYNROC", a synthesis titanium oxide-base ceramic.^Among glasses, borosilicate formulations have been the most extensively developed.^The solidification of high-level liquid waste is now an industrial reality.^A German demonstration plant has been operating at Mol (Belgium) since 1985, and two plants have been in operation in France since 1978 (Marcoule) and 1989 (La Hague).^Another unit is scheduled to go on stream at Sellafield (UK) before the end of 1990, and three additional facilities are now under construction: two are nearing completion in USA (Savannah and West Valley) and work recently began on a third at Rokashomura (Japan).^(orig.).

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
2537	Crystallization of plutonium-uranyl nitrate solution	LOTUS is the advanced fuel reprocessing concept based on the utilization of the low temperature technologies such as freeze-drying, vacuum distillation and crystallization.^The regeneration of solvent with salt-free and the treatment of radioactive solution with high decontamination factors will be achieved by the application of these technologies, resulting in the minimization of waste generation, reduction of radioactivity released to the environment, stable operation for a long period and cost reduction.^In the present work, the behavior of uranyl-nitrate solution, plutonium-nitrate solution and plutonium-uranyl-nitrate solution at the low temperature up to-60degC was observed and resulted in the probability to separate uranium and plutonium in the plutonium-uranyl-nitrate solution and to concentrate plutonium-nitrate solution by means of crystallization and freeze-drying.^(author).
2568	Applications of inductively coupled plasma - Mass spectrometry to the analysis of heavy metal elements in environmental and radioactive samples at the Savannah River Site	An ICP-MS spectrometer at the Savannah River Laboratory has been adapted to achieve a high sensitivity for both stable and radioactive heavy metals in a variety of samples from pristine and contaminated areas, glass leaching studies for nuclear waste isolation, and isotopic analyses of separation process samples.^ICP-MS can achieve detection limits of less than 100 pptr (parts per trillion) while performing multi-element analyses.^Examples of results from samples containing cesium, rare earths, mercury, lead, and actinide elements will be presented.
2572	Immobilization of sodium and potassium in Synroc	Substitution of K for Na in certain nuclear fuel reprocessing cycles may allow an increase of high level radioactive waste loading in Synroc, because K can be incorporated in the barium hollandite phase more easily than Na.^The use of rare-earth additions to stabilize Na in the perovskite phase may also have merit.^(author).
2588	Potential for using a six-phase alternating current power supply system for in situ vitrification	In situ vitrification (ISV) has been identified as a potential treatment technology for stabilizing underground tanks at Hanford and other US Department of Energy (DOE) sites.^A key requirement for this application is an electrical system that can supply the power needed to vitrify a tank in a single setting.^This paper describes an engineering-scale test conducted at the Pacific Northwest Laboratory (PNL) to assess the efficiency of a six-electrode, six-phase energy supply system in melting soil.^The test was conducted with a 30-kW six-phase system.^Based on the test results, a six-electrode, six-phase system shows potential for scaleup to larger systems.^5 refs., 5 figs., 2 tabs.
2604	High flux particle bed reactor systems for rapid transmutation of actinides and long lived fission products	The technology to reprocess nuclear waste has evolved over the years through the PUREX/TRUEX/CURE processes to a level where partitioning into relatively pure streams" of Pu, actinides, and fission products, respectively, is feasible.^If the resulting Pu and actinides could then be transmuted into stable (or short-lived) isotopes, the remaining waste product would have an ingestion toxicity below that of uranium ore after {approximately}300 years.^Therefore, burning/transmutation of actinides and other long lived fission products (LLFP) would have a significant beneficial impact on the waste disposal problems.^An initial assessment of a number of actinide/LLFP burner concepts based on the Particle Bed Reactor (PBR) will be described.^3 refs., 1 tab.

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
2613	Conversion of three mixed-waste streams	At the present time, commercial mixed waste (containing both radioactive and hazardous components) is not handled by any disposal site in this country.^Thus, a generator of such material is faced with the prospect of separating or altering the nature of the waste components.^A chemical or physical separation may be possible.^However, if separation fails there remains the opportunity of chemically transforming the hazardous ingredients to non-hazardous substances, allowing disposal at an existing radioactive burial site.^Finally, chemical or physical stabilization can be used as a tool to achieve an acceptable waste form lacking the characteristics of mixed waste.^A practical application of these principles has been made in the case of certain mixed waste streams at Aerojet Ordnance Tennessee.^Three different streams were involved: (1) lead and lead oxide contaminated with uranium, (2) mixed chloride salts including barium chloride, contaminated with uranium, and (3) bricks impregnated with the barium salt mixture.^This paper summarizes the approach of this mixed-waste problem, the laboratory solutions found, and the intended field remediations to be followed.^Mixture (1), above, was successfully converted to a vitreous insoluble form.^Mixture (2) was separated into radioactive and non-radioactive streams, and the hazardous characteristics of the latter altered chemically.^Mixture (3) was treated to an extraction process, after which the extractant could be treated by the methods of Mixture (2).^Field application of these methods is scheduled in the near future.
2614	Aqueous dissolution of laboratory and field samples from the in-situ vitrification process	In-situ vitrification (ISV) is being evaluated in several countries as a remediation technology for immobilizing both hazardous and radioactive buried wastes.^A combination of laboratory data and modeling results are presented that establishes the scientific basis for predicting the long-term stability of an ISV glass in the environment.^Laboratory experiments included tests with ISV samples obtained from pilot- and intermediate-scale field tests, a nuclear waste glass, and a natural obsidian.^8 refs.
2620	Process for direct conversion of reactive metals to glass	This document discovers a method of handling highly radioactive alkali metal that is low in cost, relatively simple to control, easily adapted to continuous production that produces a product in a stable disposable form and at the same time provides a minimum release of radioactivity.^Radioactive alkali metal is introduced into a cyclone reactor in droplet form by an aspirating gas.^In the cyclone metal reactor the aspirated alkali metal is contacted with silica powder introduced in an air stream to form in one step a glass.^The sides of the cyclone reactor are preheated to ensure that the initial glass formed coats the side of the reactor forming a protective coating against the reactants which are maintained in excess of 1000{degree}C to ensure the formation of glass in a single step.^1 fig.

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
2622	The DIMEX experiment ^Results and analysis	The DIMEX (dipole plasma microwave exposure) experiment tests the feasibility of using a magnetic dipole confined plasma as a plasma cloak.^A plasma is magnetically confined about a small permanent magnet that is sheathed in stainless steel to produce stabilization of MHD instabilities.^The plasma is formed by firing a low-voltage electron beam down the magnetic field lines to the magnet pole, using argon gas as the chamber fill gas.^A plasma shell is observed to form around the magnet, having a plasma free region near the magnet, an abrupt rise to peak density just outside of the plasma free region, and a slow, monotonically decreasing density profile outward to the chamber walls from the region of peak density.^Peak electron densities of 10{sup 11}/cc at electron temperatures of 1 eV, have been measured.^Following beam shutoff, a quiescent period of plasma confinement, characterized by very little diffusion, is observed, lasting 1.5 msec.^This period of low diffusion is followed by a period of Bohm-like diffusion with density decay times of 2 msec.^Microwaves at 1 GHz are strongly absorbed at intensities up to {approximately}10W/cm{sup 2}.^At intensities above this value strong reflection begins.^Analyses of DIMEX experimental results are presented along with the latest experimental data.
2628	Hulls and structural material waste conditioning by high pressure compaction	Since 1986 KfK is developing a conditioning process.^Main subjects of the investigations were the development of the production technique and the planning of the most important equipments of the process under remote conditions.^The process is based on an extensive program of experiments.^Inactive bulks of hulls and structural material components were compacted using maximum axial pressure load of about 300 MPa.^The product density as function of press force was experimentally determinated.^The mechanical loads of the press and tools were estimated for the design of these equipments.^The hydraulic press consists a horizontal four-cylinder press.^The maximum force of the press is 25 MN.^The main advantage is the modular design of the press which is open on all sides.^Especially the free accessibility from top is ensured.^The report also represents relevant radiological data of the alternative product.^Co-60 is the dominating activity of the product due to the effects of the heat production.^An amount of 10 kg hull waste or 25 kg top and bottom pieces of the spent fuel assemblies per package is already beyond the Co-60 limit of the KONRAD regulations.^The nuclear thermal power of a filled container is approximately sixty times lower compared with a vitrified HLW-container.^Since the product shows thermal stability beyond 200{sup 0}C, this it is suited for a combined disposal together with vitrified HLW-containers in salt bore holes of a geological disposal.^The preliminary cost evaluation is based on a reprocessing throughput of 500 t{sub HM} per year and volume reduction factor of 5.3.^Accordingly there are produced 300 waste packages with hulls only or 625 units with hulls and top and bottom pieces which require 1.6 or 2.3 millions DM respectively.

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
2639	The SYNROC studies at the Japan Atomic Energy Research Institute	Three kinds of oxide-route SYNROC samples, which were prepared with three different methods of hot uniaxial pressing, hot isostatic pressing, and air sintering, examined with emphasis on microstructures and short-term leaching mechanisms.^Effects fo precursor materials and fabrication process on MCC-1 short-term leach test were investigated.^In the test at 90 deg.^C for 7 days, the normalized leach rates of Na, Cs and Ca from the hydryde-route SYNROC were similar to those from the SYNROC prepared by the hydroxide-route process.^This implies that the amount of the glass phase was very small in both SYNROCS.^An accelerated alpha radiation stability test has been initiated our using Cm-244.^Normalized leach rates of {sup 244}Cm from Cm-doped SYNROC did not vary significantly with time and averaged 1,1.10{sup -3} g.m{sup -2}.d{sup -1}.^The well-type NaI (T1) scintillation spectrometer was found to be convenient for quantitative analysis of {sup 244}Cm leached form Cm-doped SYNROC.^(author).^13 refs, 13 figs, 10 tabs.
2643	In situ vitrification: Process and products	In situ vitrification (ISV) is an electrically powered thermal treatment process that converts soil into a chemically inert and stable glass and crystalline product.^It is similar in concept to bringing a simplified glass manufacturing process to a site and operating it in the ground, using the soil as a glass feed stock.^Gaseous emissions are contained, scrubbed, and filtered.^When the process is completed, the molten volume cools producing a block of glass and crystalline material that resembles natural obsidian commingled with crystalline phases.^The product passes US Environmental Protection Agency (EPA) leach resistance tests, and it can be classified as nonhazardous from a chemical hazard perspective.^ISV was developed by the Pacific Northwest Laboratory (PNL) for the US Department of Energy (DOE) for application to contaminated soils.^It is also being adapted for applications to buried waste, underground tanks, and liquid seepage sites.^ISV's then-year development period has included tests on many different site conditions.^As of January 1991 there have been 74 tests using PNL's ISV equipment; these tests have ranged from technology development tests using nonhazardous conditions to hazardous and radioactive tests.^2 refs., 6 figs., 7 tabs.

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
2645	Development of an in-line grout meter for improved quality control: January 1990 progress report	Stabilization/solidification (S/S) technology is the most widely used technique for the treatment and ultimate disposal of both radioactive and chemically hazardous waste.^Cement-based products, commonly referred to as grouts, are the predominate materials of choice due to their associated low processing costs, compatibility with a wide variety of disposal scenarios, and ability to meet stringent processing and performance requirements.^It has long been recognized that there is a need for a monitor to use with freshly prepared grouts that would facilitate improved quality control.^In the past, efforts in this area have not proven successful due to the fact that the freshly prepared grout tended to cake on the monitor probe, thus greatly reducing its effectiveness.^This report documents progress to date on efforts at Oak Ridge National Laboratory (ORNL) in support of the Westinghouse Hanford Company Grout Technology Program to develop an in-line monitor for mix ratio verification and, hence, improved quality control in the Westinghouse Hanford Grout Treatment Facility (GTF).^Data have been presented which show that the electrical resistance of freshly prepared grouts is linear with respect to grout mix ratio over a wide range of values.^The data serve to establish the merits for application to the Westinghouse Hanford GTF.^The data further establish that special care must be maintained during design and installation to ensure that the electrodes are properly sealed.^The data also indicate that a separate meter will be required in order to assess waste-feed variations.^In addition, the data provide guidance on future efforts to be addressed in subsequent progress reports.^In particular, the data suggest that waste-feed variability may be the most important future task.^Future efforts will be directed towards assessing the effects of (1) variation in waste solids content and (2) variation in waste dilution.^14 figs., 7 tabs.
2653	An optimum silica flour-bentonite mixture for an engineered barrier	To dispose of low-level and mixed wastes (MAR) by burial, it is necessary to design an impermeable closure, which limits water infiltration through the waste.^Bentonite has very low permeability to water but can be subject to volume alterations.^Over time, these alterations can lead to channeling and subsequent permeability increases.^The fluid conductivity and bulk properties of silica flour and bentonite mixtures were tested to find a mixture that would retain the low conductivity of the bentonite while maintaining volumetric stability.^Silica flour was chosen for its small grain size and spherical shape, and its similarity to silty soil.^Test results indicate that a 90% silica flour and 10% bentonite mixture will provide the optimum properties for this application.^5 refs., 2 figs., 2 tabs.
2671	High-power proton linac for transmuting the long-lived fission products in nuclear waste	High power proton linacs are being considered at Los Alamos as drivers for high-flux spallation neutron sources that can be used to transmute the troublesome long-lived fission products in defense nuclear waste.^The transmutation scheme being studied provides a high flux (> 10{sup 16}/cm{sup 2}{minus}s) of thermal neutrons, which efficiently converts fission products to stable or short-lived isotopes.^A medium-energy proton linac with an average beam power of about 110 MW can burn the accumulated Tc99 and I129 inventory at the DOE's Hanford Site within 30 years.^Preliminary concepts for this machine are described.^3 refs., 5 figs., 2 tabs.

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
2688	Underground tank remediation by use of in situ vitrification	Pacific Northwest Laboratory (PNL) is developing a remedial action technology for underground storage tanks through the adaptation of the in situ vitrification (ISV) process.^The ISV process is a thermal treatment processes that was originally developed for the stabilization of contaminated soil contaminated with transuranic waste at the Hanford Site in southeastern Washington for the Department of Energy (DOE).^The application of ISV to underground storage tanks represents an entirely new application of the ISV technology and is being performed in support of the DOE primarily for the Hanford site and the Oak Ridge National Laboratory (ORNL).^A field scale test was conducted in September 1990 at Hanford on a small cement and stainless steel tank (1-m dia.) that contained a simulated refractory sludge representing a worst-case sludge composition.^The tank design and sludge composition was based on conditions present at the ORNL.^The sludge contained high concentrations of heavy metals including lead, mercury, and cadmium, and also contained high levels of stable cesium and strontium to represent the predominant radionuclide species present in the tank wastes.^The test was highly successful in that the entire tank and surrounding soil was transformed into a highly leach resistant glass and crystalline block with a mass of approximately 30 tons.^During the process, the metal shell of the tank forms a metal pool at the base of the molten soil.^Upon cooling, the glass and metal phases were subjected to TCLP (toxic characteristic leach procedure) testing and passed the TCLP criteria.^Additional sampling and analyses are ongoing to determine the bulk composition of the waste forms, the fraction of volatile or semi-volatile species released to the off-gas treatment system, and to determine whether any soil surrounding the monolith was contaminated as a result of the ISV process.^4 refs., 5 figs., 3 tabs.
2690	Multiphase, multi-electrode Joule heat computations for glass melter and in situ vitrification simulations	Waste glass melter and in situ vitrification (ISV) processes represent the combination of electrical thermal, and fluid flow phenomena to produce a stable waste-from product.^Computational modeling of the thermal and fluid flow aspects of these processes provides a useful tool for assessing the potential performance of proposed system designs.^These computations can be performed at a fraction of the cost of experiment.^Consequently, computational modeling of vitrification systems can also provide and economical means for assessing the suitability of a proposed process application.^The computational model described in this paper employs finite difference representations of the basic continuum conservation laws governing the thermal, fluid flow, and electrical aspects of the vitrification process -- i.e., conservation of mass, momentum, energy, and electrical charge.^The resulting code is a member of the TEMPEST family of codes developed at the Pacific Northwest Laboratory (operated by Battelle for the US Department of Energy).^This paper provides an overview of the numerical approach employed in TEMPEST.^In addition, results from several TEMPEST simulations of sample waste glass melter and ISV processes are provided to illustrate the insights to be gained from computational modeling of these processes.^3 refs., 13 figs.

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
2728	The recovery of neptunium, plutonium and americium from high active waste by TRPO extraction	A mixed trialkyl (hexyl to octyl) phosphine oxide, TRPO, has been chosen as the extractant for the recovery of Np, Pu and Am from HAW owing to it's extraction ability, excellent solvent behavior and low cost.^TRPO is an effective extractant not only for tetra- and hexa-valent, but also for tervalent actinides from nitrate solutions with acidity not higher than {approximately}1.5 mol l. Physical property and radiolytic stability of TRPO and process chemistry based on 30% TRPO-kerosene as solvent will be presented in this talk.^Emphasis will be given to the extraction of Am in the presence of macro amount lanthanides, the adjustment of Np valency by electrolytic reduction, selective stripping of actinides and loading capacity of the solvent.^Optimized process parameters for 99.9% recovery of Am and their experimental verification will be discussed.
2788	Liquid radioactive waste dewatering in thin-film rotor devices	Application possibility of vertical thin-film rotor devices during the first stage of solidification of liquid radioactive wastes is investigated.^Effect of rotor design on device serviceability is studies, effect of technological parameters on process stability and on physico-chemical characteristics of the prepared concentrates is determined.^Concentrates from both design devices (VRK-15 and VRK-50) were vitrificated in cold crusible induction smelter at 1500{plus minus}50 deg C with preparation of silicate glass.
2813	Monazite	Monazite nuclear waste ceramics represent one member of a class of waste disposal materials whose development was predicted on examinations of the established stability of well-characterized mineralogical systems.^In particular, this naturalistic approach resulted in the advancement of both the monazite concept and the Synroc concept.^(The Synroc waste form consists of titanate-based minerals and is treated elsewhere in this volume.) The idea for using a nuclear waste disposal medium based on a synthetic analogue of the mineral monazite was developed by considering the unusual combination of desirable features of this material.^These characteristics are detailed in the article.
2837	Plasma - ultrahigh temperature technology ^A versatile process for smelting special alloys, breaking down toxic waste as well as consolidating radioactive structural pieces and residual matter	Prospective methods for the destruction and stabilisation of special wastes as well as for the consolidation of pieces of radioactive structures and residual matter are described.^Results obtained with prototype and industrial plants are presented.^8 refs.; 12 figs.; 3 tabs.
2914	Radioactive waste solidification technique using cement glass	A new radioactive waste treatment system has been developed to minimize the waste volume and to solidify the waste into a stable form, namely a 'complete inorganic package with maximum volume reduction'.^The primary techniques of this system are drying and pelletizing for volume reduction and cement glass solidification for the final package.^This paper describes the new solidification technique, which has recently been approved for use in commercial plants.^(author).

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
2967	Preparation of fully-active Synroc and its radiation stability. Final report	Samples of Synroc have been made from fully-active waste with a composition chemically equivalent to that which will arise from the Thermal Oxide Reprocessing Plant (THORP) at Sellafield.^ The samples have an activity of 20 GBq.g/sup -1/ and leach-rates, for Cs-137 and Sr-90, measured at 90/sup 0/C, were about 20 and 100 times lower than for equivalent fully-active glass samples.^ The actinide leach rates were significantly higher than the nearest equivalent samples made at Australian Nuclear Science and Technology Organisation (ANSTO), by a factor of 3 for Cm-244 and by a factor of 200 for Pu-239.^ The major cause is thought to be due to premature neutralisation of the waste solution during manufacture, leading to incomplete actinide incorporation in the Synroc phases.^ The actinide leach rates for the fully-active samples were nevertheless 15 to 450 times less than those for fully active glass samples.^ Synroc samples doped with Pu-238 and Cm-244 have also been made to test the material's radiation stability.^ Some Pu-238 doped samples made earlier from less well characterised materials, but with the correct phase composition, have decreased in density by about 6% after a dose equivalent to 300,000 years for real waste.
2975	In situ vitrification - a status of the technology	The In Situ Vitrification (ISV) process is a new technology developed from its conceptual phase to selected field-scale applications in the last 5 years.^The US Department of Energy (DOE) has sponsored the ISV program to develop alternative technology for potential application to contaminated soil sites.^The ISV process converts contaminated soils and wastes into a durable glass and crystalline waste form in place by melting using joule heating.^The ISV process has been developed through a series of 25 engineering-scale (laboratory) tests, 10 pilot-scale (small field) tests, and four large-scale (full-scale field) tests.^Its major advantages for stabilizing radioactive and hazardous wastes are found to be: safety in terms of minimizing worker and public exposure; long-term durability of waste form (more than 1 million years); cost effectively ($150 to $300/m/sup 3/; applicability to a wide variety of soils and inclusions; and potential for eliminating exhumation, transport, and handling.
2979	Evaluation of NDA techniques applying to the measurement of alpha-activities in medium or low level radioactive reprocessing wastes, (4). Examination of performance of neutron detector in high exposure rate	In order to evaluate non-destructive assay techniques for the measurement of alpha-activities in medium and low level reprocessing wastes, the performances of several kinds of neutron detectors were examined in high gamma-radiation fields, in which we measured the allowable level of gamma-ray exposure rate and proposed the improvement of detectors in rather high gamma-radiation fields.^The allowable level was estimated to be ca.^4R/h for a He-3 counter, ca.^20R/h for a BF/sub 3/ counter and ca.^800R/h for a B-10 lined counter.^We proposed the necessities of expedition and stabilization of signal treatment, lightening of the material of detector wall and utilization of a surface barrier semiconductor detector in rather high gamma-radiation fields.

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
2982	Titanate ceramics for the immobilization of sodium-bearing high-level nuclear waste	The phase chemistries and microstructures of titanate-based ceramics containing simulated high-level nuclear waste with varying sodium contents were compared.^Incorporation of relatively low sodium levels resulted in more complex phase assemblages.^The principal hosts for sodium were hibonite and freudenbergite, and, when iron and sodium were present in combination, loveringite was also stabilized.^During fabrication, oxygen potential was controlled by Ti-TiO/sub 2/ or TiH/sub 2/-Ti/sub 3/O/sub 5/ solid-state buffers.^These metal and hydride oxygen getters behaved similarly, neither disturbing the phase assemblages nor significantly altering the partitioning of waste elements between radiophases.^It is believed that the hydrothermal stability of the sodium-bearings ceramics (containing up to 2.7 wt% Na/sub 2/O) will be comparable to sodium-free material, since less durable sodium-rich phases are encapsulated in a resistant matrix.^Extensive formation of glassy phases may cause embrittlement at higher sodium loadings.
2991	Saline liquid wastes-solidification by cementation	A program is presented for studying the immobilization of saline liquid wastes arising from an experimental reprocessing plant, destined to obtain mixed oxides for a demonstration program of Pu recycle in natural uranium reactors (Atucha I).^The object of this plan is to turn the liquid wastes of medium activity into a solid form which must be chemical, thermal and radiologically stable.^The selection of an optimum solid waste is based in considerations that include processing confidence, safety and economy.^In the initial stage, the improvement of a cement matrix properties (mechanical properties, leaching resistance, etc.) by the incorporation of a polymer (PIC), different aggregates or special treatments is tried.
2993	Fabrication and performance of SYNROC	A systematic study has been made of the effect of fabrication conditions on the chemical durability of SYNROC.^The most important factors are the type of precursor used, redox control and hot-pressing conditions.^Hot pressing at 1150-1200/degrees/C for 2 hours at a pressure of 14-21 MPa is sufficient to produce SYNROC of near theoretical density.^Transuranic elements and fission products have been incorporated into SYNROC in separate glove-box and hot cell fabrication facilities.^The most leachable fission products are cesium, technetium and barium but their leach rates decrease rapidly with time.^The transuranic elements, rare earths, zirconium and ruthenium have very low leach rates.
2999	Use of liquid membranes for treatment of nuclear wastes	The reprocessing operations produce liquid wastes in which the main components are nitric acid and sodium nitrate.^The goal of the experiments is to separate trace amounts of radioactive elements from these acidic and high sodium nitrate content solutions.^CMPO, a neutral bifunctional organophosphorus compound, and crown compounds (DC18 C6 - B21 C7) are able to extract respectively actinides, strontium and cesium from these high salinity solutions.^The supported liquid membrane (SLM) render the use of expensive tailor-made extractant molecules like CMPO or crown ethers possible.^The results obtained for the extraction of actinides and strontium are promising, but research must now be oriented towards improving the stability of the membrane.

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
3006	In situ vitrification demonstration for the stabilization of buried wastes at the Oak Ridge National Laboratory	A demonstration of In Situ Vitrification (ISV) technology for the stabilization of radioactively contaminated soil sites at the Oak Ridge National Laboratory (ORNL) was successfully completed during July 1987.^This demonstration is the first application of the ISV process not performed at the Hanford Site, where the technology was developed and patented for the U.S.^Department of Energy by Pacific Northwest Laboratory (PNL).^The joint ORNL-PNL pilot-scale demonstration was performed on a 3/8-scale trench (2 m deep x 1 m wide x 10 m long) that was constructed to simulate a typical seepage trench used for liquid low-level radioactive waste disposal at ORNL from 1951 to 1966.^In the ISV process, electrodes are inserted around a volume of contaminated soil, power is applied to the electrodes, and the entire mass is melted from the surface of the soil down through the contaminated zone, thus making a glassy-to-microcrystalline waste form that incorporates the contaminants.^Gases produced during the melting are collected, treated, monitored, and released through an off-gas process trailer.^In the ORNL demonstration, a 25,000-kg mass of melted rock approximately 1.2 m thick x 2.1 m wide x 4.9 m long was formed during 110 h of operation that consumed approximately 29 MWh of power.^Data obtained on the operational performance of the test and waste-form durability will be used to assess the feasibility of applying the ISV technology to an actual waste trench.
3009	Treatment of hazardous metals by in situ vitrification	Soils contaminated with hazardous metals are a significant problem to many Defense Program sites.^Contaminated soils have ranked high in assessments of research and development needs conducted by the Hazardous Waste Remedial Action Program (HAZWRAP) in FY 1988 and FY 1989.^In situ vitrification (ISV) is an innovative technology suitable for stabilizing soils contaminated with radionuclides and hazardous materials.^Since ISV treats the material in place, it avoids costly and hazardous preprocessing exhumation of waste.^In situ vitrification was originally developed for immobilizing radioactive (primarily transuranic) soil constituents.^Tests indicate that it is highly useful also for treating other soil contaminants, including hazardous metals.^The ISV process produces an environmentally acceptable, highly durable glasslike product.^In addition, ISV includes an efficient off-gas treatment system that eliminates noxious gaseous emissions and generates minimal hazardous byproducts.^This document reviews the Technical Basis of this technology.^5 refs., 7 figs., 2 tabs.
3017	Chromatographic decontamination of concentrated intermediated level waste solutions	A method has been developed for decontaminating the carbonate washing liquid of the PUREX process stream for treatment of the extraction agent (tributylphosphate, dissolved in kerosine), and the limits of the method with regard to solutions very rich in salt.^The chromatographic method makes use of the fact that the actinides and carbonates form very stable complexes which are retained by weakly basic, anionic exchangers that can take up to 300 g U/kg of resin.^The goal is to achieve a separation of the concentrated, nitric MAW solution into a small quantity of HAW and a large one of LAW, aiming at the same time at minimizing the secondary LAW amounts which in most cases consist of exchanger materials.

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
3023	Use of liquid membranes for treatment of nuclear effluents	The possibility of using thin layers of liquid extractants immobilized on supports interposed between two aqueous solutions is examined.^The ions are transported across the membrane against their concentration gradient: the driving force is provided by the chemical potential difference of some chemical species, present on the two sides of the membrane.^At CEN Cadarache, the supported liquid membranes are utilized for the decontamination of evaporator concentrate arising from reprocessing plants in order to split it in two fractions: a fraction containing most of the inactive salts for storage in shallow land burial and a small highly active fraction for a geological repository.^Actinides, cesium and strontium, the main long lived nuclides present in evaporator concentrate, must be removed with relatively high decontamination factors: 100 for actinides, 50 for cesium and strontium.^Very selective extractants are utilized to remove these ions.^Efficiency and stability of membranes are discussed.
3030	Measurement of effective K/sub D/ factors for the long-lived uranium and thorium isotopes in samples of London Clay (Bradwell) and mudrock (Fulbeck)	Measurements of the actinide isotopic activities and their activity ratios in the natural decay series provide a direct method for determining sorption parameters in field samples and give an indication of the stability of a given rock/water system.^ Both of these aspects are of relevance to the far-field of the radioactive waste disposal in geological formations.^ The distributions of /sup 238/U, /sup 234/U, and /sup 230/Th, and /sup 232/Th and /sup 228/Th radionuclides from the decay series originating with /sup 238/U and /sup 232/Th respectively, have been studied for the different liquid and solid phases in London Clay at Bradwell and mudrock at Fulbeck.^ The distribution coefficients (K/sub D/) and retardation factors (R) have been deduced from the radiometric data obtained by isotope dilution/alpha spectrometry techniques.
3061	Sensitive method for quantitative measurements of the concentration of various gamma-emitting radionuclides in reactor plumes using a germanium detector	The work is aimed at studying the possibility of direct radionuclide concentration measurements in gaseous reactor release with a semiconducting detector (SCD) on the basis of high-frequency high-resolution germanium, as well as at applying the results of such measurements to estimate parameters of the model used and to calculate maximum permissible release.^The method of identification and determination of the power of radioactive noble gas release is described.^The results of calculations are compared with the experimental data.^The minimum detected release power is estimated.The conclusion is made that the above detectors are applicable to measure radioactive releases both in the case of normal reactor operation and in emergencies as they can be used to estimate concentration of separate radionuclides.^The application of germanium SCD enables one to obtain information on the time of release transfer in data on such short-lived daughter radionuclides as /sup 88/Rb.

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
3071	Immobilisation of cesium in SYNROC	SYNROC is a synthetic mineral designed to immobilize radioactive waste from the reprocessing of spent nuclear fuels.^The principal phases in this mineral are hollandite ($Ba_x(AlTi)_8O_{16}$), zirconolite ($CaZrTi_2O_7$) and perovskite ($CaTiO_3$).^The radioactive waste elements are accommodated in solid solution within the crystal structures of these phases.^One of the major problems of radioactive waste disposal is the immobilisation of ^{137}Cs.^This element, in its ionic state Cs^+, is both large and massive and, is normally associated with soluble compounds.^In SYNROC, cesium is accommodated between the barium ions in the tunnels of the Hollandite phase.^Considerable success has been achieved substituting high concentrations of cesium in reduced aluminous hollandites (i.e.^$(CsBa)_x(AlTi)_8O_{16}$) by hot pressing although some of these on their own are unstable under oxidising conditions at high temperatures.^Oxidation can be inhibited in SYNROC by incorporating Ti powder within the multiphase matrix.^Hollandites with low concentrations of cesium (i.e. up to 10 % occupancy) are stable on their own up to 1300 degrees C. Nevertheless, the loss of cesium in SYNROC by leaching is an order of magnitude less than the borosilicate glasses presently being employed as storage materials.^Neutron diffraction studies, as well as electron and x-ray diffraction studies, have been carried out to determine the structural characteristics of hollandites and the way in which cesium is accommodated in the structure.^(author) 19 refs., 9 figs., 1 tab.
3093	Sorptive removal of technetium from alkaline heavy metals sludge filtrate containing nitrate ion	A so-called "raffinate" waste stream is generated from various uranium recovery and equipment cleaning and decontamination activities at the X-705 facility of the Portsmouth (Ohio) Gaseous Diffusion Plant (PORTS).^The composition of this waste stream is generally characterized by high concentrations of nitric acid, toxic heavy metals, and low level of radioactive nuclides (^{235}U, ^{99}Tc).^We have found that cross-linked poly-4-vinyl-pyridine (PVP) resin (such as Reillex 402) is more efficient than strongly basic anion exchange resin for removed of Tc in wastes containing high concentrations of nitrate ion, resin loading Netherlandsnitrate is greatly reduced, and PVP resins are very stable with respect to chemical and radiological degradation.^Also the inexpensive inorganic reagents, elemental iron (degreased iron filings, about 40 mesh) and ferrous sulfide (in particular Greigite (Fe_3S_4)) are very efficient for the removal of Tc and also soluble mercury from aqueous nitrate wastes.^2 refs., 1 fig., 4 tabs.
3097	In Situ Vitrification: Recent test results for a contaminated soil melting process	In Situ Vitrification (ISV) is being developed at Pacific Northwest Laboratory for the Department of Energy and other clients for the stabilization of soils and sludges contaminated with radioactive and hazardous chemical wastes.^ISV is a process that immobilizes contaminated soil in place by converting it to a durable glass and crystalline product that is similar to obsidian.^In June 1987, a large-scale test of the process was completed at a transuranic- contaminated soil site.^This constituted the first full-scale demonstration of the ISV process at an actual site.^This paper summarizes the preliminary results of this test and describes the processes' potential adaptation to radioactive and hazardous chemical waste contaminated soils.^10 refs., 10 figs.

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
3123	Present status of technical development in new solidification techniques for high level wastes. Synroc solidification technique	It is necessary to isolate the high level radioactive wastes produced in the reprocessing of spent nuclear fuel from human environment for long period by stably solidifying radioactive nuclides.^For the purpose, the method of solidifying with borosilicate glass is developed in various countries.^As to this method process is relatively simple, the technical development is advanced, and any element can be easily taken in because of the amorphism, but the thermodynamic stability is low, and the resistance to hot water is weak.^As the substitute solidifying method, there is Synroc method, which is one of rock solidifying method.^The Synroc is composed of a stable crystalline mineral phase based on TiO/sub 2/, and it is expected that when it is disposed in the ground, the compatibility with ambient rocks is particularly excellent.^Besides, waste components are taken in the crystalline structure, therefore, the thermodynamic stability, the resistance to groundwater leaching and the long term stability against radiation damage can be expected.^The kinds and composition of Synroc, the method of its manufacture, the structure and properties of Synroc and so on are reported.^(Kako, I.).
3139	Research and development of high level radioactive waste form	The solidified bodies of high level radioactive wastes function as the first barrier for preventing the release of radioactive nuclides in the disposal system.^In this paper, the present status of research and development is explained about the methods of evaluation and comparison, the proper composition, the behavior of leaching, the radiation damage, and the soundness in terms of geological epoch, centering around borosilicate glass and Synroc which are considered to be the most promising as the solidified bodies at present.^When 1 t of the spent fuel burned up to about 30,000 MWd-MTM in a LWR is reprocessed, the radioactive waste liquid as high level as 400,000 Ci at maximum of 1 m/sup 3/ is generated.^Most of the radioactivity are generated by the fission products having the half value period shorter than 30 years, but partly the transuranium elements having the half value period over 10,000 years are contained.^Therefore, those must be solidified by using proper media, and disposed in deep strata to isolate them for long period from the biosphere.^Glasses, ceramics, concrete and composites have been examined, and the evaluation and selection were carried out in USA.^The borosilicate glass is easy to manufacture and industrialize, while Synroc has the excellent stability.^(Kako, I.).
3174	Preparation of Synroc and its radiation stability. Progress report July 1986	Samples of Synroc have been made from simulated Highly Active Waste and 'Sandia' precursor supplied by the Australian Atomic Energy Commission, with a view to gaining experience for making fully active samples.^ The Synroc pellets were of good density (approx.^ 4.5g.cm/sup -3/), and and with comparable leach rates to samples made in Australia but the micro-structure was rather coarser.^ Some Pu-238 doped pellets made previously from low surface area precursors but with the correct phase structure have now received a radiation dose equivalent to an age of 400,000 years for Synroc containing fully active waste.^The samples are still intact but their volumes have increased by 6%, in agreement with Australian results on samples damaged by neutron irradiation.^

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
3210	In situ vitrification demonstration for the stabilization of buried wastes at the Oak Ridge National Laboratory	A demonstration of In Situ Vitrification (ISV) technology for the stabilization of radioactively contaminated soil sites at the Oak Ridge National Laboratory (ORNL) was successfully completed during July 1987.^This demonstration is the first application of the ISV process not performed at the Hanford Site, where the technology was developed.^The joint ORNL-PNL pilot-scale demonstration was performed on a 3/8-scale trench (2 m deep x 1 m wide x 10 m long) that was constructed to simulate a typical seepage trench used for liquid low-level radioactive waste disposal at ORNL from 1951 to 1966.^In the ISV process, electrodes are inserted around a volume of contaminated soil, power is applied to the electrodes, and the entire mass is melted from the surface of the soil down through the contaminated zone, thus making a glassy-to-microcrystalline waste form that incorporates the contaminants.^Gases produced during the melting are collected, treated, monitored, and released through an off-gas process trailer.^In the ORNL demonstration, a 25-t mass of melted rock approximately 1.2 m thick x 2.1 m wide x 4.9 m long was formed during 110 h of operation that consumed approximately 29 MWh of power.^Data obtained on the operational performance of the test and waste-form durability will be used to assess the feasibility of applying the ISV technology to an actual waste trench.
3216	Large liquid scintillation counter for fast neutron capture cross section measurements	For determining fast neutron radiative capture cross sections a new capture gamma-ray counter of approximately 4..pi.. geometry has been developed.^The counter is in a sphere shape with a diameter of 1 meter filled with 680 liter liquid scintillator.^The tank is shielded by 10 cm thickness of lead and 40 cm thickness of Li-paraffin.^A coincidence method is used for reducing background due to neutron capture by the hydrogen in the liquid scintillator.^The counter has been successfully used for measuring the neutron capture cross sections of Au-197, Ta-181 and Tm-169 on a 2.5 MeV pulsed Van de Graaff accelerator.
3227	Integrated oxidation-glass formation process for waste sodium disposal	For permanent disposal of the large quantities of mildly contaminated waste liquid metals (primarily sodium) used as coolants in nuclear reactors, and currently stored at different DOE sites, it is necessary to convert the metal to a stable and environmentally acceptable disposable form at minimum cost.^Work on air oxidation of sodium done by Mine Safety and Appliances, Inc.^(MSA) in 1980 showed the feasibility of converting the metal to its oxides.^This study investigates the development of an integrated sodium burn-glass formation process to convert waste sodium to a disposable glass form.

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
3241	Encapsulation of noble gas in zeolites	The noble gases neon, argon, krypton and xenon were encapsulated hydrothermally as pure gases as well as in the form of mixtures in type A zeolite of various cationic compositions.^As opposed to the starting material the encapsulates are X-ray amorphous and posess a very small specific surface area.^Irrespective of the thermal pretreatment of the zeolites the optimal loading occured within a certain temperature window.^The amount of gas trapped was essentially a function of the fixation pressure.^Within the pressure range 50-2200 bar the obtained loading was independent from the type of noble gas.^When mixtures of noble gases were encapsulated a small enrichment of the heavier noble gas, caused by kinetic and thermodynamic effects, was observed.^The thermal stability of the encapsulates was found to be very high.^Even at temperatures as high as 750/sup 0/C a recrystallization to anorthite was only observed after practically all the trapped gas had been released.^Experiments destined to clarify the mechanism of gas leckage at temperatures below 750/sup 0/C suggest a diffusion controlled mechanism described by a ..sqrt..t law.^Even at loadings of 45 ml/(STP) Kr/g the leckage predicted under conditions as expected during longterm storage of Kr-85 is extremely small.^Kinetic data demonstrate that the gas is not trapped in the form of agglomerates but rather exists homogeneously distributed within the encapsulate.^This result is substantiated by electron beam microanalysis.
3256	Sensitive method for quantitative measurements of the concentration of various gamma-emitting radionuclides in reactor plumes using a germanium detector	The Gaussian plume model has been adapted to calculate the primary photon fluence rate for a given release rate, height and distance over long periods and during different types of weather conditions.^Corrections due to the varying efficiency of a Ge detector for different photon energies, angle of incidence and plume configuration have been developed.^Using these, together with background measurements, minimum detectable release rates from nuclear power stations during normal operation have been estimated for radioactive inert gases.^A carefully calibrated Ge detector, with proper lead shielding, is a sensitive instrument for long-term average measurements of radioactive plumes and is far superior to GM counters and ionizing chambers.^The release rate may be evaluated more accurately than before by means of different dispersion models.
3270	Sensitive method for quantitative measurements of the concentration of various gamma-emitting radionuclides in reactor plumes using a germanium detector	The Gaussian plume model has been adapted to calculate the primary photon fluence rate for a given release rate, height and distance over long periods and during different types of weather conditions.^Corrections due to the varying efficiency of a Ge detector for different photon energies, angle of incidence and plume configuration have been developed.^Using these, together with background measurements, minimum detectable release rates from nuclear power stations during normal operation have been estimated for radioactive inert gases.^A carefully calibrated Ge detector, with proper lead shielding, is a sensitive instrument for long-term average measurements of radioactive plumes and is far superior to GM counters and ionizing chambers.^The release rate may be evaluated more accurately than before by means of different dispersion models.

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
3300	In situ vitrification of Oak Ridge National Laboratory soil and limestone	Process feasibility studies were successfully performed on two different developmental scales to determine the technical application of in situ vitrification (ISV) to Oak Ridge National Laboratory (ORNL) intermediate-level waste.^In the laboratory, testing was performed on crucibles containing quantities of 50% ORNL soil and 50% ORNL limestone.^In the engineering-scale testing, a 1/12-scaled simulation of ORNL Trench 7 was constructed and vitrified, resulting in waste product soil and limestone concentrations of 68% and 32%, respectively.^Results from the two scales of testing indicate that the ORNL intermediate-level waste sites may be successfully processed by ISV; the waste form will retain significant quantities of the cesium and strontium.^Because /sup 137/Cs is the major component of the radionuclide inventory in the ORNL seepage pits and trenches, final field process decontamination factors (i.e., off gas at the ground surface relative to the waste inventory) of 10/sup 4/ are desired to minimize activity buildup in the off-gas system.^These values were realized during the engineering-scale test for both cesium and strontium.^The vitrified material effectively contained 99.996% of the cesium and strontium placed in the engineering-scale test.^This is equivalent to decontamination factors of greater than 10/sup 4/.^Volume reduction for the engineering-scale test was 60%.^No migration of the cesium to the uncontaminated surrounding soil was detected.^These favorable results indicate that, once verified in a pilot-scale test, an adequately designed ISV system could be produced to treat the ORNL seepage pits and trenches without excessive activity accumulation in the off-gas treatment system.
3301	Development of equipment for stabilization and isolation of low-level liquid waste disposal cribs: A descriptive summary	A prototypic system that can be used to stabilize and isolate low-level liquid radioactive waste disposal sites has been developed.^This system mechanically stabilizes underground waste disposal structures by consolidation of the material within and surrounding condition waste zone.^Isolation is provided by encasement in a cementitious monolith containing the waste structure and radioactive materials within and in close proximity to the structure.^The dynamic consolidation and grout injection system has been developed and tested at the Hanford Site.^Several categories of liquid, low-level waste disposal structures can be treated with the developed system to prevent subsidence and enhance long-term confinement of contaminants.
3304	In situ vitrification of transuranic wastes: An updated systems evaluation and applications assessment	In situ vitrification (ISV) is a process whereby joule heating immobilizes contaminated soil in place into a durable glass and crystalline waste form.^Numerous technological advances made during the past three years in the design, fabrication, and testing of the ISV process are discussed.^Performance analysis of ISV focuses on process equipment, element retention (in the vitrified soil during processing), melt geometry, depth monitors, and electrodes.^The types of soil and waste processed by ISV are evaluated as process parameters.^Economic data provide the production costs of the large-scale unit for radioactive and hazardous chemical wastes (wet and dry).^The processing of transuranic-contaminated soils are discussed with respect to occupational and public safety.^Alternative applications and operating sequences for various waste sites are identified.^The technological data base warrants conducting a large-scale radioactive test at a contaminated soil site at Hanford to provide a representative waste form that can be evaluated to determine its suitability for in-place stabilization of transuranic-contaminated soils.

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
3306	Distribution coefficients used for safety assessment for shallow land radioactive waste burial	Distribution coefficients (Kd) used for an estimation of radionuclide migration in soil were reviewed.^The relation between Kd and soil, water quality and chemical forms in groundwater were examined.^The Kd values for generic safety assessment on shallow land burial are proposed as follows on the basis of knowledge of measured Kd values and expected chemical forms in groundwater.^Group 1: tritium (Kd = 0 ml/g); Group 2: Tc, I etc. existing in groundwater as an anion (Kd = 0 -- 10 ml/g); Group 3: alkali earth elements which form insoluble carbonate and sulfate (Kd = 5 -- 50 ml/g); Group 4: rare earth elements or transition elements which form hydroxide or complex (Kd = 10 -- 100 ml/g); Group 5: actinide elements which form polymer, or elements existing in water as a stable cation such as Cs etc. (Kd = 100 -- 1,000 ml/g).^ Contains 55 refs.
3369	Experimental approach to the sintering methodology for a complex ceramic system	A methodology for the densification of a complex ceramic is developed through the experimental assessment of the parameters affecting the sintering of the material.^The material studied is an extremely complicated one known by its acronym SYNROC.^It is a dense, thermodynamically stable ceramic intended for the containment of high level nuclear wastes (HLW).^It is composed of five matrix oxides (Al/sub 2/O/sub 3/, BaO, CaO, TiO/sub 2/, and ZrO/sub 2/) which then properly sintered form three major crystalline phases (hollandite perovskite, and zirconolite) for the purpose of hosting up to approx.^=20 wt% simulated HLW species.^In order to examine and optimize the parameters affecting the processing and ultimately the microstructural integrity of such a complex system, specific analytical and characterization techniques had to be revamped or developed.^From the information gained from the study of these areas and their associated parameters SYNROC was successfully characterized and optimally sintered.^Because of the favorable resolution of this extreme nonideal case, the analytical methods and processing procedures established here are considered generally applicable to a wide range of sintered particulate materials.
3370	In-situ vitrification: a status of the technology	The In Situ Vitrification (ISV) process is a new technology developed from its conceptual phase to selected field-scale applications in the last 5 years.^The US Department of Energy (DOE) has sponsored the ISV program to develop alternative technology for potential application to contaminated soil sites.^The ISV process converts contaminated soils and wastes into a durable glass and crystalline waste form in place by melting using joule heating.^The ISV process has been developed through a series of 25 engineering-scale (laboratory) tests, 10 pilot-scale (small field) tests, and four large-scale (full-scale field) tests.^Its major advantages for stabilizing radioactive and hazardous wastes are found to be: safety in terms of minimizing worker and public exposure; long-term durability of waste form (more than 1 million years); cost effectiveness ($150 to $300/m/sup 3/); applicability to a wide variety of soils and inclusions; and potential for eliminating exhumation, transport, and handling.
3433	Electrical technique for in-place stabilization of contaminated soils	In situ vitrification is discussed as an emerging technology that is currently being tested for the potential in-place stabilization of radioactive, transuranic wastes at Pacific Northwest Laboratory.^The advantages to in situ vitrification are listed and include: long-term stabilization of radioactivity; cost effectiveness; applicability to varying soil and conditions; minimal occupational exposure to the waste during processing; and, low energy requirements.

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
3434	In-situ vitrification: a large-scale prototype for immobilizing radioactively contaminated waste	Pacific Northwest Laboratory is developing the technology of in situ vitrification, a thermal treatment process for immobilizing radioactively contaminated soil.^A permanent remedial action, the process incorporates radionuclides into a glass and crystalline form.^The transportable procss consists of an electrical power system to vitrify the soil, a hood to contain gaseous effluents, an off-gas treatment system and cooling system, and a process control station.^Large-scale testing of the in situ vitrification process is currently underway.
3435	New sensors for the 24024 mass per unit area measurement system	A new generation of sensors has been developed for the Robotron 24024 mass per unit area measurement system.^In comparison to previous sensors the stability of the measuring signal, the response time, and the resolution of profile measurements have been improved and the total range of measurement has been enlarged.^By constructional measures and additional shielding The the working container and ionization chamber the radiation protection measures can be made more effective.
3486	Investigating the construction of [open quotes]pyramid[close quotes] super-structures to dispose of radioactive and hazardous waste	The technical challenges involved in disposing of radioactive and hazardous waste using systems that rely on [open quotes]natural barriers[close quotes] have been underestimated.^Technology has advanced dramatically in the areas of materials, science, and engineering.^As a result, traditional approaches to waste disposal must be rethought, focusing instead on ways to apply new technology breakthroughs to waste disposal problems.^This paper will discuss problems currently faced by waste disposal systems that rely on natural barriers for containment, propose a general design and approach for constructing pyramid super-structures to dispose of (and store) waste, and present the benefits of such a system.^The originality of this paper is that it proposes the construction of fully retrievable waste disposal systems that capitalize on the unique geometric properties of a pyramid and rely on engineered barriers and preventive measurements for containment, rather than natural barriers.^In addition, this paper offers a new perspective on waste disposal issues confronting many countries.^The desired effect is that by challenging conventional thought, new ideas could be developed to help solve existing problems.^This paper is of specific interest to: (1) Policy makers, decisionmakers, and managers because the paper discusses the root causes of problems facing waste disposal and possible solutions.^(2) Regulators because the paper discusses reasons for relying more on engineered barriers and preventive measurements to achieve confidence and reliability in containment.^(3) Design engineers the paper offers new concepts for engineered barriers.^21 refs., 2 figs., 2 tabs.
3506	Microtunneling techniques to form an insitu barrier around existing structures	The use of microtunneling techniques to construct underground structures has been available for many years.^The technique, typically termed pipe roofing, is well proven and, to date, has been used principally to form underground access under existing structures where the use of cut and cover or conventional tunneling techniques was either impossible or too expensive.^The first project involving the use of Iseki equipment, the first use of the pipe roofing technique as far as we are aware, was in 1971.^This was followed by a number of projects in Japan, together with projects in Singapore and Hong Kong, totaling 12 projects.^More recently, projects have been completed in Singapore and Japan and are currently in progress in Japan and Malaysia.^This paper will briefly describe the use of the technique and the use of the new Iseki Unclemole for hard rock excavation to construct the largest pipe roof project ever undertaken in the world, together with other projects.^The pipe roofing technique allows the construction of underground structures, in a variety of shapes and forms, by the use of remotely operated microtunnel machines.^The paper will describe the potential application of this technique in the areas of hazardous and nuclear waste management.

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
3632	Diffusion of radionuclides in concrete/bentonite systems	In a repository for nuclear waste, different construction materials will be used.^Two important materials among these are concrete and bentonite clay.^These will act as mechanical barriers, preventing convective water flow and also retard transport due to diffusion of dissolved radionuclides by a combination of mechanical constraints and chemical interactions with the solid.^An important issue is the possible change of the initial sodium bentonite into the calcium form due to ion exchange with calcium from the cement.^The initial leaching of the concrete has been studied using radioactive spiked concrete in contact with compacted bentonite.^The diffusion of Cs, Am and Pu into 5 different types of concrete in contact with porewater have been measured.^The measured diffusivity for Cs agrees reasonable well with data found in literature.^For Am and Pu no movement could be measured (less than 0.2 mm) even though the contact times were extremely long (2.5 y and 5 y, respectively).^This report gives also a summary of the previously published results about sorption and diffusion of radionuclides in cement performed in Prav/KBS/SKB projects 1980-1990.^25 refs.
3652	Natural zeolites as barriers against migration of radionuclides from radioactive waste repositories	The radioactive wastes from the Nuclear Power Plant in Bulgaria, both low- and intermediate level and high-level wastes have to be safely disposed and isolated from the human environment.^Natural inorganic sorbents are known to act as effective barriers against radionuclide migration from radioactive waste repository.^Zeolites could be used as buffer, backfill and sealing material in repository in tuff, as an additives to the bentonite to increase the retention of cesium-137, and their presence in the host rock or the surrounding geological structures will increase the sorption properties of the strata.^We have studied the ion exchange properties of natural zeolites (clinoptilolites and mordenite) from different deposits in Bulgaria in order to obtain the suitable sorbent for cesium-137, strontium-90 and cobalt-60.^Additional experiments were made with selected sorbent to estimate it's sorption properties.^(authors).^4 tabs., 8 refs.
3683	Microwave processing of geopolymer-cement based waste forms	Mineral geopolymeric alkali-silico-aluminates are new acid-resistant cementitious materials, with zeolitic properties.^Hardening involves the chemical reaction of alumino-silicate oxides (Al[sup 3+] in IV-V fold coordination), with alkali polysilicates, yielding polymeric Si-O-Al bonds.^Microwave processing yields solid, monolithic, high-strength (100 MPa compression) geopolymer wastes forms (zeolitic in nature), which immobilize hazardous materials; heavy metals and radioactive wastes are locked into the three dimensional zeolitic framework.^Superfast efficient microwave drying yields temperature stable waterfree ceramic waste forms of any dimension and shape, which may be heated up to 1000[degrees]C without cracking and failure, or vitrified.^Microwave-dried Geopolymer radioactive waste forms possess the properties required for safe underground disposal by minimizing steam explosion and hydrogen release.^This paper addresses also the applications of Geopolymers as acid-resistant geological barriers, safe and innocuous containment and disposal of hazardous wastes and mine tailings, and the disposal and confinement of low-level and medium level radioactive wastes.^21 refs., 8 figs., 3 tabs.

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
3820	Radioactive waste processing method	In a method of processing radioactive wastes by confining radioactive wastes by using solidified mortars, a resin incorporated mortar layer comprising fine aggregates (mountain sands or river sands having a grain size of less than 5mm), a synthetic resin (acryl resin) and a mixing material (calcium carbonate or silica sand powder) blended at a predetermined ratio is formed on the inner surface of a drum can.^Then, after supplying the radioactive waste therein, a mortar filler having portland cement, a water reducer, aggregates and water blended at a predetermined ratio is in the drum can.^This can certainly prevent intrusion of underground water and leakage of radioactivity, even if the drum can should suffer from corrosion when it is buried underground thereby enabling to confine them in the can stably for a long period of time.^(T.M.).
3842	In situ vitrification: Demonstrated capabilities and potential applications	A large-scale demonstration of the in situ vitrification (ISV) process was performed in April 1990 on the 116-B-6A Crib in the 100 Area of the Hanford Site in southeastern Washington.^The 116-B-6A Crib is a radioactive mixed waste site and was selected to demonstrate the applicability of ISV to soils contaminated with mixed wastes common to many US Department of Energy (DOE) sites.^Results from the demonstration show that the ISV process is a viable remediation technology for contaminated soils.^The demonstration of the ISV process on an actual contaminated soil site followed research and development efforts by the Pacific Northwest Laboratory (PNL) over the last 10 years.^PNL's research has led to the development of the ISV process as a viable remediation technology for contaminated soils and the creation of a commercial supplier of ISV services, Geosafe Corporation.^Development efforts for ISV applications other than treatment of contaminated soils, by PNL and in collaboration with Oak Ridge National Laboratory (ORNL) and Idaho National Engineering Laboratory (INEL), show the ISV process has potential applicability for remediating buried waste sites, remediating underground storage tanks, and enabling the placement of subsurface vitrified barriers and engineered structures.^This paper discusses the results from the April 1990 large-scale demonstration and provides a general overview of the current capabilities of the ISV process for contaminated soils.^In addition, this paper outlines some of the technical issues associated with other ISV applications and provides a qualitative discussion of the level of effort needed to resolve these technical issues.
3852	The migration of Cs-137 and Co-60 through cementitious barriers applicable to radioactive waste conditioning	An experimental method was developed to measure the penetration of radioactivity through cementitious barrier materials such as those considered for radioactive waste management.^The radionuclides [sup 60]Co and [sup 137]Cs were immobilized by a cementation process and conditioned in concrete containers in order to simulate the storage of radioactive waste materials in a trench system repository.^These results will be used in the design of a future Yugoslav radioactive waste storage center.^(Author).

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
3886	Innovative developments in in situ vitrification: Vitrified underground barriers	Three U.S.^Department of Energy contractors are developing a remedial action technology called in situ vitrification (ISV) for contaminated soils and advanced applications such as buried wastes and underground storage tanks.^The ISV is a thermal treatment process initially developed at Pacific Northwest Laboratory (PNL) for contaminated soil applications.^This paper describes work conducted by PNL involving vitrified underground barrier applications.^The PNL has conducted several engineering-scale laboratory tests to evaluate ISV for underground barrier applications.^For this application, clean soil surrounding a waste site could be vitrified in order to isolate wastes from undesirable contact or transport.^Engineering-scale experiments have demonstrated that a subsurface floor can be established resulting in a horizontal, planar glass block, and vertical planar walls can be created and bonded to the underlying horizontal floor.^In addition, modeling indicates that the potential for significant cracking is unlikely.^Vitrified barriers may provide a significant technology alternative for both interim or permanent remedial actions.^Vitrified barriers are expected to be functional for geologic periods and can be emplaced without soil excavation.
3901	A system for measuring moisture transients in clay-based barrier materials	This paper discusses the Buffer/Container Experiment which is one of the large-scale insitu experiments being conducted by AECL Research at its Underground Research Laboratory.^The experiment is intended to examine the thermal-hydraulic-mechanical performance of sand-bentonite buffer material in a single emplacement borehole arrangement under in situ boundary conditions.^Thermocouple psychrometers and thermal needles are being used as moisture sensors to track moisture transients in the buffer as the experiment progresses.^Excitation and logging of the moisture sensors are largely automated.^Procedures are being implemented to provide full automation of the moisture sensor system and to facilitate data conversion and management.^Preliminary results from the Buffer/Container Experiment show that, up to five months after installation, the majority of the moisture sensors continue to function reliably.
3920	Solidification method for radioactive waste	As a method for processing low level radioactive wastes, it has been known that they are solidified by using inorganic solidifying materials such as cement and water glass.^In the present invention, it is considered that not only the low level wastes but also middle level wastes are solidified by cement or the like and then put to land disposal.^Therefore, hydrophobic materials such as oils and silicon are coated in the solidification vessel, and solidification is conducted subsequently.^With such procedures, even if water intrudes by some causes from the outside, since thin layers made of hydrophobic materials have a permanent water-repellent effect, the intrusion of water to the inside of the solidification material can be prevented as much as possible.^Accordingly, integrity of the solidification materials can further be improved.^(T.M.).

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
3993	Suitability of geopolymeric concretes for nuclear waste disposal	Concrete barriers are in essential role in most of the disposal concepts for nuclear waste.^As to the binders, the used high-quality, strong and dense concretes may be based both on the present types of cements and on new types of special cements.^One feasible special cement discussed in this literature report is the geopolymeric cement, which is, at its cleanest, a completely lime-free binder composed mainly of aluminium silicates.^However, in 1990 the lime-free aluminium silicate cement had not yet reached the stage of development required of a widely marketed factory product.^On the other hand, as an applicable product the development work started as early as in the 70s in France and in the USA has reached a blended cement consisting both of geopolymeric and Portland cements.^The main advantages of the geopolymeric concrete compared to the ordinary Portland cement concrete are based on richer and stronger chemical bonds of the cement stone.^The strong three-dimensional networks of bonds make the geopolymeric concrete stronger, denser and, above all, more ductile.^The geopolymeric concrete is particularly suitable for hazardous waste applications, since hazardous materials have been found to be locked inside the geopolymeric networks.^The properties of the geopolymeric cements and concretes and the implemented applications seem to be highly promising, but as to the nuclear waste applications there is not sufficient amount of reliable experimental information available yet.^The domestic cement and concrete industry will be in key position in accumulation of information and operating experiences.^(orig.).
4178	Performance of SYNROC under conditions relevant to repository disposal	A new method of making the SYNROC precursor has been developed which involves direct hydrolysis of ethanolic mixed titanium, zirconium and aluminium alkoxides into an aqueous slurry of barium and calcium hydroxides.^Summary of major findings of about 4000 leach tests carried out with SYNROC and the effect of process variables and leaching parameters on the chemical durability of SYNROC are presented in the paper.^Most of the results summarized were obtained for the early developed precursor.^Recent changes in the precursor and processing conditions have necessitated limited further testing to determine the effect, if any, of these changes on chemical durability.^Leach testing of SYNROC doped with fission products has continued with the aim of measuring long-term rates that cannot be reliably measured using non-radioactive specimens.^Release of {sup 137}Cs dominated the fission products source term.^The study on transuranic elements has concentrated on neptunium, the most leachable actinide element in SYNROC, short-term release is dominated by the initial spike from Cs and other readily soluble species at grain boundaries and in non-equilibrium phases.^Long-term release is probably controlled by matrix solubility.^A phenomenological model is developed which accounts for both short-and long-term release.^Research has also been carried out on the interactions between SYNROC leachates and granites.^The controlling mechanisms for fixation of radionuclides on granites appear to be absorption and ionic exchange.^(author).^7 refs, 7 figs, 4 tabs.

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
4179	Performance of high level waste forms and engineered barriers under repository conditions	The IAEA initiated in 1977 a co-ordinated research programme on the "Evaluation of Solidified High-Level Waste Forms" which was terminated in 1983.^As there was a continuing need for international collaboration in research on solidified high-level waste form and spent fuel, the IAEA initiated a new programme in 1984.^The new programme, besides including spent fuel and SYNROC, also placed greater emphasis on the effect of the engineered barriers of future repositories on the properties of the waste form.^These engineered barriers included containers, overpacks, buffer and backfill materials etc. as components of the "near-field" of the repository.^The Co-ordinated Research Programme on the Performance of High-Level Waste Forms and Engineered Barriers Under Repository Conditions had the objectives of promoting the exchange of information on the experience gained by different Member States in experimental performance data and technical model evaluation of solidified high level waste forms, components of the waste package and the complete waste management system under conditions relevant to final repository disposal.^The programme includes studies on both irradiated spent fuel and glass and ceramic forms as the final solidified waste forms.^The following topics were discussed: Leaching of vitrified high-level wastes, modelling of glass behaviour in clay, salt and granite repositories, environmental impacts of radionuclide release, synroc use for high--level waste solidification, leachate-rock interactions, spent fuel disposal in deep geologic repositories and radionuclide release mechanisms from various fuel types, radiolysis and selective leaching correlated with matrix alteration.^Refs, figs and tabs.
4363	Synroc	According contractor)most current high-level waste (HLW) management strategies, protection of the biosphere on timescales exceeding a few thousand years relies primarily upon the capacity of the geological barrier to minimize access of groundwater to the wasteform and to retard migration of dissolved radioactive species to the biosphere.^The primary objective of the Synroc strategy, reported in this paper, is to provide a wasteform which has a much greater resistance to leaching by groundwaters than borosilicate glass and which is capable of maintaining its integrity in suitable geological environments for periods exceeding one million years.^The creation of an independent immobilization barrier provides an additional high degree of security in the event of unexpected failure of the geological barrier.^A further objective of this strategy is to utilize a wasteform, Synroc, which is capable of generating public confidence in its ability to function effectively as an immobilization barrier for very long periods.^The reference form of Synroc consists of an assemblage of four main titanate minerals - zirconolite CaZrTi{sub 2}O{sub 7}, hollandite Ba{sub 1.2} (A1,Ti){sub 8}O{sub 16},perovskite CaTiO{sub 3} and titanium oxide(s) {times}{sub n}O{sub 2n {minus} 1}.^These minerals have the capacity to incorporate nearly all of the elements present in HLW into their crystal structures as solid solutions.^Similar minerals have survived in a wide range of natural geochemical-geological environments for up to 2000 million years.^It is this evidence of geological stability provided by nature, combined with experimental observations showing that these minerals are highly resistant to attack by hydrothermal solutions, which shows that Synroc should provide a superior method of immobilizing HLW.

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
4479	Synroc. Chapter 4	The primary objective of the Synroc strategy for high-level waste (HLW) management is to provide a wasteform which has a much greater resistance to leaching by groundwaters than borosilicate glass and which is capable of maintaining its integrity in suitable geological environments for periods exceeding one million years.^A furhter objective of this strategy is to utilize a waste form, Synroc, which is capable of generating public confidence in its ability to function effectively as an immobilization barrier for very long periods.^Section 2 of this chapter deals with the composition and preparation chemistry, mineralogy and microstructure of Synroc-C.^Section 3 discusses chemical durability: the effect of fundamental variables and fabrication parameters on leach rates.^Section 4 discusses radiation damage in teh various mineral components of Synroc-C caused by fast-neutrons, alpha-particles and other fission fragments.^Section 5 deals with thermal and mechanical properties.^In section 6 the ANSTO/ANU Synroc demonstration plant is described and section 7 is a comparative assessment of Synroc versus borosilicate glass.^(author).^185 refs.; 44 figs.; 23 tabs.^Includes 185 refs.
4525	Progress report on safety research of high-level waste management for the period April 1987 to March 1988	Researches on high-level waste management at the High Level Waste Management Laboratory and the Waste Safety Testing Facility Operation Division of the Japan Atomic Energy Research Institute in the fiscal year of 1987 are reviewed in the three sections of the report.^The topics are as follows: (1) On performance and durability of waste forms and engineered barrier materials, accelerated alpha radiation stability of glass form and Synroc has been investigated and stress corrosion cracking of canister materials was examined under simulated conditions.^(2) Sorption of /sup 237/Np on granite samples and behavior of iron during weathering of granites were studied with respect to safety evaluation for geological disposal.^(3) Actual waste was transported from the Tokai Reprocessing Plant and hot operation using the actual waste was initiated at WASTEF.
4665	New portable hand-held radiation instruments for measurements and monitoring	Hand-held radiation monitors are often used to search pedestrians and motor vehicles for special nuclear material (SNM) as part of a physical protection plan for nuclear materials.^ Recently, the Los Alamos Advanced Nuclear Technology group has commercialized an improved hand-held monitor that can be used for both physical-protection monitoring and verification measurements in nuclear material control and waste management.^ The new monitoring instruments are smaller and lighter; operate much longer on a battery charge; are available with NaI(Tiota) or neutron and gamma-ray sensitive plastic scintillation detectors; and are less expensive than other comparable instruments.^ They also have a second operating mode for making precise measurements over counting times as long as 99 s.^ This mode permits making basic verification measurements that may be needed before transporting nuclear material or waste outside protected areas.^ Improved verification measurements can be made with a second new hand-held instrument that has a stabilized detector and three separate gamma-ray energy windows to obtain spectral information for SNM quantity, enrichment, or material-type verification.

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
4921	Measurement techniques for very low activities of plutonium and transplutonium elements in urine	Systematic radiotoxicological monitoring of urine requires radiochemical and metrological methods suitable for the detection of very low excretion rates.^The operating conditions of sample treatment call for reliable, reproducible and highly efficient substoichiometric techniques, as well as alpha counting and alpha spectrometric facilities with very low background.^Depending on the radiotoxic substances sought (plutonium alone or in association with transplutonium elements), two different techniques are routinely used as Marcoule: separation of uranium with an anion-exchange resin or simultaneous determination of transuranium elements by the 'gross alpha' technique.^In both cases, the radionuclides to be measured are quantitatively entrained by a microprecipitate of lanthanum fluoride on a cellulose acetate filter of very low porosity.^This procedure has three basic advantages - excellent counting geometry, negligible self-absorption (weight of the precipitate 25 ..mu..g.cm/sup -2/) and good spectrometric resolution (35 keV).^The average chemical separation efficiency for both methods (verified by systematically adding excess amounts of /sup 242/Pu to the samples) is 90%.^The activity is measured by two different methods, depending Univ.the desired detection limits: solid scintillation using a ZnS disc or the partially depleted surface-barrier technique.^The efficiency obtained is simultaneously dependent on the choice of components, design of the measurement system and optimization of the signal-to-noise ratio.^By adjusting the measurement facilities it is possible to achieve a counting efficiency of 46% in the 4..pi.. geometry by the ZnS scintillation technique and 32% by the surface-barrier technique.^The selected counting times are 900 min for measurement with ZnS and 4320 min for the surface-barrier detector, which give detection limits of 0.55 mBq (15 fCi) and 0.11 mBq (3 fCi), respectively.
4961	Measurement of the sorption of actinides on minerals using microanalytical techniques	The use of advanced surface-analytical techniques to study the sorption of the actinides uranium and plutonium on to rocks and their constituent minerals, in the context of radioactive waste disposal, is described.^Nuclear microprobe analysis was used to quantify the extent of sorption of actinides via Rutherford back-scattering (RBS); data on the minerals on which sorption had occurred were provided by particle-induced X-ray emission.^Both surface and sub-surface concentrations of actinides were measurable.^Secondary-ion mass spectrometry (SIMS) was used to measure qualitatively the distribution of sorbed actinides and their penetration rates into minerals.^The equipment used at Harwell is described.^Complementary use of both techniques in parallel is highly advantageous; RBS is used to quantify actinide surface loadings, with limited lateral and depth resolution, but, allied to SIMS, which has excellent spatial resolution, samples can be analysed both quantitatively and with high spatial resolution.^Concentrations of uranium and plutonium sorbed on to minerals can be routinely determined with sensitivities down to 1 ng cm[sup -2].^The data obtained are used to identify the minerals in a rock that are important for actinide sorption.^(author).

Table A-2. R&D Activities with Unknown Determination (continued)

ID No.	Title	Objective/Abstract
4994	Colloid remediation in groundwater by polyelectrolyte capture	This paper describes an ongoing study to characterize groundwater colloids, to understand the geochemical factors affecting colloid transport in groundwater, and to develop an in-situ colloid remediation process.^The colloids and suspended particulate matter used in this study were collected from a perched aquifer site that has radiation levels several hundred times the natural background and where previous researchers have measured and reported the presence of radiocolloids containing plutonium and americium.^At this site, radionuclides have spread over several kilometers.^Inorganic colloids collected from water samples are characterized with respect to concentration, mineralogy, size distribution, electrophoretic mobility (zeta potential), and radioactivity levels.^Presented are the methods used to investigate the physiochemical factors affecting colloid transport and the preliminary analytical results.^Included below are a description of a colloid transport model and the corresponding computational code, water analyses, characterization of the inorganic colloids, and a conceptual description of a process for in-situ colloid remediation using the phenomenon of polyelectrolyte capture.
5171	Development of radionuclide migration monitor, (3). Measurement of radionuclide migration velocity	This report describes that the nondestructive method determining the radionuclide migration velocity in a soil layer is established incorporated with the previous reports (I,II) which are presented in terms of the nondestructive measurements of the distribution of radionuclide concentration in a soil layer.^The radionuclide mobility in a soil layer can be evaluated quantitatively by the radionuclide migration velocity obtained from the comparison of the distribution of radionuclide concentration in a soil layer with the solution of mass-transport equation.
5174	Development of radionuclide migration monitor, (2). Nondestructive measurement of radionuclide concentration distribution in aquifer soil sample	A nondestructive measuring system has been developed for the measurement of the radionuclide concentration distribution in an aquifer soil sample to measure the radionuclide mobility in the soil layer.^In this system, the counting rate distribution of ..gamma..-ray is measured by using detectors inserted into the aquifer soil sample, and the simultaneous equations whose coefficient matrix consists of theoretically calculated measuring efficiencies of detector are obtained.^The concentration distribution in the aquifer soil sample is determined by solving the simultaneous equations assuming that the concentration distribution decreases exponentially.^From the results of function tests, it is found that the radionuclide concentration distribution in an aquifer soil sample can be measured by this system.^The report describes the outline of the system and the results of function tests.
5228	Remote Excavation System	The Remote Excavation System (RES) is a military tractor, the Small Emplacement Excavator, which has been modified for tele-robotic operation. The puropose of this technology development effort is for buried waste retrieval for DOE and unexploded ordinance removal for the U.S. Army. Remote buried waste retrieval reduces potential human radiation exposure.

APPENDIX B

DOE R&D Tracking Database Analysis

Table B-1. DOE R&D Projects with Yes Determination [1]

Project ID	Project Title[2]	Project Description[3]
P/AMES—92215158201	INSTRUMENTATION AND DIAGNOSTICS (AIR TOXICS INSTRUMENTATION)	On-line monitoring of mercury (Hg) and hydrogen chloride (HCl) in coal gasification/combustion gases is needed for diagnostics and process control for advance coal conversion technologies. In this study, currently available methods for monitoring Hg . . .
P/AMES— TTPCH13C231	INFRARED ANALYSIS OF WASTES	The polymer encapsulation of low-level mixed waste is a promising method for producing a certifiable, low-volume immobilized waste, which is being developed in the DOE complex. The present grouting method for immobilization has difficulties . . .
P/ANL—000075	AQUEOUS BIPHASIC TREATMENT OF URANIUM CONTAMINATED SOILS	This task will continue to conduct a treatability study to determine the feasibility of using an Aqueous Biphasic Separation system for the selective removal of uranium from contaminated soils and from contaminated soils which have been . . .
P/ANL—000087	AEM EXAMINATION OF FERNALD SOILS	Characterize the speciation and mode of binding of uranium in soils found at the DOE facilities at Fernald, Ohio, using analytical electron microscopy and parallel electron energy loss spectroscopy. Provide this information to principal . . .
P/ANL—000109	PHOSPHATE BONDED CERAMICS FOR STABILIZING PROBLEM LOW-LEVEL MIXED WASTE	The goal of this project is to demonstrate chemically bonded phosphate ceramics as a low-temperature solidification and stabilization technology for . . .
P/ANL—000110	PHP SLAG CHEMISTRY AND SLAG/METAL PROCESSING	The purpose of this task is to define the design basis and validate the final design of the slag/metal removal and handling system to be built for the field-scale radioactive PHP system by SAIC and Retech, Inc. This task will ensure that the design . . .
P/ANL—000118	TUCS/PHOSPHATE IMMOBILIZATION	The Efficient Separations and Processing Integrated Program needs statement ES-3 requests technologies for stabilization of radionuclide and hazardous buried waste contaminants to reduce and/or eliminate the potential for migration of these . . .

[1] The same activity may appear more than once, especially if it spans more than one year and the project identification number (ID) changes over time.

[2]The project name is presented as it appears in the DOE R&D Tracking Database.

[3]The project description is truncated due to DOE R&D Tracking Database generation difficulties.

P/ANL—001542	DETERMINATION OF TRANSMUTATION EFFECTS IN CRYSTALLINE WASTE FORMS	The objective of this study is to characterize the effects of transmutation in a candidate waste form for cesium-137 by investigating samples of a cesium aluminosilicate mineral, pollucite, that have undergone . . .

Table B-1. DOE R&D Projects with Yes Determination (continued)

Project ID	Project Title	Project Description
P/ANL—001549	IMMOBILIZATION OF FISSIOIN PRODUCTS IN PHOSPHATE CERAMIC	The work supported under this project will focus on the development of chemically bonded phosphate ceramics as a solidification and stabilization (S/S) technology for waste streams containing specific fission products such as cesium, strontium, and . . .
P/ANL—001550	EX-SITU WASTE TREATMENT & PROCESSING SYSTEMS	The Fernald Environmental Management Project Vitrification Facility will produce an off-gas stream consisting of high concentrations of radon gas. There exists a need to effectively remove the radon from the off-gas prior to discharge to the . . .
P/BNL—NE-213	SUBSURFACE BARRIER TECHNOLOGIES	One of the major challenges facing the waste management community today is the remediation of contaminated groundwater. Contaminants such as Trichloroethylene, Chloroform, Chromium, Cadmium, Uranium, and Strontium have leached into the groundwater . . .
P/ID—AC07-93ID13170	AZ-PAK EVALUATION FOR MACRO ENCAPSULATION OF LOW-LEVEL RADIOACTIVE LEAD	Arrow Construction, Inc. Has proposed a MACRO encapsulation process for low-level radioactive contaminated lead called HAZ-PAK. The encapsulation process involves thermally butt-fusing the bottom end cap to a thick wall, high density polyethylene . . .
P/ID—FG07-91ID13042	RESEARCH AND DEVELOPMENT MONITORING	This project includes (1) hydrological studies in wells open through large intervals, (2) unsaturated zone contamination and transport processes, (3) surface water–groundwater interactions and regional groundwater flow, and (4) independent testing of . . .
P/ID—FG07-96ER62306	DEVELOPMENT OF IN SITU MICRO SENSOR FOR THE MEASUREMENT OF CHROMIUM AND URANIUM IN GROUNDWATER AT DOE SITES	The goal of this program is to develop, optimize and deploy a silicon-based micro machined stripping analyzer for field monitoring trace levels of chromium and uranium. Such systems will integrate the sample-handling steps and necessary chemical . . .
P/ID—FG07-96ER62318	REDUCTION AND IMMOBILIZATION OF RADIONUCLIDES AND TOXIC METAL IONS USING COMBINED ZERO VALENT IRON AND ANAEROBIC BACTERIA	Large groundwater plumes contaminated with toxic metal ions, including radionuclides, exist at several DOE facilities. Previous research indicated that both zero valent iron and sulfate reducing bacteria can yield significant decreases in . . .
P/INEL—1132	MICROBIAL CORROSION OF CONCRETE: SENSORS AND TEST CELLS	—
P/INEL—189A6359	LOW -LEVEL RADIOACTIVE DECONTAMINATION	Radial chemistry investigation to support NRC decontamination interests . . .

Table B-1. DOE R&D Projects with Yes Determination (continued)

Project ID	Project Title	Project Description
P/INEL—189A6876	FIELD LYSIMETER INVESTIGATION	Field testing of low-level radioactive waste forms in disposal environments at two sites using lysimeters. ..
P/INEL—2017	TESTING AND DEMONSTRATION OF A SUPER-CRITICAL CARBON DIOXIDE DECONTAMINATION TECHNIQUE	—
P/INEL—2027	SUBSURFACE RADIOLOGICAL ASSAY INSTRUMENTS	—
P/INEL—2033	POLYMERIC SOLVENTS FOR MINIMIZING POLLUTION IN CHEMICAL SYNTHESIS, SEPARATIONS, AND CLEANING OPERATIONS (V04-URC)	—
P/INEL—2037	PORTABLE PHOTON ANALYSIS SPECTROMETER (PPAS) FOR TRU AND X- & Y-RAY ASSAY	—
P/INEL—2083	GAS TRACER TEST FOR VERIFYING THE INTEGRITY OF STABILIZED OR ISOLATED BURIED WASTE	—
P/INEL—2086	USE OF DEPLETED URANIUM AND RADIOACTIVELY CONTAMINATED STAINLESS STEELS IN CASKS	—
P/INEL—2122	POLYETHYLENE SULFUR CEMENT: FEASIBILITY AS A WASTE ENCAPSULATION MATERIAL FOR LOW-LEVEL MIXED WASTE	—
P/INEL—2192	WASTE DRUM REFURBISHMENT	—
P/INEL—2194	MEMBRANE PROCESSES FOR THE RECOVERY OF RADIOACTIVE SPECIES FROM ACIDIC SOLUTIONS (URC)	—
P/INEL—2195	RADIONUCLIDE REMOVAL FROM CONTAMINATED SOIL AND DEBRIS LEACHATES BY THE APPLICATION OF 3M WEB TECHNOLOGY	—

Table B-1. DOE R&D Projects with Yes Determination (continued)

Project ID	Project Title	Project Description
P/INEL—2197	DEVELOPMENT, FIELD DEMONSTRATION, AND COMMERCIALIZATION OF VADOS ZONE SENSORS FOR CHARACTERIZATION, REMEDIATION, AND POST-CLOSURE MONITORING AT CONTAMINATED SITES	—
P/INEL—2248	AUTOMATIC IDENTIFICATION/ROBOTICS INTEGRATION FOR DRUM HANDLING AND AUTOMATED LABORATORY	—
P/INEL—2288	SHIELDED PH-TRU CONTAINER	—
P/INEL—2342	AN APPROACH TO ESTIMATING PLANT CONTAMINANT UPTAKE FOR RISK ASSESSMENT EXPOSURE MODELING	—
P/INEL—2344	DEVELOPMENT OF COATINGS FOR WASTE CONTAINER REFURBISHMENT	—
P/INEL—2349	CHARACTERIZE AND MODEL FINAL WASTE FORMULATIONS AND OFF-GAS SOLIDS FROM THERMAL TREATMENT PROCESSES	—
P/INE—3364	PLUTONIUM VITRIFICATION	—
P/INEL—95-233	DOE-SR/BENCH-SCALE ARC MELTER	This project is to design and construct a bench-scale arc melter for evaluating plasma arc processing of DOE waste. . .
P/INEL—ADSID06DD21	DEMO OF IMPROVED TECH AT ONGOING DECOMMISSIONING	Demonstrate waste retrieval in V tanks . . .
P/INEL—ADSID06DD21-CE	DEMO OF IMPROVED TECH AT ONGOING DECOMMISSIONING	Demonstrate waste retrieval in tanks . . .
P/INEL—ADSID06LF21	SITE CHARACTERIZATION, DEMO AND EVALUATION	This task will develop and evaluate innovative technologies for tru/mixed buried waste site assessment. The focus of this effort will be on non-invasive geophysical site characterization technology evaluation. The purpose of this task is to develop a ...

Table B-1. DOE R&D Projects with Yes Determination (continued)

Project ID	Project Title	Project Description
P/INEL—ADSID1001WN	WASTE OPERATIONS	Ensures Idaho Chemical Processing Plant waste processes will function as designed; treatment processes are optimized; support and optimize lab tests, pilot plant test, mockups.
P/INEL—ADSID121217	SUPER CRITICAL WATER OXIDATION	Hydro thermal oxidation.
P/INEL—ADSID141003	HIGH TEMPERATURE MELTER DEVELOPMENT	High temperature melter.
P/INEL—ADSID142001	FIXED HEARTH PLASMA TREATMENT PROCESS	Fixed hearth plasma demonstration.
P/INEL—ADSID142009	COOPERATIVE TELEROBOTIC RETRIEVAL	Cooperative telerobotic.
P/INEL—ADSID142013	TECHNOLOGY DEVELOPMENT	Technical deployment.
P/INEL—ADSID442001	DECONTAMINATION SYSTEM AND END EFFECTS	Decon systems and end effectors.
P/INEL—ADSID74MW51	PHP PILOT SCALE TESTING	This technical task plan describes a subset of work efforts under the plasma hearth process technology development project. The purpose of the overall project is to develop, test and evaluate a new concept for mixed waste treatment based . . .
P/INEL—ADSID751005	HOT SPOT RETRIEVAL CONVEYANCE DEMO	Hot spot retrieval.
P/INEL—ADSID751010	MAWS TECH DEMO	Maws technical demonstration.
P/INEL—ADSID752007	SCWO TEST BED	Hydro thermal oxidation.
P/INEL—ADSID75C221	INTEGRATED GEOPHYSICAL AND HYDROLOGIC CHARACTERIZATION OF FRACTURED ROCK SYSTEMS	This project seeks to develop a suite of reliable geological, geophysical, and hydrological tools that can be collectively employed to characterize contaminant flow and transport in fractured rocks. Specific components of this project include (a) . . .
P/INEL—ADSID76LF21	SITE CHARACTERIZATION, DEMO, AND EVALUATION	This project will develop and evaluate innovative technology for tru/mixed buried waste site assessment. Focus of this effort will be on non-invasive geophysical site characterization technology evaluation and face characterization implementation.
P/INEL—V004	POLYMERIC SOLVENTS FOR MINIMIZING POLLUTION IN CHEMICAL SYNTHESIS, SEPARATIONS, AND CLEANING OPERATIONS	—

Table B-1. DOE R&D Projects with Yes Determination (continued)

Project ID	Project Title	Project Description
P/LANL—402	HYBRID K-EDGE/XRF SOLUTION ASSAY INSTRUMENT	Develop and extend the hybrid K-edge/densitometer technique for measuring process solutions bearing U, Pu, and interferences to include low-level waste solutions. This will provide a single analysis method for assay of input, product, and waste . . .
P/LANL—413	INTEGRATED ASSAY OF UNMEASURED INVENTORY IN DRUMS	Develop an integrated assay capability for accountability measurements of previously unmeasured inventory in 30- and 55-gal drum shipping containers. The project will use a segmented gamma-ray scanner augmented with plutonium isotopic and . . .
P/LANL—420	COMBINED NEUTRON/GAMMA-RAY NDA INSTRUMENT	Develop and apply a method for integrating data from combined neutron and gamma-ray instruments for waste, scrap, and residue materials in a variety of container types and sizes. Provide a combined waste assay and verification function. To benchmark . . .
P/LANL—94803	DEVELOPMENT OF A LOW-LEVEL IN-LINE ALPHA COUNTER	With the increasing awareness of water contamination issues and the rising consequences of any form of contamination, real-time continuous monitoring is rapidly becoming a necessity. in particular, monitoring for the presence of any radioactive . . .
P/LANL—A813	HYBRID K-EDGE/XRF SOL ASSAY INSTRUMENT	Developed and extended the hybrid K-edge/densitometer technique for measuring process solutions bearing U, Pu, and interferences to include low-level waste solutions. Began the formal documentation process for hybrid K-edge/densitometer hardware, . . .
P/LANL—CX66	ANCHO CANYON STUDY	Conduct study to understand the fate and transport of heavy metals, radionuclides, and high explosives away from firing sites located at in Ancho Canyon, TA-39.
P/LANL—KB23	URANIUM WASTE AND RESIDUE CAPABILITY	La-ur-96-4015. Uranium recovery and processing: as part of the larger effort for materials storage and disposition activities related to the legacy uranium inventory at Los Alamos, this project includes support for demonstration of reprocessing . . .
P/LAN—M46D	MWFA TREATMENT TRAIN AND DEVELOPMENT	La-ur-96-3973. Improved transuranic (tru) and low-level waste assay accuracy is needed to meet increasingly stringent shipping and disposal regulations and is widely recognized as a top priority by doe waste generators and regulators. Conventional . . .

Table B-1. DOE R&D Projects with Yes Determination (continued)

Project ID	Project Title	Project Description
P/LANL—M46T	DEVELOPMENT OF REAL TIME MONITOR AIR-BORNE ALPHA	La-ur-96-3977. Staggering volumes of low-level radioactive waste await disposal. Thermal treatment such as incineration is currently accepted as the best method for reducing this tremendous volume of material, but concerns over effective emissions . . .
P/LBNL—4050	NEW LIGANDS BASED ON NATURAL COMPLEXING AGENT–K. R. RAYMOND	Develop selective complexing agents to reduce the actinide content and volume of both high-level and residual low-level waste at DOE sites.
P/LBNL—4670	RADIONUCLIDE WASTE STUDY	Experimental Investigation of Thermodynamic & Kinetic Properties of Chemical Species in Mixed Radionculide Waste . . .
P/LLNL—382	SUPPORT OF DOE FACILITIES IN IMPLEMENTATION, TESTING AND EVALUATION OF LLNL-DEVELOPED NDA TECHNIQUES	The objective of this project is to assist DOE sites in implementation of LLNL developed NDA technology; in particular, assist WSRC's H-Canyon facility; Livermore's Materials Management; LANL's TA-55 facility, and Hanford storage facilities.
P/LLNL—95-ERP-129	CERAMIC WASTE CONTAINERS FOR NUCLEAR WASTE DISPOSAL	—
P/LLNL—EHW—0018	PATHWAY ANALYSIS	The goals of the proposed project are to: (1) Compare Plutonium (Pu) transport, uptake, cycling and resuspension from the various source terms where Pu has been dispersed into the environment; (2) Determine the range and distribution of these . . .
P/LLNL—EHW—0070	PATHWAY ANALYSIS	The goals of the proposed project are to: (1) Compare plutonium (Pu) transport, uptake, cycling and resuspension from the various source terms where Pu has been dispersed into the environment; (2) Determine the range and distribution of these . . .
P/LLNL—EMW—0021	MIXED WASTE MANAGEMENT FACILITY AND ACCELERATED SITE CLEANUP INITIATIVE	The scope of the MWMF is to design, construct, and start-up pilot-scale facility to demonstrate and evaluate the operation and integration of technologies for the treatment of low-level, organic, and mixed waste; to demonstrate alternatives to the . . .
P/LLNL—EMW—0026	ENVIRONMENTAL REMEDIATION AND WASTE MANAGEMENT TECHNOLOGY DEVELOPMENT	The Environmental Technologies Program conducts fundamental and applied research to develop, demonstrate, and commercialize innovative technologies for solving environmental problems. Environmental Technologies Program projects/proposals involve a broad range of technical . . .

Table B-1. DOE R&D Projects with Yes Determination (continued)

Project ID	Project Title	Project Description
P/LLNL—MD004	PLUTONIUM IMMOBILIZATION PROCESS TESTING	Conduct a lab scale test to support plutonium immobilization processes and facilities as required for programmatic environmental impact statement preparation and identification of discriminators between proposed immobilization alternatives.
P/METC—AC21-93MC30172	CHARACTERIZATION OF RADIOACTIVE CONTAMINATION INSIDE PIPES WITH THE SEAMIST SYSTEM	The project objective is to demonstrate the feasibility of integrating low-energy gamma radiation detection with the ``SEAMIST'' instrument deployment system. The effort will demonstrate the modified SEAMIST system's ability to deploy instrumentation . . .
P/METC—AC21-93MC30173	WASTE INSPECTION TOMOGRAPHY: A FIELD-OPERABLE SCANNER FOR NONINVASIVE CHARACTERIZATION OF NUCLEAR WASTE CONTAINER	The project objective is to construct a transportable inspection system to characterize containers of radioactive waste by nondestructive evaluation and assay.
P/METC—AC21-94MC29249	ELECTROMAGNETIC MIXED-WASTE PROCESSING SYSTEM FOR ASBESTOS DECONTAMINATION	The overall objective of this project is to develop and demonstrate a cost-effective technology that decomposes asbestos, removes hazardous organic constituents from the decomposed asbestos, and removes radioactive and heavy metals from the . . .
P/METC—AR21-95MC32088	DEVELOPMENT OF AN ON-LINE, REAL-TIME ALPHA RADIATION MEASURING INSTRUMENT FOR LIQUID	The overall objectives are to develop a large surface area detector with multiple-jet-impingement mass transfer for maximum sensitivity and response speed, and to demonstrate the instrument's effectiveness through testing on a variety of DOE site(s). . .
P/ORAL—613035001	INEEL—EPICOR-II ION EXCHANGE RESIN STUDIES	Provide services of installation and monitoring of lysimeters containing TM1-2 EPICOR-II resin waste forms at ORNL.
P/ORNL—MD005	PLUTONIUM IMMOBILIZATION PROCESS TESTING	Conduct a lab scale test to support plutonium immobilization processes and facilities as required for programmatic environmental impact statement preparation and identification of discriminators between proposed immobilization alternatives.
P/PNNL—10603	8016 SPECIAL WASTE FORM LYSIMETER	Perform field and laboratory leaching tests of solidified commercial low-level wastes. . .
P/PNNL—16706	HANFORD WASTE VITRIFICATION PLANT	PNL shall develop vitrification system technology for application in the Hanford Waste Vitrification Plan . . .
P/PNNL—18689A	8545 DC ARC PLASMA AND CERAMIC MELTER	Develop a technology status report describing the technology proposed in RL-920015 in sufficient detail to enable evaluation by the BWID Technology Evaluation Group.

Table B-1. DOE R&D Projects with Yes Determination (continued)

Project ID	Project Title	Project Description
P/PNNL—18689D	8545 DC ARC PLASMA AND CERAMIC MELTER	Develop a technology status report describing the technology proposed in RL-920015 in sufficient detail to enable evaluation by the BWID Technology Evaluation Group.
P/PNNL—18719B	8524 CHEM SEPERATIONS FOR NUC WASTE DISPOSAL	To identify and develop new chemical separations/ treatment options that could result in major cost reductions at Hanford and other DOE sites for the disposal of radioactive waste.
P/PNNL—19153	8574 IN SITU DETERMINATION OF RADIO AND METAL CONTAMINANTS	Design and fabricate an in-stitu sensor capable of measuring both radioactive and stable metal contaminants in and around a landfill.
P/PNNL—19307	8527 IN SITU MAPPING OF SOIL RADIONNUCLIDES	Develop new radiation monitoring equipment for surface and subsurface mapping of soil radionuclides . ..
P/PNNL—19949B	PRETREATMENT PROGRAM MANAGEMENT	Develop alternate technologies for pretreatment of low- and high-level waste from production for the treatment of Hanford tank wastes.
P/PNNL—19989	DIRECT MEASUREMENT OF SR-90 IN SURFACE SOILS	Develop new research radiation monitoring equipment for surfact detection of Sr-90.
P/PNNL—20066	FY-1993 PLANNING FOR DST RETRIEVAL	Develop technologies for retrieval of high- and low-level waste from SST and DST tanks for the treatment of Hanford tank wastes.
P/PNNL—20138	COMPACT PROCESSING UNIT DEMONSTRATION	Will accomplish the development, design, and demonstration of compact processing units for cesium removal and elimination of organic constituents.
P/PNNL—20164A	VITRIFICATION OF ROCKY FLATS WASTE	Conduct glass development using surrogate and actual low-level radioactive waste.
P/PNNL—20193B	LOW-LEVEL WASTE IMMOBILIZATION	Hanford Grout Technology Provide technology developmt support to immobilize selected pre-treated High-Level Waste, Low-Level Waste, and Transuranic waste at Hanford, . . .
P/PNNL—20331B	ORNL STORAGE TANK SLUDGE MOBILIZATION AND MIXING	Provide computer simulations of sludge mobilization and tank homogenization in horizontal storage tanks
P/PNNL—20347A	CONDUCT TECHNICAL REVIEW OF CURRENT NUREG/CR-5512	Conduct technical review of current NUREG/CR-5512 groundwater model component and develop supplemental technical rationale and methodology for using independent groundwater models for a hierarchy of screening assessments . . .

Table B-1. DOE R&D Projects with Yes Determination (continued)

Project ID	Project Title	Project Description
P/PNNL—20391A	CHARACTERIZATION OF RADIONUCLIDE-CHELATING AGENT COMPLEXES	Characterize and identify radionuclide organic complexes in decontamination wastes from operating nuclear power stations . . .
P/PNNL—20864B	ORNL PIT-1 ISV DEMONSTATION	Provide technical services for in Situ Vitrification treatability study at Oak Ridge National Laboratory seepage pit #1 . . .
P/PNNL—20888A	HIGH RESOLUTION IMAGING OF BURIED WASTE	Conduct high-resolution imaging of buried waste using holographic impulse radar array . . .
P/PNNL—21172A	TECHNICAL SUPPORT FOR GEIS ON DECOMMISSIONING OF NUCLEAR FAS	Develop a technical model for estimating the costs of decontaminating concrete surfaces as a function of the allowable residual radioactivity level at licensed nuclear facilities . . .
P/PNNL—21178A	TECHNICAL SUPPORT IN LOW-LEVEL WASTE SITE PERFORMANCE ASSESSMENT	Provide technical services for conducting low-level waste (LLW) facility performance assessments and develop guidance for LLW performance assessment related to modeling hydrology surface water flow and transport, and LLW site infiltration . . .
P/PNNL—21342	IN SITU VITRIFICATION ROLL-UP	Multi-faceted project that will perform applied R&D to address key issues remaining on the ISV technology; i.e., Melt low-alkali soils; enhance depth capabilities; develop an understanding of behavior of volatiles in and around melt.
P/PNNL—21405A	HIGH-LEVEL WASTE VITRIFICATION	PNL Vitrification Technology Development Project Provide technology development support to immobilize select pre-treated HIGH-LEVEL Waste, Low-Level Waste and Transuranic waste at Hanford . . .
P/PNNL—21408B	SIX PHASE SOIL HEATING/RF	Provide engineering design services for the Six-Phase Soil Heating system to be used in conjunction with in situ Vitrification at T-3, Environmental Test Site . . .
P/PNNL—21415	COMPUTER MODELING BEHAVIOR OF RADIONUCLIDES IN SOIL AS	Computer Modeling Behavior of Radionuclides in Soil AS Conduct computer Modeling of the Behavior of Radionuclides in a Soil Aquatic System.
P/PNNL—21778A	LOW-LEVEL WASTE PERFORMANCE ASSESSMENT	Measure radionuclide leaching rates from ion exchange resin by groundwater, and determine radionuclide speciation and pH-Eh distributions . . .
P/PNNL—21792	ASSESSMENT OF BEHAVIOR AND FATE OF RADIONUCLIDES	Perform an assessment of the behaviour and fate of radionuclide contaminants during movement through municipal sewage processing systems . . .

Table B-1. DOE R&D Projects with Yes Determination (continued)

Project ID	Project Title	Project Description
P/PNNL—22143A	SALT SPLITTING OF SODIUM DOMINATED RADIOACTIVE WASTE USING	Salt Splitting of Sodium Dominated Radioactive Waste Using Ceramic Membranes. Perform electrochemical separations of surrogate solutions containing salt, strontium, cesium, etc., into sodium hydroxide solution.
P/PNNL—22264B	ISV FOR BURIED WASTE AT ORNL.	Design engineering-scale in situ Vitrification tests tests for application of ISV to buried waste sites; prepare appropriate Environment, Safety and Health documents to cover engineering-scale tests to be done in FY-1995 . . .
P/PNNL—22456A	ELECTRODE DEVELOPMENT FOR WASTE VITRIFICATION	Electrode Development for Waste Vitrification Investigate electrode protection methods, lab-scale tests using different electrode materials and passivation. Will be completed in FY-1995.
P/PNNL—23109A	IN SITU VITRIFICATION OF SRS SOILS	Conduct engineering-scale in situ vitrification tests on Savannah River Site soils . . .
P/PNNL—23182A	WASTE FORMS BASED ON SEPARATIONS MEDIA	Develop and evaluate alternate waste forms for problem waste feeds that will be generated as a result of radionuclide separations . . .
P/PNNL—23655A	EXPEDITED SITE CHARACTERIZATION	Demonstrate Beta Scintillation Sensor at Client's Site for Characterization of Strontium-90 in Conveyor-Transported Soils . . .
P/PNNL—24539A	SELECTION OF KD VALUES PROJECT	Prepare guidance on selecting Kd (parameter coefficient for computer coding) for values for technical staff responsible for prioritizing/ remediating/ cleaning up sites and waste management decisions . . .
P/PNNL—24769A	GROUND WATER FLOW AND TRANSPORT ANALYSIS	Utilize and adapt infiltration and transport analysis methods from DOE laboratory developed low-level waste infiltration evaluation methodology and ongoing groundwater monitoring studies to support decommissioning plan reviews . . .
P/PNNL—24822A	PROVIDE EXPERIMENTAL RADIONUCLIDE SOLUBILITY DATA AND	Provide experimental radionuclide solubility data and information to calculate source term released from Site Decommissioning Program Management waste slags for use in waste disposal site performance assessments . . .
P/PNNL—24875A	PERMEABLE TREATMENT ZONES	Optimize the effectiveness of permeable barriers to be used in vadose or aquifer sediments to minimize contaminant migration.
P/PNNL—26238A	PERFORM MONITORING OF LYSIMETERS AND FINAL DRAINAGE DURING	Perform monitoring of lysimeters and final drainage during the summer of 1996 . . .

Table B-1. DOE R&D Projects with Yes Determination (continued)

Project ID	Project Title	Project Description
P/PNNL—26512A	GLASS DEVELOPMENT FOR LANL	Develop and characterize glass formulations for simulated evaporator bottoms wastes . . .
P/PNNL—MD007	PLUTONIUM IMMOBILIZATION PROCESS TESTING	Conduct a lab-scale test to support plutonium immobilization processes and facilities as required for programmatic environmental impact statement preparation and identification of discriminators between proposed immobilization alternatives.
P/SNL—0688	LSFA CONTAINMENT	3 tasks include 1) Alternative Landfill Cover Demonstration, 2) Capillary Barrier, and 3) Smart Geomembrane. Task 1 provides alternatives to EPA cover designs that will be less expensive and perform more effectively in arid climtes. Task 2 will . . .
P/SNL—0689	LSFA VERIF ICATION AND MONITOR	This project includes the design, installation and field test of a tracer verification and monitoring system, tracer tests, and the installation of a monitoring system at a barrier test facility for subsurface barriers.
P/SNL—1076	WASTE FORM PERFORMANCE ASSESSM	Waste form performance assessment
P/SNL—3515090000	USING A PREHISTORIC WASTE DISPOSAL SITE AS A NATURAL ANALOGUE FOR THE VALIDATION OF RISK ASSESSMENT GROUND WATER FLOW AND TRANSPORT MODELS	—
P/SNL—3791	WASTE MIN PROGRAM MANAGMEMT AND DEVELOPMENT	SNL/NM waste minimization program.
P/SNL—3801	LLW DISPOSAL–CA	LLW disposal–CA
P/SNL—3802	MW DISPOSAL–CA	MW disposal–CA
P/SNL—3805	LLW STORAGE–CA	LLW storage–CA
P/SNL—3806	MW STORAGE–CA	MW storage–CA
P/SNL—3809	LLW TREATMENT–CA	LLW treatment–CA
P/SNL—3810	MW TREATMENT–CA	MW treatment–CA
P/SNL—3813	WASTE MIN PLANNING–CA	Waste Min planning–CA
P/SNL—3861	HQ WASTE MIN PROJECTS–CA	HQ waste min projects–CA

Table B-1. DOE R&D Projects with Yes Determination (continued)

Project ID	Project Title	Project Description
P/SNL—4779	VADOSE ZONE MONITORING	The purpose of this effort is to develop methods for characterizing and monitoring containment transport in the vadose zone with deep-water tables. This includes the evaluation of transport mechanisms of volatile organic compounds and tritium in the . . .
P/SNL—4781	SUBSURFACE BARRIER EMPLACEMENT	This TTP describes work to evaluate, test, and demonstrate the emplacement of horizontal barriers beneath waste sites from directionally drilled boreholes, using advanced barrier materials. A review of potential barrier emplacement techniques will be . . .
P/SNL—4782	ALTERNATE. LANDFILL COVER DEMONSTRATION	This project will demonstrate the capabilities of alternative landfill cover components and systems to provide long-term containment of waste buried in mixed-waste landfills that are located in arid/semi-arid western climate regions. The objectives . . .
P/SNL—4783	MEASUREMENT WHILE DRILLING	The aim of this project is to develop, demonstrate, and evaluate a measurement-while-drilling system within new boreholes for the characterization of buried radioactive and mixed wastes. The end product of this effort will be a commercially viable . . .
P/SNL—A1764	PERFORMANCE ASSESSMENT METHODOLOGY DEVELOPMENT–LLW DISPOSAL SAFETY ASSESSMENT	Development of low-level waste performance assessment methodology.
P/SNL—L1153	DEVELOPMENT AND VALIDATION OF LOW-LEVEL WASTE PERFORMANCE ASSESSMENT METHODOLOGY	Development and validation of low-level waste performance assessment methodology.
P/SNL—W6574	IMPLEMENTATION OF LOW-LEVEL WASTE PERFORMANCE ASSESSMENT METHODOLOGY	This program involves development and implementation of a methodology for evaluating sites identified in the NRC's site management and decommissioning plan.
P/SRTC—9500131003	IN SITU REMEDIATION BY ELECTROKINETICS	This project will develop the electrokinetics for the removal of heavy metal contamination (primarily mercury) from soils at the Savannah River Site. A field demonstration was conducted in FY-1994. in FY-1995 the field demonstration continued for . . .

Table B-1. DOE R&D Projects with Yes Determination (continued)

Project ID	Project Title	Project Description
P/SRTC—9500131004	IN-TANK SLUDGE INTERFACE DETECTION/TIME	The objective of this development project was to provide a liquid-level measurement and moisture monitoring tool for use in high-level waste tanks. This technique may be deployed in a liquid observation well, or directly in tank waste. It may be able . . .
P/SRTC—9500132001	REMOTE VIEWING SYSTEMS	The objective of the Underground Storage Tank Integrated Demo program is to develop, demonstrate, test, and evaluate robotics and remote technology for environmental restoration of DOE weapons facilities Underground Storage Tanks and to . . .
P/SRTC—9500132004	VIT PROCESS LIMITS TESTING	The focus of this project is to conduct surrogate and hot bench-scale studies (at Savannah River Site and Russia) and pilot-scale surrogate vitrification studies (at Clemson) and pilot-scale radioactive tests (in Russia) on high organic content and . . .
P/SRTC—9500132005	ELECTROCHEMICAL DESTRUTION OF NITRATES	Electrochemical treatment processes were developed for the destruction of organic compounds and nitrates/nitrites and the removal of other hazardous species from liquid waste stored throughout the DOE complex. The development program consisted of . . .
P/SRTC—9500132007	ADVANCED CHEMICAL SEPARATIONS AT SRS	The task consisted of testing of Extended Sludge Processing (ESP) developed separations materials for application to SRS tank and other radioactive wastes, testing of an ESP developed material for separation and concentration of radioisotopes from . . .
P/SRTC—9500141023	OFF-GAS TREATMENT SAMPLING AND ANALYSIS	There are an estimated 3.0 million pounds of residual chlorinated solvents in the sediments and groundwater in the A/M Area of SRS. These chlorinated solvents are some of the most prevalent contaminants found in the subsurface across the country, . . .
P/SRTC—9500141024	SRS TRITIUM ANALYSIS SYSTEMS	The Environmental Restoration Department of the Solid Waste Management and Environmental Restoration Division is responsible for contaminated granter remediation at SRS. As a part of that effort, analysis of surface groundwaters are . . .
P/SRTC—9500142002	AQUEOUS PHASE CATALYTIC EXCHANGE	A U. S. Manufactured catalyst was evaluated for use in detritiation of waste water from SRS and other DOE facilities. The catalyst of most interest is manufactured by Hamilton Standard, a subsidiary of United Technologies, and thus would be available . . .
P/SRTC—9500151002	HOT CELLS SELECTIVE LEACHING	Studies using the ACT-DECON process with sludge simulants and Hanford Tank Sludge have found the process is effective in removing radioactive elements from the sludge [1]. The ACT-DECON process uses oxidative carbonate chemistry and a chelating . . .

Table B-1. DOE R&D Projects with Yes Determination (continued)

Project ID	Project Title	Project Description
P/SRTC—9500152001	COMPACT MELTER VITRIFICATION DEMONSTRATION	This task was the Mixed Waste Integrated Program funding portion of the vitrification expedited demonstration with three major thrusts: (1) completing the 1,000 Kg hot treatability studies of ORR mixed wastes in support of the field-scale mobile . . .
P/SRTC—9508011102	SOLID-WASTE VERIFICATION BY DIGITAL RADIOGRAPHY	The WSRC Solid Waste Department requested SRTC to conduct a demonstration of their lens-coupled, area-array digital radiography system capability to image the contents of B-25 burial boxes. The boxes are 4 X 4 X 8 ft and made of carbon steel. There . . .

Table B-2. DOE R&D Projects with Unknown Determination[1]

Project ID	Project Title[2]	Project Description[3]
P/AL—FC04-94AL98934	TRP NO. 1031 ASSISTIVE DEVICES: NEW MEXICO TECHNOLOGY DEPLOYMENT PILOT PROJECT	This is a project to refine and demonstrate a new model for accelerating technology extraction and deployment from federally owned technical institutions. The objective of this program is to identify and extract transition technology from existing . . .
P/AL—FG04-91AL74167	CARLSBAD ENVIRONMENTAL MONITORING AND RESEARCH PROGRAM	The Carlsbad Environmetal Monitoring and Research Program was created to provide an independent source of environmental data in the vicinity of the Waste Isolation Pilot Plant. Baseline data will be used to study the variation of concentrations of . . .
P/AMES—90380002	CHEMICAL SEPARATIONS AND ANALYSES	This project includes research in the following areas: Analytical Separations; Analytical . . .
P/AMES—96-06	ELECTROCHEMICALLY TRANSFORMABLE EXTRACTION PHASES FOR THE SEPARATION OF THE CRITICAL METAL ION CONSTITUENTS IN HIGH-LEVEL WASTES	—
P/AMES—TTPCH13C211	SENSING OF HEAD SPACE GASES AND CONTINUOUS IN SITU MONITORING	This project will not be funded in FY-1996 based on overall consensus of the technical and programmatic reviewers. This was a high-risk project in whch the overall anticipated payoffs have not been realized. The research team is excellent, but it is . . .
P/AMES—TTPCH14C212	ACOUSTIC CHARACTERIZATION OF WASTES IN DOUBLE SHELL USTS	—
P/AMES—TTPCH15C241	ICP-MS FOR ANALYSIS OF MICROLITER SAMPLES AND SOLIDS	The speed, accuracy and precision of inductively coupled plasma-mass spectrometry (ICP-MS) will be improved for determination of stable elements and radionuclides. A new nebulizer called monodisperse dried microparticulate injections (MDMI) will be . . .

[1] The same activity may appear more than once, especially if it spans more than one year and the project identificaiton number (ID) changes over time.

[2] The project name is presented as it appears in the DOE R&D Tracking Database.

[3] The project description is truncated due to DOE R&D Tracking Database generation difficulties.

Table B-2. DOE Projects with Unknown Determination (continued)

Project ID	Project Title	Project Description
P/ANL—000070	TECHNICAL SUPPORT TO INNOVATIVE AND UNIVERSITY RESEARCH STAFF (EM53.1).	This task provides comprehensive technical support (including management of specific programs such as the Hazardous Substance Research Centre Program) to the Director of Innovative and University Research staff (EM53.1). Task activities are in three . . .
P/ANL—000071	ANL PFA METALS AND RADIONUCLIDES PRODUCT LINE	This effort will be carried out as two distinct tasks by Argonne National Laboratory in cooperation with other government and private organizations. Task 1 will attempt to improve the current MAG-SEP^{SM} treatment system through basic . . .
P/ANL—000077	CPT VADOSE ZONE TESTING FOR SUBSURFACE CONTAMINANTS	This study was conducted to further improve the specific methodologies for using Cone Penetrometer Technology (CPT) to characterize soil contamination. The CPT program for the characterization of soil contamination was conducted at locations where . . .
P/ANL—000079	DEVELOP CONTINUOUS EMISSION MONITORING FOR THERMAL TREATMENT	This task addresses the need mandated by the Clean Air Act of 1990 to monitor air toxics. The objective is to complete the development of a Fourier Transform Infrared (FTIR) spectrometer combined with a heated long-path cell as a continuous emission . . .
P/ANL—000089	IN SITU GROUNDWATER TREATMENT USING MAGNETIC SEPARATION	The purpose of this effort is to demonstrate the technical and economic feasibility of the MAG-SEP groundwater remediation process. This process was developed by Bradtec in conjunction with Barrier Member Containment, Inc. This remediation system . . .
P/ANL—000091	MAWS COMPOSITIONAL ENVELOPE	Most DOE sites have a large variety of predominantly inorganic waste streams (e.g. Sludges, ashes, contaminated soils/water, and transite/asbestos) to which the MAWS concept can be applied and has potential for significant cost savings resulting from . . .
P/ANL—000092	EXTENSION OF MAWS APPROACH TO GLASSY SLAG FINAL WASTE FORMS	Many DOE sites have large volumes of waste which may not be amenable to disposal in glass waste forms. These wastes contain large amounts of scrap metals, high contents of elements which form crystals within the waste form (such as Cr, Ni, Ti, Fe, . . .

Table B-2. DOE Projects with Unknown Determination (continued)

Project ID	Project Title	Project Description
P/ANL—000096	ADVANCE CHEMICAL SEPARATION PROCESSES-CLEAN OPTION STRATEGY	The objectives of this project are two-fold: (1) develop and test new Tc-selective resin for the removal of technetium from alkaline waste and a new Cs-Sr selective ion exchange resin for the simultaneous removal of cesium and strontium also . . .
P/ANL—000097	BIPHASIC SYSTEMS FOR RADIOACTIVE WASTE PRETREATMENT	The objective of this task is to determine the feasibility of using Aqueous Biphasic Separation (ABS) systems based on polyethylene glycols (pegs) for the selective extraction and recovery of I, Se, and Tc from caustic solutions containing high . . .
P/ANL—000098	PLASMA HEARTH PROCESS RADIOACTIVE TESTING	The DOE has large volumes of diverse mixed waste; storage and disposal options are lacking, as well as treatment capability and capacity. Argonne will apply its shielded and alpha-qualified facilities to development and demonstration of technologies . . .
P/ANL—000113	IMPROVED CONCRETE CUTTING METHODS	This task will develop improved concrete cutting methods and technologies for the dismantlement of contaminated concrete structures.
P/ANL—000114	PORTABLE CONCENTRATOR FOR PROCESSING PU-CONTAINING SOLUTIONS	This activity will investigate the activity of a specific screw concentrator system for use as a portable, modular treatment for plutonium-containing liquids. The purpose will be to identify the areas of applicability of an evaporation/concentration . . .
P/ANL—000393	CHEMICAL SCIENCES: CHEMICAL SEPARATIONS SCIENCE	The objectives of this program are (1) to develop new and improved reagents that may be applied to help solve major problems in environmental remediation and waste management and (2) to elucidate the basic chemistry involved in utilizing these new . . .
P/ANL—000400	CHEMICAL SCIENCES: HEAVY ELEMENTS CHEMISTRY RESEARCH	The objective of this program is the determination, modeling, and prediction of the chemical and physical properties that are characteristic off-elements (lanthanides and actinides) and their compounds. The relationship between the structural and . . .

Table B-2. DOE Projects with Unknown Determination (continued)

Project ID	Project Title	Project Description
P/ANL—000446	APPLICATION OF AQUEOUS BIPHASIC EXTRACTION TO RADIOACTIVE WASTE	Aqueous biphase extraction systems will be developed as a means of treating solid radioactive wastes. The separation concept involves the selective partitioning of colloid-size particles between two immiscible aqueous phases. Wet grinding of . . .
P/ANL—000553	ENVIRONMENTAL RESEARCH: RADIONUCLIDE SPECIATION IN GROUNDWATER SYSTEMS	The interactions of multivalent actinides with microbes are being investigated. Both 1) the effects of actinides and actinide-organic complexes on the biodegradation of organic chelating agents and 2) the effects of the biodegradation processes on . . .
P/ANL—000617	SOL-GEL TECHNOLOGY FOR REFRACTORIES	The goal of this program is to continue the collaborative effort between Magneco/Metrel and Argonne National Laboratory in the area of refractory development (with improved properties) using a sol-gel technology. These refractories will be . . .
P/ANL—001528	SUPERCONDUCTING OPEN-GRADIENT MAGNETIC SEPARATION FOR THE PRETREATMENT OF RADIOACTIVE OR MIXED WASTE VITRIFICATION FEEDS	Scientists need to gain a better understanding of the magnetic separation processes that can be used to separate deleterious constituents (crystalline, amorphous, and colloidal) in vitrification feed streams for borosilicate glass production without . . .
P/ANL—001541	IN SITU SPECTRO-ELECTROCHEMICAL STUDIES OF RADIONUCLIDE CONTAMINATED SURFACE FILMS ON METALS AND THE MECHANISM OF THEIR FORMATION AND DISSOLUTION	The aim of this research is to gain a fundamental understanding of the structure, composition, and mechanism of formation of radionuclide-containing surface films on metals that are relevant to the problem of decontamination of piping systems and . . .
P/ANL—001545	PHOSPHATE-BONDED CERAMIC WASTE FORMS	The goal of this work is to develop chemically bonded ceramics for low-temperature treatment and stabilization of DOE problem low-level mixed waste streams as identified by the Mixed Waste Focus Area. Effort is being directed toward . . .
P/ANL—001547	ULTRASONIC SENSORS FOR IN SITU MONITORING OF PHYSICAL PROPERTIES	The objective of this project is to develop ultrasonic sensors for in situ monitoring of physical properties of radioactive tank waste. Initial focus of this program is on development of sensors for fluid viscosity and volume-percent solids . . .

Table B-2. DOE Projects with Unknown Determination (continued)

Project ID	Project Title	Project Description
P/ANL—001551	PHOSPHATE BONDING WITH HARMONIC COMPACTION	The objective of this project is to evaluate, plan, and possibly demonstrate a combined technology of phosphate bonding chemistry and the harmonic compaction technology developed by Ryan and Murphy Inc. to treat mixed wastes. Efforts will . . .
P/ANL—001552	ANL IMMOBILIZATION	One of the technologies being developed by the Tank Focus Area (TFA) is removal of cesium from the soluble fraction of the high-level waste at Hanford and solidification of the separated cesium in a form suitable for disposal. Currently under . . .
P/ANL—92111	BASIC RESEARCH AND DEVELOPMENT IN WASTE MANAGEMENT: SEPARATION SCIENCES	Develop new and improved separations applicable to waste partioning.
P/ANL—94-167R1	WASTE MANAGEMENT OF CHLOROFLUOROCARBONS	—
P/ANL—94076	FUNDAMENTAL STUDIES OF HAZARDOUS METAL-ION SEPARATIONS CHEMISTRY	Studies of fundamental toxic-metal-ion separations chemistry of water-soluble chelating polymers; studies of radium, cesium and strontium selective complexing agents; determination of the role of organic solvent in selective alkali and alkaline earth . . .
P/ANL—95-020N	DEVELOPMENT OF LASER TECHNOLOGY FOR THE DECONTAMINATION OF SURFACES	—
P/ANL—95-030N	DEVELOPMENT OF ANALYTICAL METHODS TO PERFORM WASTE CHARACTERIZATION OF ACTIVATED METALS	—
P/ANL—95-050N	DEVELOPMENT OF PERFORMANCE ASSESSMENT CAPABILITY FOR WASTE DISPOSAL	—
P/ANL—95-054N	REMOVAL OF LITHIUM OXIDE AND LITHIUM FROM LITHIUM CHLORIDE BASED PROCESS SALTS	—
P/ANL—95-095N	BENEFICIAL USES FOR NUCLEAR WASTE	—
P/ANL—96-205	METAL VOLATILITY IN WASTE PROCESSES	—

Table B-2. DOE Projects with Unknown Determination (continued)

Project ID	Project Title	Project Description
P/ANL—96-210	EXAFS INVESTIGATION OF TRANSMUTATION EFFECTS ON WASTE FORM STABILITY	—
P/ANL—96-254	STABILIZATION OF RADIONUCLIDES IN CERAMIC WASTE FORMS BY A NOVEL ELECTROCHEMICAL APPROACH	—
P/BNL—1312913150	DESIGN STUDY FOR UPGRADE OF A FACILITY FOR PROCESSING LOW-LEVEL LIQUID RADIOACTIVE WASTE	—
P/BNL—13149	BENCH-SCALE/PILOT SCALE TREATABILITY STUDY	
P/BNL—4736	BARRIER MATERIALS EVALUATION OF WASTE FORMS DUMPED IN THE KARA AND BARENTS SEAS	—
P/BNL—4736-OLD13173-NEW13180-NEW	BARRIER MATERIALS EVALUATION OF WASTE FORMS DUMPED IN THE KARA AND BARENTS SEAS	—
P/BNL—86021	LOW TEMPERATURE GLASSES FOR SINGLE SHELL TANK WASTES	—
P/BNL—87726	POLYETHYLENE ENCAPSULATION OF ION EXCHANGE-RESIN WASTES	—
P/BNL—87729	A STUDY OF HIGH-LEVEL RADIATION WASTE MATERIAL TRANSMUTATION USING AN ACCELERATOR	—
P/BNL—87729-OLD87730-NEW	A STUDY OF HIGH-LEVEL RADIATION WASTE MATERIAL TRANSMUTATION USING AN ACCELERATOR	—
P/BNL—AS-237-BPSD	BIODEGRADATION OF SELECTED ORGANIC COMPOUNDS AND COMPLEXING AGENTS OF RADIONUCLIDES AND METALS	The objective of this research is to determine the mechanisms of microbial degradation of organic complexing agents which mobilize radionuclides and toxic metals at contaminated sites. The persistence of chelating agents in the disposal . . .

Table B-2. DOE Projects with Unknown Determination (continued)

Project ID	Project Title	Project Description
P/BNL—AS-237-BPSD005042	BIODEGRADATION OF SELECTED ORGANIC COMPOUNDS AND COMPLEXING AGENTS OF RADIONUCLIDES AND METALS	The objective of this research is to determine the mechanisms of microbial degradation of organic complexing agents which mobilize radionuclides and toxic metals at contaminated sites. The persistence of chelating agents in the disposal environment . . .
P/BPO—DE-FC21-95MT30081	DEVELOPMENT OF A METHOD FOR TREATMENT AND UNDERGROUNDDISPOSAL OF NORM	Develop and demonstrate a method for treatment and underground disposal of naturally occurring radioactive materials resulting from natural gas production.
P/CH—FC02-94CE41107	TECHNOLOGY TRANSFER AND EDUCATIONAL ACTIVITIES IN AREA OF INDUSTRIAL WASTE REDUCTION AND POLLUTION PREVENTION	CWRT and DOE will cooperatively sponsor research in areas of waste reduction and pollution prevention. The results of this research will be shared with industry and academia through publications and educational seminars.
P/CH—FG02-86NE37967	UNIVERSITY RESEARRCH PROGRAM IN ROBOTICS FOR ENVIRONMENTAL RESTORATION AND WASTEMANAGEMENT	This program will consist of applied research in support of the Robotics Technology Development Program.
P/CH—FG02-86NE37968	ROBOTICS FOR ADVANCED NUCLEAR REACTORS	The primary objective of this research is to investigate issues associated with the development of sensor-based robotic systems of specific applications in the area of environmental restoration and waste management. This work is done in conjunction . . .
P/CH—FG02-89ER60846	GROUNDWATER COLLOIDS: THEIR MOBILIZATION FROM SUBSURFACE DEPOSITS	The primary goal of this research is to develop a quantitative understanding of the roles of colloids in subsurface environments. Such roles include enhancing contaminant transport and altering subsurface permeability. A key objective is to be able . . .
P/CH—FG02-92ER81349	DURABLE, LOW-COST CERAMIC MATERIALS FOR USE IN HOT GAS FILTRATION EQUIPMENT	Ceramic candle filtration is an attractive technology for particulate removal at high temperature. Due to their simple and cost effective design, temperature capability, and high filtration efficiency, ceramic candles are one of the few clean up . . .

Table B-2. DOE Projects with Unknown Determination (continued)

Project ID	Project Title	Project Description
P/CH—FG02-93ER61534	MODELING OF COUPLED PROCESSES IN SUBSURFACE TRANSPORT OF REACTIVE CONTAMINANTS	This research program is developing and applying mathematical modeling tools to describe the complicated chemical, microbiological, and transport processes that occur simultaneously when heavy-metal radionuclides are codisposed of with strong organic . . .
P/CH—FG02-95ER82007	ELECTROKINETICALLY ENHANCED MICELLAR EXTRACTION OF RADIONUCLIDES/HEAVY METALS AND ORGANICS FROM SOIL	Remediation of soils with low hydraulic permeability, and contaminated with radionuclides/heavy metals and organics is a complicated task. The complexities arise partly because these contaminants tend to remain sorbed on the soil, and partly because . . .
P/CH—FG02-95ER82026	ADVANCED NUCLEAR AIR PRECLEANER	Disposal of nuclear-contaminated air filters has a major financial, environmental impact. Periodic replacement of these costly filters can be greatly extended if combined with novel nonbarrier submicron exclusion prefilters. These self-cleaning . . .
P/CH—FG02-95ER82037	USE OF COMPUTER ROBOTICS TO REDUCE HUMAN CONTACTS WITH THE WASTE STREAM AND LOWER THE COSTS FOR RECYCLABLE MATERIALS	Recent improvements in computer technology and robotics have made it possible to design a computer assisted robotic system to assist workers to recover recyclable materials from waste materials on a moving conveyor belt. The Phase I program measured . . .
P/CH—FG02-96ER86041	MOBILIZATION AND REMOVAL OF STRONTIUM AND CESIUM FROM SOIL BY CHEMICAL TREATMENT AND PHYTOREMEDIATION	Many DOE sites contain soil contaminate with radionuclides including 137-CS and 90-Sr posing significant health risks. Existing technology to alleviate this risk is costly and requires highly-specialized transportation techniques. The applicant proposed . . .
P/CH—FG02-96ER86043	EXTRACTION OF METALS AND RADIONUCLIDES FROM SOILS AND GROUNDWATER BY IN SITU MICROBIAL AND GEOCHEMICAL SOLUBILIZATION AND PRECIPITATION PROCESSES	—
P/EML—95-01	REMOTE ATMOSPHERIC MEASUREMENT PROGRAM	Provides near real time gamma-ray analyses of atmospheric aerosols collected at remote sites.
P/GO—FC36-88ID12726	ADVANCED MOISTURE SENSOR RESEARCH AND DEVELOPMENT WORKSHOPS	The project objective is to develop a hydrogen transient nuclear magnetic resonance sensor for measurement of the moisture content of materials being dried to minimize drying energy requirements and maximize process efficiency.

Table B-2. DOE Projects with Unknown Determination (continued)

Project ID	Project Title	Project Description
P/GO—FC36-93CH10561	CATALYTIC MEMBRANE PROGRAM	The purpose of this project is to investigate the use of ceramic membranes in dehydrogenation reactions. Many industrial dehydrogenation reactions are operated under conditions where conversion is limited by the presence of free-evolving hydrogen . . .
P/ID—AM07-95ID13384	DOE/NRL/RUSSIA JOINT ENVIRONMENTAL PROJECT TO CHARACTERIZE CHEMICAL AND NUCLEAR SOURCE TERMS AT SIBERIAN FACILITIES AND ANGARA AND YENISEY RIVERS	The objective of this work is to investigate and measure both the chemical and radionuclide source terms and potential pathways for discharges into the Kara Sea from land based sources in the Siberia watershed of the Ob and Yenisey Rivers. The . . .
P/ID—FC07-95ID13395	ENHANCED RESEARCH IN GROUND-PENETRATING RADAR AND MULTI-SENSOR FUSION WITH APPLICATION TO THE DETECTION AND VISUALIZATION OF BURIED WASTE	The objective of this project is to interface the work in theoretical and alogrithmic capabilities in multisensor fusion, inverse scattering, and computational modeling with DOE sensors.
P/ID—FG07-96ER14695	POLYOXOMETALATED FOR RADIOACTIVE WASTE TREATMENT	The proposed research is directed towards the use of inorganic polyoxometalate complexes in the early transition metals for (1) selective separation/concentration and (2) conversion to alternative waste forms, of radioactive components, present in . . .
P/ID—FG07-96ER14716	ELECTROCHEMICAL PROCESSES FOR IN SITU TREATMENT OF CONTAMINATED SOILS	This project is to study electrochemical processes for the in situ treatment of soils contaminated by mixed wastes, i.e., Organic and inorganic. Soil samples collected from selected DOE waste sites will be characterized for specific organic and metal . . .
P/ID—FG07-96ER14732	SURFACE NUCLEAR MAGNETIC RESONANCE IMAGING OF WATER CONTENT DISTRIBUTION IN THE SUBSURFACE	The objective of this proposal is to advance the technology of nuclear magnetic resonance for direct measurement of water content distributions in the subsurface. The proof-of-concept of this method has been demonstrated by Russian scientists . . .
P/ID—FG07-96ER14733	SPECIATION AND STRUCTURAL CHARACTERIZATION OF PLUTONIUM AND ACTINIDE-ORGANIC COMPLEXES IN SURFACE AND GROUNDWATERS	Study the chemical nature of the actinides and their association with specific organic ligands in the natural environment. Bring to this study a range of newly developed technologies which the PI's have used to study the physical organic and . . .

Table B-2. DOE Projects with Unknown Determination (continued)

Project ID	Project Title	Project Description
P/ID—FG07-96ER45617	CHEMICAL AND CERAMIC METHODS TOWARD THE SAFE STORAGE OF ACTINIDES USING MONAZITE	No abstract provided. . .
P/ID—FG07-96ER62310	THE SONOPHYSICS AND SONOCHEMISTRY OF LIQUID WASTE QUANTITFICATION ANDREMEDIATION	The legacy of waste left from the cold war has created an environment that requires unique and breakthrough technologies in order to solve some of DOE's waste problems. It has been demonstrated that acoustic cavitation-the birth, growth, and violent . . .
P/ID—FG07-96ER62315	COMPARISON OF THE BIOAVAILABILITY OF ELEMENTAL WASTE LADEN SOILS USING IN VIVO AND IN VITRO ANALYTICAL METHODOLOGY, AND REFINEMENT OF EXPOSURE/DOSE MODELS	The bioavailability of lead, uranium, chromium, cadmium, and cesium will be examined using soils obtained from DOE waste sites before and after treatment with a remediation technology. An in-vitro synthetic bio-fluid extraction model of the human . . .
P/ID—FG07-96ER62321	ADVANCED EXPERIMENTAL ANALYSIS OF CONTROLS ON MICROBIAL FE (III) OXIDE REDUCTION	Microbial reduction of Fe(III) oxides under anaerobic conditions has numerous important consequences for the transport of fate of heavy metals, radionuclides, organic contaminants, and organic/metal co-contaminants in subsurface environments. To . . .
P/INEL—1118	ADVANCED COMBINED ENVIRONMENTS TEST STATION (URC G035)	—
P/INEL—189E2084	EVALUATION OF EFFLUENT AND RADIOLOGICAL SURVEILLANCE ISSUES	Evaluation of effluent and rad surveillance . . .
P/INEL—2023	PETROLEUM AND ENVIRONMENTAL MODEL BENCHMARKING AND EVALUATION	—
P/INEL—2053	DEVELOPMENT OF METHODS FOR RAPID SITE SPECIFIC HAZARD ASSESSMENTS	
P/INEL—2111	REMOTE MONITORING OF IN SITU CONTAMINATION USING OPTICAL SPECTROSCOPY (URC COLLABORATION)	—
P/INEL—2228	INEL-UI LABORATORY FOR LIQUID EXTRACTION AND ION EXCHANGE RESEARCH, THIRD YEAR	—

Table B-2. DOE Projects with Unknown Determination (continued)

Project ID	Project Title	Project Description
P/INEL—2245	LASER ENHANCED ZERO ADDED WASTE CUTTING, ABRADING AND DRILLING	—
P/INEL—2247	ENVIRONMENTAL SEPARATIONS	—
P/INEL—4512	CHEMICAL SENSOR SYSTEMS FOR REALTIME, INTELLIGENT AIR QUALITY WASTE-SITE CHARACTERIZATION AND WIDE AREA MONITORING APPLICATIONS	—
P/INEL—95-275	DOE-SR - LANDFILL STABILIZATION	Management support to landfill focus area . . .
P/INEL—95-276	DOE-SR - LANDFILL STABILIZATION	System engineering support to landfill focus area . . .
P/INEL—96-179	MASON & HANGER-BIOPROCESSING APPARATUS	Bioprocessing of stored mixed wastes . . .
P/INEL—96-244	LANL - PDP DRUM PURCHASE	Design of benign matrix drums for the non-demonstructive assey performance demonstration program for the national tru program . . .
P/INEL—ADSID133503	INTEGRATION OF INCINERATION, WASTE DISPOSAL	Intregration of incineration.
P/INEL—ADSID34EG	WAG 4-CFA ASSMNT	Assess contamination at CFA for potential risk to human health and environment.
P/INEL—ADSID34EGA	WAG 4-CFA ASSMNT	Assess contamination at CFA for potential risk to human health and environment.
P/INEL—ADSID34EGB	WAG 4 - CFA ASSESSMENT	Cleanup contamination at the CFA that causes potential risk to human health and environment.
P/INEL—ADSID40EG	WAG 7-RAD WASTE MGMT ASSMNT	Assess contamination at RWMC for potential risk to human health and environment.
P/INEL—ADSID40EGA	WAG 7-RAD WASTE MGMT ASSMNT	Assess contamination at RWMC for potential risk to human health and environment.
P/INEL—ADSID40EGB	WAG 7-RAD WASTE MGMT CLEANUP	Cleanup contamination at the RWMC that causes potential risk to human health & environment.
P/INEL—ADSID413203	ROBOTICS LAB AUTOMATION	Robotics rollup.

Table B-2. DOE Projects with Unknown Determination (continued)

Project ID	Project Title	Project Description
P/INEL—ADSID4308EG	TSA RETREIVAL ENCLOSURE	The purpose of this project is to provide facilities and equipment to retrieve approximately 250,000 drum equivalents of existing mixed waste located at the radioactive waste management complex. Mixed waste does not comply with storage . . .
P/INEL—ADSID72C241	IN SITU SIMS ANALYSIS	Prepare sims technologies for transfer to private industry technology comparisons . . .
P/INEL—ADSID72C241-CE	IN SITU SIMS ANALYSIS	The in situ SIMS (secondary ion mass spectrometry) program resulted in the design and construction of an ion trap secondary ion mass spectrometer (it-shims), which is capable of the rapid analysis of environmental samples for adsorbed surface . . .
P/INEL—ADSID74MW77	RCRA DEFINITION	Evaluate commercial capabilities to perform non-intrusive RCRA elemental measurements on waste drums . . .
P/INEL—ADSID75C121	ROBOTICS INEEL ROLLUP	Contamination analysis automation: the objective of the robotics technology development program contamination analysis automation program FY-1996 is to develop beta version of the concentration and soxhlet organic modules and deploy them . . .
P/INEL—ADSID75C121-CE	ROBOTICS INEL ROLLUP	Contamination analysis automation: the objective of the robotics technology development program contamination analysis automation program FY-1996 is to develop beta version of the concentration and soxhlet organic modules and deploy them . . .
P/INEL—ADSID76C311	BENCH SCALE TESTING SEPARATION INEEL	Support on-going program with Russian scientists for separations of radionuclides from Idaho National Engineering and Environmental Laboratory will involve testing of solvent extraction processes in a remotely-operated centrifugal . . .
P/INEL—ADSID76C311-CE	BENCH SCALE TESTING SEPARATION INEEL	Support on-going program with Russian scientists for separations of radionuclides from Idaho National Engineering and Environmental Laboratory (INEEL) acidic waste streams. This will involve testing of solvent extraction processes in a remotely-operated centrifugal . . .
P/INEL—ADSID76LF11	LSFA PROGRAM	Management of landfill stabilization focus area tru-arid line . . .

Table B-2. DOE Projects with Unknown Determination (continued)

Project ID	Project Title	Project Description
P/INEL—ADSID76LF22-CE	BWID	Nondestructive examination/nondestructive assay . . .
P/INEL—ADSID76MW58	SUPERCRITICAL CARBON DIOXIDE EXTRACTION	Participate in joint evaluation of the Rocky Flats supercritical fluid extraction system and make the determination whether or not INEEL can construct a smaller-scale system comparable to the Rocky Flats system such that radionuclide partitioning . . .
P/INEL—ADSID76MW96	NDE/NDA OF CONTAINERIZED TRU WASTE	Develop a system utilizing artificial intelligence techniques and other tools to integrate waste assay data for improved results . . .
P/INEL—ADSIDHQ3341A	WASTE MIN SUPPORT	Install a cuber system and construct cover over receiving area. Cuber to cube general cold waste stream and burn cubes in CFSGF with coal . . .
P/INEL—ADSIDHQ35202	LLW-COMMERCIAL	Safe and efficient management of GTCC LLW generated by NRC.
P/INEL—FLU5AC311	NORM STUDY	Analysis of naturally occurring radioactive materials in oilfield equipment and wastes . . .
P/INEL—V016	APPLICATION OF SOLUTION THERMODYNAMICS AND AEROSOL DYNAMICS TO ENHANCE RESIDUE SEGREGATION IN MOLTEN METAL WASTE TREATMENT	—
P/INEL—V018	REMOTE MONITORING OF IN SITU CONTAMINATION USING OPTICAL SPECTROSCOPY	—
P/INEL—V022	CHEMICAL AND PETROGRAPHIC CHARACTERIZATION OF VITRIFICATION PRODUCTS	—
P/INEL—V101	MEMBRANE PROCEDURE FOR RECOVERY OF RAD. SPEC. FROM ACIDIC	—
P/LANL—400	SHUFFLER WASTE MATRIX CORRECTION	Develop a technique to improve waste and residue assays using Californium shufflers by correcting for the position of the uranium within a moderating matrix. Shufflers are used for materials control and accountability measurements on uranium in . . .

Table B-2. DOE Projects with Unknown Determination (continued)

Project ID	Project Title	Project Description
P/LANL—403	NDA TECHNOLOGY EXCHANGE AND IMPLEMENTATION	Provide a focal point for transfer of integrated safeguards technologies to DOE processing, dismantlement, and storage facilities. Support guidance, assistance, and ad-hoc training to assure that technology developed through the DOE safeguards R&D . . .
P/LANL—405	ADD-A-SOURCE WASTE DRUM ASSAY SYSTEM	Design, develop, and build a high-efficiency, passive neutron drum counter with added multiplicity counting capability and improved detection sensitivity. The counter will measure the plutonium content in waste drums and other bulk containers, and . . .
P/LANL—417	CTEN/TGS PACKAGE MONITOR	Analyze different assay instruments to determine their capabilities and weaknesses for detecting contraband fissile material in the packages or waste containers. Information is obtained experimentally from available instruments at Los Alamos TA-18, . . .
P/LANL—425	EXPERIMENTAL INVENTORY VERIFICATION SYSTEM	The Experimental Inventory Verification System based surveillance system developed through OSS R&D funding designed to help reduce physical inventory frequency for nuclear materials in process or storage at DOE . . .
P/LANL—93156	INSTRUMENTATION FOR HIGH EFFICIENCY - HIGH SENSITIVITY ACTINIDE ANALYSIS	As we strive to measure vanishingly small numbers of atoms of actinides, whether it is to determine environmental contamination, studies in geochronology or geochemistry, or weapons diagnostics, a limitation is imposed on the measurements by the . . .
P/LANL—93807	SEPARATIONS TECHNOLOGY DEVELOPMENT FOR ACCELERATOR TRANSMUTATION TECHNOLOGY CONCEPTS	This proposal describes separations technology development needed for Acccelerator-Driven Transmutation Technologies concepts, particularly those associated with plutonium disposition (Acccelerator Based Conversion) and high-level . . .
P/LANL—94450	INTEGRATED TREATMENT PROGRAM FOR METAL AND ORGANIC-CONTAINING LEGACY WASTES	Throughout its years of operation, the DOE complex has generated a range of legacy wastes which contain both hazardous organics and toxic and/or radioactive metals. In addition to metals such as lead, chromium (as chromate in paints) and actinides, . . .

Table B-2. DOE Projects with Unknown Determination (continued)

Project ID	Project Title	Project Description
P/LANL—95450	RADIONUCLIDE SEPARATIONS USING PILLARED LAYERED MATERIALS	The goal of this proposal is to develop porous Pillared Layered Materials to efficiently adsorb radionuclides, such as strontium (90-Sr) and cesium (137-Cs) from liquid nuclear wastes stored in underground storage tanks at DOE sites (Hanford, Oak . . .
P/LANL—95453	DECONTAMINATION OF RADIOACTIVE LIQUIDS BY FREEZE CONCENTRATION AND FRACTIONAL PRECIPITATION	Freeze concentration is a physical process whereby the partial freezing of the aqueous portion of a solution allows the segregation of the solute and solvent. This process has been sucessfully demonstrated on hazardous waste. In this process water . . .
P/LANL—95663	POLYMERS FOR NUCLEAR MATERIALS PROCESSING	The use of open-celled microcellular foams as solid sorbents for metal ions and other solutes could provide a revolutionary development in separation science. Macroreticular and gel bead materials are the current state-of-the-art for solid sorbents . . .
P/LANL—95665	STRUCTURAL AND MAGNETIC CHARACTERIZATION OF ACTINIDE MATERIALS	Neutron scattering techniques are used to study the complex structures and physical properties of actinide elements, compounds, and alloys. We propose to extend and enhance our capabilities in this study by developing new models for analysis of data . . .
P/LANL—A346	DETERMINATION OF PU AND METALS	Determination of Pu and Metals XX Project description not available at this time. Call John Meier at 505-667-6698 or e-mail at JVMEIER@LANL.GOV to request a project description. Project description not available at this time. Call John Meier at . . .
P/LANL—B308	FRACTURE MATRIX TRANSP COUPLING	The goal of this study for these analyses is to identify the most sensitive processes to the transport of Radionuclide, and to help guide geochemical site characterization including coupled phenomena. Specific objectives are construction of summarized . . .
P/LANL—B30A	SOLID TUFF COLUMN EXPERIMENTS	This report will summarize the technique utilized to validate radionuclide sorption data under unsaturated conditions using solid tuff column experiments and the strengths and weaknesses of this approach. Transport of colloids in fractured tuff under . . .

Table B-2. DOE Projects with Unknown Determination (continued)

Project ID	Project Title	Project Description
P/LANL—B311	RADIONUCLIDES SPECIATION MEASURE BY PAS	This interim report will present the temperature dependent hydrolysis and preliminary evaluations of carbonate constants for Pu using a combination of NMR and PAS techniques. Plutonium is an actinide with considerable uncertainty in its speciation . . .
P/LANL—B318	CONCEPTUAL MODEL OF MINERAL EVOLUTION	Determine the stability of minerals and glasses along flow paths to the accessible environment to assess impacts of waste emplacement on mineral stability and the resultant effects on radionuclide migration. Develop a conceptual model of mineral and . . .
P/LANL—B321	RETARDATION SENSIVITY ANALYSIS	The goal of this study for these analyses is to identify the most sensitive processes to transport of Radionuclide and to help guide geochemical site characterization including coupled phenomena. Specific objectives are construction of summarized . . .
P/LANL—B32A	CONDUCT FRACTURED TUFF COLUMN EXPERIMENT	This is a key supporting milestone for the Radionuclide Transport Model. This report will summarize the results of performing fractured tuff column experiments using the radionuclides (Np and Cs). The transport of the radionuclides in tuff containing . . .
P/LANL—B346	SOLUBILITY AND SPECIATION MODELING	This is a key supporting milestone for the Radionuclide Transport Model. This report will describe the modeling of Np solubility using the best available database. The reliability of the general results will be tested (and will in turn test) by . . .
P/LANL—B382	CONDUCT RETADATION SENSITIVITY ANALYSIS	The goal of this study for these analyses is to identify the most sensitive processes to transport of Radionuclide and to help guide geochemical site characterization including coupled phenomena. Specific objectives are construction of summarized . . .
P/LANL—D637	IWRP - BOEING SUPERCRITICAL CO2 CLEANING	—
P/LANL—D643	IWRP - SUPERCRITICAL C02 CLEANING MGMT	—

Table B-2. DOE Projects with Unknown Determination (continued)

Project ID	Project Title	Project Description
P/LANL—E334	ACTINIDE ORGANOMETALLIC CHEMISTRY	These investigations seek to extend our understanding of the chemical behavior and electronic structure of complexes of the actinide elements. Synthetic studies of the nonaqueous coordination and organometallic chemistry of the actinides play a key . . .
P/LANL—E347	ACTINIDES IN NEAR NEUTRAL SOLUTIONS	The project objective is to provide fundamental physicochemical knowledge pertinent to the behavior of transuranic (Np-Am) elements under environmental near-neutral pH conditions. Through examination of complexation, redox stability, and solubility . .
P/LANL—E372	ENVIRONMENTAL MANAGEMENT SCIENCE PROGRAM	The project objective is to systematically prepare and study high-valent actinide complexes formed under highly-alkaline conditions similar to that of aging radioactive waste tanks at Hanford, Savannah River, INEEL, West Valley, and Oak Ridge. Under . . .
P/LANL—E437	P-ELECTRON SPECTROSCOPY OF TRANSURANICS	The transuranic photoemission facility (utilizing laser-plasma -produced tunable UV light) is now nearly complete and ready for the introduction of radioactive materials - Pu, Np and their compounds. The light source mimics a first generation . . .
P/LANL—F138	CO-CONTAMINANT READT. CHEM./INTERFACES	This project consists of an experimental study of the fundamental processes underlying the sorption, partitioning and chemical alteration of specific radionuclides and radionuclide/ organic co-contaminant mixtures on reference and natural . . .
P/LANL—K14B	U233 B MINOR ACTINIDES DISPOSITION	Project description not available at this time. Call John Meier at 505-667-6698 or e-mail at JVMEIER@LANL.GOV to request a project description.
P/LANL—K72A	DRUM INVENTORY VERIFICATION ASSAY SYSTM	Project description not available at this time. Call John Meier at 505-667-6698 or e-mail at JVMEIER@LANL.GOV to request a project description.
P/LANL—K73A	NON-INVASIVE	Project description not available at this time. Call John Meier at 505-667-6698 or e-mail at JVMEIER@LANL.GOV to request a project description.

Table B-2. DOE Projects with Unknown Determination (continued)

Project ID	Project Title	Project Description
P/LANL—K73B	STABILIZATION STANDARDS DEVELOPMENT	La-ur-96-3985. Analytical chemistry technology development: emphasis is on improvement of analytical core capabilities and the ability for Los Alamos to address current and future analytical chemistry problems within the stockpile management program.
P/LANL—K73C	STABILIZATION PROCESS DEVELOPMENT	Project description not available at this time. Call John Meier at 505-667-6698 or e-mail at JVMEIER@LANL.GOV to request a project description.
P/LANL—K73G	PACKAGING TECHNOLOGY DEVELOPMENT	Project description not available at this time. Call John Meier at 505-667-6698 or e-mail at JVMEIER@LANL.GOV to request a project description.
P/LANL—KB21	PU WASTE & RESIDUE CAPABILITY	La-ur-96-4014. Plutonium recovery and processing: for the express purpose of continuing with the task of treatment and disposition of the legacy transuranic (tru) inventory at Los Alamos, this project includes support for process development tasks . . .
P/LANL—KB27	NUCLEAR MATERIALS PACKAGING	La-ur-96-4042. Nuclear materials packaging: included in this project to package transuranic materials for long-term storage is the development effort for storage containers meeting DOE-STD-3013, evaluation of long-term storage criteria, development . . .
P/LANL—KB42	NUCLEAR MATERIALS STORAGE	The Plutonium Packaging project in FY-1995 developed and demonstrated quality-assured packaging of plutonium to meet DOE-STD-3013-94. A double welded system of nested containers fabricated with tungsten-inert-gas welds was qualified for long-term . . .
P/LANL—M02A	PROCESS SCIENCE OF SLUDGES	Project description not available at this time. Call John Meier at 505-667-6698 or e-mail at JVMEIER@LANL.GOV to request a project description.
P/LANL—M36N	DEVELOPMENT AND IMPLEMENTATION OF ENVIRONMENTAL TECHNOLOGY	The initial objective of the program is to provide a methodology and the necessary fiscal and management controls to develop, demonstrate, and implement treatment technologies for mixed low-level radioactive and hazardous wastes that exist at DOE/AL . . .

Table B-2. DOE Projects with Unknown Determination (continued)

Project ID	Project Title	Project Description
P/LANL—M404	LANL PRE-TREATMENT	La-ur-96-3965. Los Alamos has participated in sludge-washing and alkaline-leaching screening tests on actual hanford tank sludges. In collaboration with pacific northwest laboratory, we have documented the behavior of various sludges; these studies . . .
P/LANL—M406	LANL PRE-TREATMENT	La-ur-96-3964. Approximately 1800 kg of technetium are present in the Hanford waste tanks. Technetium's potential to migrate in groundwater and its long half-life (t1/2 = 213,000 years) make it a major contributor to the long-term hazard associated . . .
P/LANL—M416	ROBOTICS LABORATORY AUTOMATION	La-ur-96-3967. Currently characterization of waste sites is extremely expensive, costing the Department of Energy billions of dollars each year. Inaddition to the enormous costs, there is a lack of necessary facilities and trained personnel. The . . .
P/LANL—M428	ELECTROLYTIC TREATMENT OF WASTE	La-ur-96-3968. Los Alamos and Faraday Technologies, Inc., Are developing an electrochemical process to treat hazardous and radioactive wastes that contain metal solutions, cyanide solutions, nitrate solutions, and various organic wastes that may . . .
P/LANL—M43A	TRANSPORTATION AUTOMATION	Project description not available at this time. Call John Meier at 505-667-6698 or e-mail at JVMEIER@LANL.GOV to request a project description.
P/LANL—M466	MAGNETIC SEPARTATION OF SOILS-16LF51	La-ur-96-3971. Large volumes of contaminated soil and fluid exist in and are being generated by the DOE Defense Complex. Chemical treatment or direct disposal of these wastes can be prohibitively expensive. One of the Los Alamos responses to this . . .
P/LANL—M46N	AIR QUALITY MONITORING FOR ALPHA CONTAMINATION	La-ur-96-3975. Detection of airborne alpha contamination is a necessary safeguard at DOE facilities where plutonium and other transuranic elements have been or are being processed. The best current detection technology is the alpha continuous air . . .

Table B-2. DOE Projects with Unknown Determination (continued)

Project ID	Project Title	Project Description
P/LANL—M46W	VACUUM DISTILATION SEPARATION OF PLUTONIUM	La-ur-96-3978. Pyrochemical processing of plutonium results in large amounts of plutonium salt residues. Traditional recovery methods of the plutonium require aqueous dissolution and subsequent recovery of the plutonium. These aqueous processing . . .
P/LANL—M499	SEQUESTERING AGENTS/REMOVAL OF TRANSURAN	La-ur-96-3979. There is an urgent need for alternative technologies for treatment of radioactive waste water to meet regulatory limits, decrease disposal costs, and minimize waste. In particular, this technology would address the need to replace . . .
P/LANL—M49A	TECHNOLOGY FOR IMPROVED TRU WASTE ASSAY	Project description not available at this time. Call John Meier at 505-667-6698 or e-mail at JVMEIER@LANL.GOV to request a project description.
P/LANL—M54A	WASTE TREATMENT PROJECT DEVELOPMENT CONTINUATION	Project description not available at this time. Call John Meier at 505-667-6698 or e-mail at JVMEIER@LANL.GOV to request a project description.
P/LANL—RA48	DETERMINATION OF RADIONUCLIDES IN ENVIRONMENTAL SAMPLES	To provide the measurement of extremely low levels of radioactive species in marine samples-sediments, and biological (reports).
P/LANL—RL38	REMEDIATION OF SAND REMOVAL OF URANIUM FROM PROJECTILE CATCH BOXES	The application of characterization and remediation technologies as a means of extracting the uranium from the sand in the catch box. (reports)
P/LANL—RU27	SPECIATION AND FATE OF RADIONUCLIDES IN THE ENVIRONMENT	Perform a systematic preparative, spectroscopic, and structural characterization of environmentally significant complexes of uranium and other nuclear waste radionuclides. (final published reports)
P/LANL—RU28	PLUTONIUM AND URANIUM METAL-FORMING TECHNOLOGIES	Develop and perform technology demonstrations of near-net-shape casting technologies for both plutonium and uranium in an effort to address waste minimization. (demonstrations and technical reports)
P/LANL—T688	OPTIMIZATION STUDY: PHOTON ACCELERATOR	Engineer and test a LANL accelerator. Evaluate device as a radioactive waste transmutation device for Japan Atomic Energy Research Institute. (report)
P/LANL—T933	OPTICAL RADIATION SENSOR	Fabricate and adapt radiation sensors based on optical fibers coupled with scintillation polymers. (prototype)

Table B-2. DOE Projects with Unknown Determination (continued)

Project ID	Project Title	Project Description
P/LANL—TA19	OPTIMIZATION STUDY: PHOTON ACCELERATOR	Engineer and test a LANL accelerator. Evaluate device as a radioactive waste transmutation device for Japan Atomic Energy Research Institute. (report)
P/LANL—XAW9	RADIONUCLIDE SEPARATIONS USING PILLARED	—
P/LBNL—366037	SELECT	Scientists at the Lawrence Berkeley National Laboratory have formed an interdisciplinary team to assess potential health and environmental risks posed by toxic agents at hazardous waste sites. This team has produced a prototype computer framework.
P/LBNL—366121	SELECT	Development of a Site Remediation Analysis and Decision Support Software . . .
P/LBNL—4318	SITE CHARACTERIZATION OF GROUNDWATER FLOW	Study pump-and-treat methodology in fractured rock systems . . .
P/LBNL—5218	FIELD EXPERIMENT AT THE TAN AND RWMC SITES (SF06SO31)	Explore and refine the use of Isotopic ratio measurements on aquifer groundwaters . . .
P/LBNL—5219	FLUID FLOW IN FRACTURED VADOSE ZONE (SF06SP31)	Subsurface flow and transport processes is critical for effective assessment decision making and remediation activities for contaminated sites.
P/LBNL—5222	BIOREMEDIATION RESEARCH DEVELOPEMENT PROGRAM	—
P/LBNL—5261	IN/US SYSTEMS, INC. - SMALL CRADA	Tri-Sorber Tritiation Manifold - This project develops a safe and efficient procedure for handling large quantities of tritium.
P/LBNL—6119	EV PRODUCTS, INC. - SMALL CRADA	Develop semiconductor materials, specifically CDTE and CDZNTE, for use in room-temperature-operation nuclear radiation detectors.
P/LBNL—781001	HAZARDOUS WASTE HANDLING FACILITY FABRICATION	Design and procure glove boxes for hazardous waste.
P/LBNL—782601	WASTE MGT/TECH SUPPORT: 35DB01040	Borehole imaging system. . .
P/LBNL—8090	GRAPHICAL IMAGING OF WASTE OR CONTAINMENT FLOW FOR . . .	This is a crada.

Table B-2. DOE Projects with Unknown Determination (continued)

Project ID	Project Title	Project Description
P/LBNL—8114	SOIL GAS TRANPORTATION	One specific aim of the research is to advance our fundamental understanding of the nature of soil-gas transport of vocs into residences as an exposure pathway for populations living near hazardous waste sites and landfills.
P/LBNL—8164	ENVIRONMENTAL RADIOACTIVITY TRANSPORT	Development of a monolithic quad germanium x-ray detector for synchrotron-based radionuclide characterization in geologic formations . . .
P/LBNL—8283	WATER CONTAMINANT CHARACTERIZATION	The purpose of this CRADA is to develop and explore the techniques of soft x-ray micro scopy for chemical state analysis using scanning . . .
P/LBNL—8345	INEL PERFORMANCE ASSESSMENT PEER REVIEW PANEL	Member of review panel . . .
P/LBNL—8357	DETERMINATION OF SOLUBILITIES AND COMPLEXATION	Measure solubilities of plutonium, americium, and neptunium in groundwater at Yucca Mountain . . .
P/LBNL—8415	FEDERAL FACILITY COMPLIANCE ACT IMPLEMENTATION	Federal facility compliance act implementation.
P/LBNL—860812	BIOLOGICAL EVALUATION OF NEW ACTINIDE CHELATING AGENTS—P. W. DURBIN	Develop new low-toxicity ligands for chelation of actinides.
P/LBNL—8970	MONITORING AND DATA ANALYSIS FOR THE VANDOSE ZONE MONITORING SYSTEM	Installation of Vandose Zone Monitoring System.
P/LBNL—BG-85-78	ULTIMATE FATE OF HAZARDOUS WASTE INJECTION STUDY	Assess parameters of confining layers of aquifers; select regional groundwater flow models applicable to deep underground injection and classify EPA listed hazardous wastes as they apply to injection zones.
P/LBNL—BG-91-278	SITE CHARACTERIZATION OF GROUNDWATER FLOW AND TRANSPORT IN FRACTURED ROCK SYSTEM	Study pump-and-treat methodology in heat fractured rock system.
P/LBNL—BG-95-15	ENVIRONMENTAL RADIOACITIVITY TRANSPORT	Development of a monolithic quad germanium x-ray detector for synchrotron- based radionuclide characterization in geologic formations.
P/LBNL—LB95004	STUDIES IN THE GEOLOGIC DISPOSAL OF NUCLEAR WASTE	—

Table B-2. DOE Projects with Unknown Determination (continued)

Project ID	Project Title	Project Description
P/LLNL—383	NDA MC&A MEASUREMENT TECHNOLOGY R&D	The objective of this project is to research and develop state-of-the-art NDA techniques and methods directed at major and significant MC&A problems of SNM associated with heterogeneous and shielded materials, SNM mixtures, lump corrections, holdup, . . .
P/LLNL—86-D-103	DECONTAMINATION AND WASTE TREATMENT	—
P/LLNL—92-D-403	TANK UPGRADES	—
P/LLNL—94-ERP-002	USING SHALLOW, HIGH-RESOLUTION SEISMIC REFLECTION TECHNOLOGY TO DELINEATE CONTAMINANT MIGRATION PATHS AND BARRIERS TO CONTAMINANT FLOW AND TRANSPORT	—
P/LLNL—94-ERP-032	MIXED-WASTE TREATMENT TECHNOLOGIES	
P/LLNL—95-ERD-045	ISOTOPE MEASUREMENTS FOR INNOVATIVE GROUNDWATER MANAGEMENT	—
P/LLNL—95-ERI-001	CHEMICAL SPECIATION OF ACTINIDES AND CHARACTERIZATION OF SOLID PHASES IN GROUNDWATER AND ENVIRONMENTAL SAMPLES: URANIUM AND NEPTUNIUM	—
P/LLNL—95-ERI-002	NOVEL COMPLEXING AGENTS FOR THE EFFICIENT SEPARATION OF ACTINIDES AND REMEDIATION OF ACTINIDE-CONTAMINATED SITES	—
P/LLNL—95-ERP-002	NONDESTRUCTIVE EVALUATION FOR PIT INSPECTION	—
P/LLNL—95-ERP-022	A NESTED-MODELING STUDY OF PRECIPITATION AND HYDROLOGY IN THE WESTERN UNITED STATES	—

Table B-2. DOE Projects with Unknown Determination (continued)

Project ID	Project Title	Project Description
P/LLNL—EHW—0067	BIODOSIMETRY TOOLS	This project is designed to substantially reduce uncertainties in biological dosimetry for low-level and moderate exposures to ionizing radiation. To accomplish this, we employ the most precise and best characterized biodosimeter available, . . .
P/LLNL—FEW—0010	NATURALLY OCCURRING RADIOACTIVE MATERIALS (NORM) ANALYSIS FOR OIL AND GAS OPERATIONS	This study is focused on determining the capabiities of existing methods and vendors to quantify accurately radium concentrations of regulatory concern, and validaing the measured data upon which regulatory decisions will be based. The study is . . .
P/LLNL—L-3901	WASTE INSPECTION TOMOGRAPHY	Develop a non-intrusive waste container inspection scanner based on well known and demonstrated techniques of emission and transmission computed tomography.
P/LLNL—L-3943	US RADIOLOGICAL TECHNOLOGIST SUPPLEMENTAL STUDY	Measure somatic cell mutation frequency in blood samples from individuals. Measure chromosomal translocation frequency in blood samples of some of these individuals.
P/LLNL—L-4433	FIN W6087-4: EARTHQUAKE INVESTIGATIONS	Investigate the impact that a seismic event had on selected facilities. Provide lessons learned that can have application to nuclear power plant industry.
P/LLNL—L-4873	TREATMENT OF AQUEOUS WASTE VIA CAPACITIVE DEIONIZATION (CDI) TECHNOLOGY	Construct and test a single dedicated CDI stack. Scoping studies will be performed to determine whether or not CDI technology can be used to effectively treat three specific types of waste generated at various bases and installations.
P/LLNL—L-5140	FIN W6087: EARTHQUAKE INVESTIGATIONS	Investigate impact that a seismic event had on selected facilities and provide lessons learned that can have application to nuclear power plant industry.
P/LLNL—L-6076	RADIOGRAPHY AND COMPUTED TOMOGRAPHY OF SOILS FOR HYDOLOGICAL STUDIES	Work with UC Davis to optimize radiography, and CT Techniques and Systems for the evaluation of soils for their hydrological properties.
P/LLNL—NNW—0001	SUPPORT TO DOE FACILITIES IN IMPLEMENTATION,TEST AND EVALUATION OF LLNL-DEVELOPED NDA TECHNIQUES	The objective of this project is to assist DOE sites in implementation of LLNL developed NDA technology; in particular, assist WSRC's H-Canyon facility; Livermore's Materials Management; LANL's TA-55 facility; and Hanford storage facilities.

Table B-2. DOE Projects with Unknown Determination (continued)

Project ID	Project Title	Project Description
P/LMES—208	ATTRIBUTES MEASUREMENT SYSTEMS FOR MONITORING SPECIAL NUCEAR MATERIAL IN STORAGE	Low cost, highly reliable inventory monitoring systems are needed to continuously monitor item and material attributes. This project will evaluate the capability of radiation monitors that use optical materials and solid-state sensors to provide . . .
P/LMES—456	VEHICLE PERSONNEL SCREENING	The Department of Energy maintains numerous facilities which previously supported nuclear weapons production. Despite the fact that many of these facilities will likely be decommissioned, a significant number of facilities will remain to support . . .
P/LMES—466	ATTRIBUTES MEASUREMENT SYSTEMS FOR MONITORING SNM	Low cost, highly reliable inventory monitoring systems are needed to continuously monitor item and material attributes. This project will evaluate the capability of radiation monitors that use optical materials and solid-state sensors to provide . . .
P/LMES—NN007	FULL SCALE DEVELOPMENT-MATERIAL CONTROL AND ACCOUNTING	Low cost, highly reliable inventory monitoring systems are needed to continuously monitor item and material attributes. This project will evaluate the capability of radiation monitors that use optical materials and solid-state sensors to provide . . .
P/METC—AC21-92MC29103	DEVELOPMENT OF A LONG-TERM, POST-CLOSURE RADIATION MONITOR	Babcock and Wilcox Company will develop a low-cost multipoint radiation monitoring system for the long-term continuous monitoring of radiation levels in the vadose zone of hazardous waste sites. The system will be based on gamma spectroscopy and will . . .
P/METC—AC21-92MC29104	MOBILE WORK SYSTEM FOR DECONTAMINATION AND DECOMMISSIONING	The project objective is to develop a high-performance remotely operated mobile worksystem capable of performing a wide range of decontamination and decommissioning tasks in nuclear environments.
P/METC—AC21-92MC29109	ROAD-TRANSPORTABLE ANALYTICAL LABORATORY SYSTEM	The project objective is to develop and design a reliable Road-Transportable Analytical Laboratory system capable of performing the full range of radiological, chemical, and biological analyses onsite in contaminated areas at DOE facilities.

Table B-2. DOE Projects with Unknown Determination (continued)

Project ID	Project Title	Project Description
P/METC—AC21-92MC29120	INNOVATIVE FOSSIL-FUEL-FIRED VITRIFICATION TECHNOLOGY FOR SOIL REMEDIATION	The project objective is to develop an innovative fossil-fuel-fired vitrification technology for the remediation of soils containing hazardous and/or radioactive constituents.
P/METC—AC21-92MC29121	REMOTE MINING FOR IN SITU WASTE CONTAINMENT	The project objective is to develop and demonstrate a system that is capable of in situ containment of underground hazardous waste sites. The proposed concept will adapt equipment and techniques from the mining and landfill industries to remotely . . .
P/METC—AC21-93MC30162	ELECTROKINETIC DECONTAMINATION OF CONCRETE	The project objective is to demonstrate the use of electrokinetic extraction technology to remove contaminants from concrete.
P/METC—AC21-93MC30164	CONCRETE DECONTAMINATION BY ELECTROHYDRAULIC SCABBLING	The objective of this effort is to develop an Electrohydraulic Scabbling System that is cost efficient, rapid, controllable, portable, operable under remote control, and that generates low volumes of secondary waste. Laboratory testing in . . .
P/METC—AC21-93MC30165	REMOTE OPERATED VEHICLE DRY ICE PELLET DECONTAMINATION SYSTEM	The project objective is to develop and deploy a remote-operated vehicle dry ice pellet blasting system for the removal of radioactive and hazardous organic contaminants from concrete surfaces at DOE's nuclear waste sites.
P/METC—AC21-93MC30170	ADVANCED TECHNOLOGIES FOR DECONTAMINATION AND CONVERSION OF SCRAP METALS	The project objective is to develop and demonstrate cost-effective and environmentally sound recycling of radioactively contaminated scrap metals to high-value intermediate and final product forms. This will be accomplished by demonstrating . . .
P/METC—AC21-93MC30171	RECYCLE OF CONTAMINATED SCRAP METAL	The project objective is to establish the commercial capability of Catalytic Extraction Processing for contaminated scrap metals.
P/METC—AC21-93MC30177	TREATABILITY STUDY USING PROMPT GAMMA NEUTRON ACTIVATION ANALYSIS TECHNOLOGY	Westinghouse Electric Corporation will develop, fabricate, and demonstrate an in situ analysis method for determining the level of radioactive and hazardous contaminants in concrete floors at selected DOE sites. The new method will decrease the need . . .

Table B-2. DOE Projects with Unknown Determination (continued)

Project ID	Project Title	Project Description
P/METC—AR21-94MC30359	LASER ABLATION OF CONTAMINANTS FROM CONCRETE AND METAL SURFACES	The goals of this program are (1) to design, build, and test a system for the removal of contaminated paint and other contaminants, such as grease, oil, and polychlorinated biphenyls from concrete and metal surfaces and (2) to capture and . . .
P/METC—AR21-94MC31191	AUTOMATED BASELINE CHANGE DETECTION USING A SAW/GC SYSTEM	Develop a reliable and accurate automated change-detection system which applies recent advances on optical sensor positioning technology to automatically detect and isolate mixed waste barrel changes which may indicate potential containment failures.
P/METC—AR21-95MC32108	SURFACE ALTERED ZEOLYTES AS A PERMEABLE BARRIER	Development of a pilot demonstration, and field installation of a permeable barrier of altered zeolite which is selective for the major classes of groundwater contaminants: soluble organics such as benzene and trichloroethylene, inorganic cations . . .
P/METC—AR21-95MC32110	MEASUREMENT OF RADIONUCLIDES USING ION CHROMATOGRAPHY AND FLOW-CELL SCINTILLATION COUNTING	The objective of this effort is to develop a technology that can measure transuranics and pure beta emitters relatively quickly and has field deployable potential.
P/METC—FC21-92MC28245	SOIL TREATMENT TO REMOVE URANIUM AND RELATED MIXED RADIOACTIVE HEAVY-METAL CONTAMINANTS	The project objective is to design and develop a physicochemical treatment process for the removal of uranium and heavy metals (radium-226, thorium-230, lead, manganese, etc.) from contaminated soil to achieve a target contamination level below . . .
P/NV—FC08-89NV10805	RESEARCH & DEVELOPMENT IN THE FIELD OF RADIATION DOSIMETRY USING ENVIRONMENTAL MATERIALS	The project objectives are (1) to automate the sample processing which will provide better response to requests for dosimetry estimates at lower costs, (2) to develop an optically-stimulated luminescence technique which uses lasers instead of . . .
P/OAK—AT03-79ER10414	CHEMISTRY OF GASEOUS LOWER VALENT ACTINIDE HALIDES	The objective of this project is to provide accurate thermochemical information for key actinide halide, oxyhalide, and related systems, starting with uranium halides, so that the basic factors underlying the chemical bonding and chemical reactivity . . .

Table B-2. DOE Projects with Unknown Determination (continued)

Project ID	Project Title	Project Description
P/OAK—FG03-88ER13851	ISOTOPE TRACER STUDIES OF DIFFUSION AND GEOLOGICAL TRANSPORT PROCESSES USING ACTINIDES	Mass spectrometric techniques will be used to investigate geochemical transport processes with a focus on the ^{238}U, ^{234}U ^{230}Th ^{232}Th system and on water, anion, and cation diffusion in silicates and oxides. Recent studies . . .
P/OAK—FG03-94ER81714	NEW SEMICONDUCTOR RADIATION SENSOR FOR EXPIDITED WASTE SITE CHARACTERIZATION	The overall goal is to develop low-cost “smart” sensors for hard x-rays suitable for use with minimally intrusive emplacement systems such as cone penetrometers and portable instruments. The sensors will efficiently measure energy spectra up to 300 . . .
P/OAK—FG03-94ER81727	COAL ASH TILES BY MICROWAVE PROCESSING	This project considers the use of microwave heating for the production of tiles and bricks from coal-derived ash. The generation of undesirable ash from coal-fired power plants has become a serious environmental problem. Microwave processing allows . . .
P/OAK—FG03-95ER82018	ION-SELECTIVE CERAMIC MEMBRANES FOR SEPARATION OF RADIOACTIVE WASTES	The principal objective of the project is to demonstrate the use of electrochemical cells for the separation of radioactive waste and salt splitting using highly ion-selective ceramic membranes. A plan of work is proposed to fabricate sodium ion . . .
P/OAK—FG03-96ER82227	INNOVATIVE METHOD TO STABILIZE LIQUID MEMBRANES FOR REMOVAL OF RADIONUCLIDES FROM GROUNDWATER	The DOE is seeking advanced technologies to remediate sites at which groundwater is contaminated with uranium and other radionuclides. One technology with significant potential uses liquid membranes, which can selectively extract uranium and enrich . . .
P/OR—FG05-87ER40329	RESEARCH IN ACTINIDE CHEMISTRY	This research emphasizes the basic studies of the behavior in solution of the actinide elements and of the chemically related lanthanide elements. The systems are chosen for investigation because the data can provide increased understanding of the . . .
P/OR—FG05-88ER13865	PHYSICAL/CHEMICAL STUDIES OF TRANSURANIUM ELEMENTS	This project provides training for pre- and postdoctoral students in chemical research with the transuranium 5f (actinide) and related 4f (lanthanide) elements. The goals of this project are to interpret and correlate the results of continuing . . .

Table B-2. DOE Projects with Unknown Determination (continued)

Project ID	Project Title	Project Description
P/ORNL—1824H099A1	RADIONUCLIDE DOSE AND RISK FACTOR CALCULATION (DCAL), PART II	The objective of this project is to update the Environmental Protection Agency's capabilities for calculating individual and collective health risks from both internal and external radionuclide exposures. The system is comprised of an . . .
P/ORNL—1824I009A1	HEALTH RISK FROM ENVIRONMENTAL EXPOSURE TO RADIONUCLUDES	Prepare a report for the Environmental Protection Agency on the health risks associated with environmental concentrations of selected radionuclides. This report will be the fourth in a series intended to provide the Federal agencies with technical . . .
P/ORNL—1824I035A1	ECOLOGICAL RISK ASSESSMENT METHODOLOGY DEVELOPMENT	This project includes four tasks related to the development and application of advanced ecological risk assessment techniques at DOE facilities: (1) Maintenance and distrubution of an ecological effects database previously developed using DOE funds, . . .
P/ORNL—1866H030A1	RADIONUCLIDE CONTAMINATION OF THE ARCTIC BASIN ECOSYSTEM	To translate and make available data and sample sets relating to contamination of the Arctic by activities of the former Soviet Union, as well as to fill information gaps by coordinating new sample collection efforts in the Kolyma River outflow . . .
P/ORNL—1866I023A1	RADIONUCLIDE CONTAMINATION OF THE ARCTIC BASIN ECOSYSTEM: CONTINUOUS	The objective of this research will be to establish continuous seawater gamma detection capabilities at three locations in Alaska, collaborate in radionuclide sampling throughout the Arctic, and participate in experimental studies of radionuclide . . .
P/ORNL—619010601	MMSC-APNEA	To assist LMSC in the development of a commercially-available Active-Passive Neutron Examination and Assay (APNEA) radioactive waste assay system.
P/ORNL—619012001	LMSC-RADIOLOGICAL DOSE ASSESSMENT	The Department of Energy Pinellas Plant proposes to ship neutralization tank F006 sludge to USPCI in Utah for disposal. For this waste to be approved for shipment and disposal a radiological performance assessment is required to ensure the protection . . .

Table B-2. DOE Projects with Unknown Determination (continued)

Project ID	Project Title	Project Description
P/ORNL—647048001	LBL-TESTBED FOR A REMOTE EXPERIMENTAL ENVIRONMENT	This task will provide a Remote Experimental Environment test bed. This test bed is to evaluate methods by which remotely located scientists can participate in the operation of a larger scientific experiment.
P/ORNL—EHHA050	NEUTRON DOSIMETRY USING BUBBLE TECHNOLOGY	The purpose is to combine bubble detector technology with more traditional TLD technology to solve problems in neutron dosimetry. Specfically, the proposal is for the development and testing of a Combined Area Neutron Spectrometer and a . . .
P/ORNL—EHHA056	FISSION NEUTRONS AND GAMMA RAYS	The specific aims of the whole-animal experiments are to determine the initial slopes of the dose-response curves, and therefore, risk estimates of low doses, can be obtained from data for low dose-rate irradiation and multiple fractions of low . . .
P/ORNL—EHHA105	DEVELOPMENT OF AEGLS	Development of Acute Exposure Guidance Levels (AEGLs) for hazardous substances is of direct interest to the Department of Energy Operations. This is a part of an FACA commitment to establish AEGLs. AEGLs are biological reference values that estimate . . .
P/ORNL—ERD8800736	JAERI - JAPAN/US ACTINIDES PROGRAM	Japan Atomic Energy Research Institute (JAERI), Tokai is currently studying the transmutation of the higher actinides. If this study shows promise, JAERI will extend the transmutation study further to include the designing of a higher actinide . . .
P/ORNL—ERD9601394	ROBOCON PROJECT SUPPORT	Provide R&D support in the area of human factors/ergonomics for applied telerobotic systems for a variety of inspection, maintenance, repair, and decontamination and dismantlement task.. . .
P/ORNL—ERKCC09	CHEMISTRY OF TRANSURANIUM ELEMENTS AND COMPOUNDS	Understanding of the chemical and solid-state physical behavior of the heavy actinides and their compounds in terms of 5f electron systematics and relativistic effects. Tasks focus on: (1) high temperature/pressure effects on bonding and peciation; . . .

Table B-2. DOE Projects with Unknown Determination (continued)

Project ID	Project Title	Project Description
P/ORNL—ERKCT03	TRANSURANIUM ELEMENT PROCESSING	Transuranium elements 96-100 (curium through fermium) are recovered from irradiated targets in the Radiochemical Engineering Development Center, which is the distribution center for the DOE heavy-element research program. Target rods are . . .
P/ORNL—ERKCT05	CHEMISTRY OF ACTINIDES AND FISSION PRODUCTS	This project is one of only a few remaining fundamental research efforts that are concerned with the physical–chemical characteristics of the actinides and fission products as related to separations schemes. Although the efforts are generally . . .
P/ORNL—ERKP031	MECHANISMS IN RADIATION CARCIN	X-radiation, in contrast to the complete carcinogen DMBA, does not enhance the in vitro growth capacity of, nor induce tumors in, exposed tracheal epithelium. Using a combined in vivo–in vitro model, collectively referred to as the epithelial focus . . .
P/ORNL—NEAF240	ACTINIDE PARTITIONING AND TRANSMUTATION PROGRAM	This task is to support the office of advanced reactor programs in the assessment of an advanced fuel cycle concept. The concept involves the use of liquid metal reactor systems to burn the long-lived actinide component of civilian nuclear . . .
P/PNNL—02683A	DECOMMISSIONING COST ESTIMATES	Prepare annual edition of client's NUREG-1307 and provide technical support related to decommissioning costs for inclusion in Generic Environmental Impact Statements . . .
P/PNNL—10122A	CHEMISTRY AND PHYSICS OF CERAMIC SURFACES	Determine chemical mechanisms of glass and crystalline ceramic interactions with reactive environments . . .
P/PNNL—10265	RADIONUCLIDE SOURCE TERM MEASUREMENTS FOR DECOMMISSIONING AS	Measure the concentrations and distributions of neutron activation products in the shipping port nuclear power plant at the time of and during decommissioning . . .
P/PNNL—15420	WASTE DISPOSAL CONTRACTS ANALYSIS	Assist DOE/RW/OPARM with the preparation of the Annual Capacity Report, provide support in Contract Issue Identification and Resolution, and assess impacts to Contract Implementation from evolving technical baselines . . .

Table B-2. DOE Projects with Unknown Determination (continued)

Project ID	Project Title	Project Description
P/PNNL—16269A	ENVIRONMENTAL RISK AND STANDARDS	Define the baseline activities that must be conducted prior to DOE's decision regarding development, application, and implementation of standards and criteria to the Environmental Restoration, Waste Mgt., and decommissioning activities . . .
P/PNNL—17629C	WASTE MANAGEMENT	Waste management and environmental compliance associated with PNL's multiprogram R&D work . . .
P/PNNL—17651A	ROBOTICS TANK WASTE RETRIEVAL	TWO Robotics Technology Development. Develop and evaluate advanced robotics technology options for manipulator-based retrieval of underground storage tank waste at sites across the DOE complex.
P/PNNL—17651C	ROBOTICS TANK WASTE RETRIEVAL	TWO Robotics Technology Development. Develop and evaluate advanced robotics technology options for manipulator-based retrieval of underground storage tank waste at sites across the DOE complex.
P/PNNL—18381B	MIXED WASTE PROGRAM	Support to HQ for Mixed Low-Level Waste Program. Will include technical and systems analysis, and data support. Support both strategic planning for the Mixed Low-Level Waste Program and activities for the FFCA task force.
P/PNNL—18389A	8581 IN SITU REMEDIATION INTEGRATED PROGRAM	To coordinate and manage the in situ remediation integrated program.
P/PNNL—18394B	8563 COMPREHENSIVE INTEGRATED PLANNING	To meet EM's needs for development of new and more efficient technologies for accomplishing environmental restoration in waste management and waste operations goals.
P/PNNL—18622A	ORGANIC COMPLEXATION AND MICROPARTICULATES	Evaluate the role of organic complexation and microparticulates in enhancement of radionuclide migration in groundwater . . .
P/PNNL—18678	8529 DEFINE/PREPARE TANK WASTE	Define and prepare non-radioactive tank waste simulants and recipes for use by other USTID tasks . . .
P/PNNL—18732	MCC SUPPORT TO SRL	Produce samples for and conduct analytical round robins. Acquire or fabricate analytical reference glasses, characterize, package, and distribute and act as a custodian.

Table B-2. DOE Projects with Unknown Determination (continued)

Project ID	Project Title	Project Description
P/PNNL—19458	REMOTE CHARACTERIZATION	Provide technical support services for ongoing efforts funded by DOE-HQ to develop a remote characterization system . . .
P/PNNL—19928	CHARACTERIZATION	Develop methods for characterization of waste samples and establish DQOs and systems to manage characterization data.
P/PNNL—19960B	REDOX MANIPULATION FIELD TEST	Enhancement of contaminant destruction and mobilization through in situ redox manipulation of the subsurface.
P/PNNL—19983A	ACT-DECON PROCESS	Verify Bradtec's ACT-DECON process to treat and recover actinides and nitrates from radioactive simulant solution.
P/PNNL—19992B	AIRBORNE RADIONUCLIDE ANALYZER/VAPOR COLLECTOR	Construct a real-time airborne radionuclide analyzer and vapor collector.
P/PNNL—20105	TECHNICAL SUPPORT TO D&D/HQ	Strategic planning and associated risk assessments to establish policy and priority for acceptance and disposition of surplus facilities.
P/PNNL—20150	MULTI-FUNCTION SCARIFIER/CONVEYER	Design and development of a multi-funtion scarifier-end effector with an integral conveyance system.
P/PNNL—20154	SIX-PHASE SOIL HEATING FOR ENHANCED REMOVAL OF CONTAMINANTS	Develop and demonstrate six-phase soil heating to enhanced removal of contaminants from soil.
P/PNNL—20160A	PNL LIGHT DUTY UTILITY ARM	Provide mechanical systems integration, mapping sensors, and sampling end effector for the Light Duty Utility Arm System.
P/PNNL—20163A	CHEMICALLY ENHANCED BARRIERS/MINIMIZE CONTAMINANT MIGRATION	Optimize effectiveness of hydraulic and diffusion barriers and permeable barriers to be used in vadose or aquifer sediments to minimize contaminant migration.
P/PNNL—20167A	CHARACTERIZATION & MONITORING TECHNIQUES	To assist in the design, implementation, and evaluation of technology demonstrations.
P/PNNL—20227A	APPLICATION OF INFILTRATION EVALUATION METHODOLOGY TO LLW PA	Expand the range of applicability of the Battelle-developed Infiltration Evaluation Methodology (IEM), document IEM changes, and provide computer codes used and guidance to client's staff for review of Low-Level Waste sites . . .
P/PNNL—21194	CONDUCT ANALYSIS OF MARINE SEDIMENT FOR CHEMISTRY AND	Conduct analysis of marine sediment for chemistry and biological effects for client's evaluation of disposal methods . . .

Table B-2. DOE Projects with Unknown Determination (continued)

Project ID	Project Title	Project Description
P/PNNL—21289	WASTE MINIMIZATION	Develop a database tracking system for waste generation that is consistent with the database developed for EM; also develop a stand-alone waste minimization training plan tailored to each generating facility or organization.
P/PNNL—21316	ELECTROMAGNETIC MOISTURE MEASUREMENTS IN SINGLE SHELL TANK	Evaluate and develop electromagnetic method to measure moisture in single shell-tank waste.
P/PNNL—21344A	SEPARATION OF TRITIATED WATER WITH MEMBRANES	Separation of Tritiated Water with Membranes Development of polyphosthazene membranes for separation of tritiated water from groundwater.
P/PNNL—21352	EVALUATE SUITABILITY OF DUAL SHELL REACTOR VESSEL FOR SUPER-CRITICAL WATER OXIDATION	
P/PNNL—21354	WD&C INTEGRATED TESTING, SUBTASK 6	Provide lead test engineering support for the waste dislodging and conveyance (WD&C) testbed.
P/PNNL—21356	PROVIDE SUPPORT TO QUEST INTEGRATED, INC.	Provide support to Quest Integrated, Inc. In development of Scarifier ultra-high-pressure water jet for waste retrieval.
P/PNNL—21360	DECONTAMINATION & DISMANTLING ROBOTICS (PNL)	Demonstrate truss-based manipulator for long reach applications.
P/PNNL—21401	ISV SPOT MELTING	Develop and implement an ISV spot melting concept in support of Hanford ER needs. Scope includes a series of investigative engineering-scale tests.
P/PNNL—21764A	PNC ACTINIDE SOLUBILITY STUDIES	Conduct thermodynamic data acquisition for carbonate complexes of actinide compounds and develop kinetic data for the crystallization and solubility of tetravalent actinides and hydrous oxides . . .
P/PNNL—21764B	PNC ACTINIDE SOLUBILITY STUDIES	Conduct thermodynamic data acquisition for carbonate complexes of actinide compounds and develop kinetic data for the crystallization and solubility of tetravalent actinides and hydrous oxides . . .
P/PNNL—22200A	M774 DEMILITARIZATION	Characterize depleted uranium particles from cartridge disassembly to identify appropriate detection instrumentation and contamination control procedures for demilitarization of cartridges . . .

Table B-2. DOE Projects with Unknown Determination (continued)

Project ID	Project Title	Project Description
P/PNNL—22280A	POLLUTION PREVENTION PROGRAM	Develop and maintain a series of pollution prevention tasks that will result in reducing or eliminating PNL waste streams.
P/PNNL—22309	PNL SUPPORT OF JOINT ER/EM-352 POLLUTION PREVENTION PROGRAM	Development Activities. Support jointly - ER and WM/EM-352 activities associated with the development of the DOE pollution prevention program.
P/PNNL—22384A	CRYOCELL FROZEN SOIL SUBSURFACE BARRIER	Manage the field demonstration and testing of the CRYOCELL forzen soil subsurface barrier at Hanford . . .
P/PNNL—22403	PROCESS EVALUATION AND COMPARISON TO BASELINE	Conduct systems studies from the perspective of the performance of ESP-IP (Efficient Separations and Processing Integrated Program) technologies compared to baseline technology at DOE/EM studies . . .
P/PNNL—22421A	STABILIZATION OF REACTOR FUEL STORAGE POOL	Evaluate standard and improved technology and design an integrated demonstration system to clean the water and sludg e in the N-Reactor 105-KE fuel storage pool.
P/PNNL—22459	VITREOUS WASTE FORM PERFORMANCE	Place DOE in defensible position with respect to emplacement of low-level waste glass in permanent storage at Hanford, through series of tests and modeling calculations. . .
P/PNNL—22538	GLASS TESTING TO SUPPORT PRIVATIZATION OF LLW VITRIFICATION	Perform laboratory testing program to evaluate single-pass flow-through test method used at PNL for use as a LLW glass product acceptance test.
P/PNNL—23093	ASSESSMENT OF ARCTIC CONTAMINATION	Assist client in preparing for bilateral discussions with Norway and the U.S. concerning radioactive contamination in the Arctic from sunken ships/submarines, subs that will be decommissioned, operational ships/subs, and other sources . . .
P/PNNL—23594A	GENERIC PIPE & DUCT FLOW SUBROUTINES FOR RADTRAD	Upgrade existing computer codes that calculate radioisotope transport through ducts and pipes and assist client with incorporation into their RADTRAD computer program . . .
P/PNNL—24319A	MURMANSK LOW-LEVEL WASTE PLANT EVALUATION	Partcipate as a team member on a trip to Murmansk, Russia to evaluate a low-level waste treatment plant . . .
P/PNNL—24712A	ALTERNATIVE ALKALINE WASHING OF HANFORD SLUDGE	Develop alternatives and enhancement to the baseline Hanford Sludge pretreatment flowsheet.

Table B-2. DOE Projects with Unknown Determination (continued)

Project ID	Project Title	Project Description
P/PNNL—24855	CRADA FOR PRELIMINARY DEVELOPMENT OF THE HANFORD HAND	Perform assessment of technologies that will be used to develop the Hanford Hand. The Hanford Hand utilizes the PNL patented . . .
P/PNNL—24869A	TUNABLE HYBRID PLASMA (THP)	Key field demonstrations and commercialization studies for the Tunable Hybrid Plasma system will be conducted by Massachusetts Institute of Technology.
P/PNNL—24887A	LIGHT AIDED DECONTAMINATION AND CUTTING	Demonstration of laser decontamination and cutting technology.
P/PNNL—24888A	ROBOTICS PNL ROLLUP TTP #RL 35C111	To develop, test and evaluate innovative and crosscutting technologies. This includes the design and implementation of automated chemical analysis systems, technology to enhance the productivity of a plasma hearth furnace, etc. . .
P/PNNL—24896A	HIGH-LEVEL VAULT INTERIM REMOVAL ACTION	Stabilization of the liquids and subsequent waste removal.
P/PNNL—24932A	ELECTRICALLY CONTROLLED CESIUM ION EXCHANGE	Prepare, test, and evaluate electroactive films for cesium ion exchange and design bench-scale ion exchange unit.
P/PNNL—25468A	SHIP AND TEST UWB UNDERGROUND IMAGING SYSTEM AT INEEL	Prepare Battelle-developed Ultra Wide Band (UWB) underground imaging system for shipping and testing at Idaho Nuclear Engineering and Environmental Laboratory (INEEL) site and provide technical services to client's personnel as required . . .
P/PNNL—25767A	CUFS FILTRATION	Conduct solid liquid seperation of various Hanford Tank wastes.
P/PNNL—25861A	DYNAMICS OF CONTAMINANT DEGRADATION IN HETEROGENEOUS MEDIA:	Conduct study of coupled reactive contaminant transport and dynamic bacterial attachment/detachment under growth and growth-limiting conditions in the subsurface . .
P/PNNL—25865A	TC REMOVAL FROM HANFORD TANK WASTE	Experiments will be conducted with actual Hanford tank wastes to investigate the removal of Tc using various processes. Initially, batch contacts will be performed with ion exchangers and sorbents. Waste quantities are 5 mL per . .

Table B-2. DOE Projects with Unknown Determination (continued)

Project ID	Project Title	Project Description
P/PNNL—25916A	EM-50 WASTE FORM SUPPORT	Provide an EM-50 integrated comparative analysis of current waste form options for mixed low-level waste; make programmatic recommendations to enhance implementation of the appropriate treatment and disposal systems . . .
P/PNNL—26108A	HQ - HANFORD RISK INITIATIVE	Activity facilitate the incorporation of risk considerations into DOE site planning and science and technology investment strategies.
P/PNNL—26160A	PERFORMANCE ASSESSMENT PANEL	Participate on DOE Peer Review Panel for Performance Assessments.
P/PNNL—26514A	PU DISSOLUTION KINETICS	Conduct glass melting studies to determine plutonium oxide dissolution kinetics. This work will complete project originated at Savannah River.
P/PNNL—PN93131/849	INTEGRATED ENVIRONMENTAL MONITORING	—
P/PNNL—PN94023/913	DEVELOPMENT OF LASER-DIODE BASED SENSORS FOR TRACE ISOTOPE ASSAYS	—
P/PNNL—PN94031/921	EVALUATION AND SELECTION OF IN-WELL SEPARATIONS PROCESS FOR METALS AND RADIONUCLIDES	—
P/PNNL—PN94041/931	DEVELOPMENT OF CARBONATE BARRIERS FOR IN SITU CONTAINMENT	
P/PNNL—PN94045/936	KINETICS, SCALE-UP, AND DEMONSTRATION OF URANIUM BIOPRECIPITATION TECHNOLOGY	—
P/PNNL—PN94047/937	MEMBRANE MATERIALS	
P/PNNL—PN94048/938	MEMBRANE SEPARATIONS	—
P/PNNL—PN94067/957	ENHANCED MIXING FOR SUPERCRITICAL FLUID OXIDATION (PHASE SEPARATIONS)	—
P/PNNL—PN95036/1012	GLASS STRUCTURE, CHEMISTRY, AND STABILITY	
P/PNNL—PN95048/1024	INTERPRETATION OF SINGLE-WELL TRACER TESTS	—

Table B-2. DOE Projects with Unknown Determination (continued)

Project ID	Project Title	Project Description
P/PNNL—PN95054/1030	MECHANISMS OF RADIOLYTIC DECOMPOSITION OF COMPLEX NUCLEAR WASTE FORMS	—
P/PPPL—27	ARC FURNACE	—
P/RL—FG06-89ER60845	ANIONIC COCONTAMINANTS AND THE BIOGEOCHEMICAL EVOLUTION OF AQUIFER HETEROGENEITY	Many important subsurface contaminants, such as organic acids, chelating agents, and metal ligand complexes are anions capable of binding to mineral surfaces. The transport of these compounds is greatly complicated by the high degree of physical and . . .
P/SNL—030940526	DEPLETED URANIUM WASTE MINIMIZATION AND MATERIAL REUTILIZATION.	Surrogate experiments and modeling electron beam melting and solidification uranium.
P/SNL—0679	DOE LLW ONFSB 94-2 IP	—
P/SNL—067950511	SYSTEMS ANALYSIS FOR THE ARCTIC NUCLEAR WASTE ASSESSMENT PROGRAM	Systems analysis of radioactive contamination by the former soviet union of the arctic ocean and adjacent seas.
P/SNL—0957	ALPHA LOW-LEVEL MIX WASTE	—
P/SNL—0958	IN SITU VISCOSITY AND DENSITY	The objective of this project is to develop and demonstrate a cost-effective, in situ monitor that measures viscosity and density of mixed-waste tank slurries. The project is scheduled for three years, FY-1996 through FY-1998. During the first fiscal year, . . .
P/SNL—0959	PROCESS MONITORING AND CONTROL	Develop and demonstrate near-infrared tunable-diode laser spectroscopy as a continuous emission monitor for ammonia in the effluent gases from DOE hazardous and mixed waste treatment processes.
P/SNL—1062	HANFORD SOLID/LIQUID WASTE	—
P/SNL—18930908	ELECTRON BEAM MELTING AND IN-PROCESS SCRAP RECYCLING OF URANIUM	Experimental program and model development of electron beam melting of uranium.
P/SNL—25910702LA	OPTICAL ORDNANCE DEVELOPMENT FOR NAVAL EXPLOSIVE ORDNANCE DISPOSAL APPLICATIONS	Development of an optical ordnance system.
P/SNL—3515350000	ORGANICALLY ENHANCED IN SITU ELECTROKINETIC REMOVAL OF URANIUM FROM SOILS	—

Table B-2. DOE Projects with Unknown Determination (continued)

Project ID	Project Title	Project Description
P/SNL—3515360000	NANOREACTORS AS NOVEL CATALYST SYSTEMS FOR WASTE STREAM REMEDIATION	
P/SNL—3523180000	INVESTIGATION OF SPRAY TECHNIQUES FOR USE IN EXPLOSIVE SCABBLING OF CONCRETE	—
P/SNL—3814	WASTE MANAGEMENT CONT OF OPS - CA	—
P/SNL—4772	LANDFILL CHARACTERIZATION SYSTEM	The system, we are developing includes all of the technologies for characterizing metal and mixed-waste contamination beneath the MWLID sites. Some of these elements which comprise the system are new or emerging technologies; others are more . . .
P/SNL—4775	DRY BARRIER APPLICATION FOR LANDFILLS	The objective of this project is to develop and demonstrate an air-enhanced dry barrier for application to landfills in arid environments. By drying a geologic (soil) layer, we decrease the unsaturated hydraulic conductivity and increase the storage . . .
P/SNL—4989	TITANATE ION EXCHANGES	The objective of this project is to develop crystalline silicotitanates (CSTs) to selectively remove cesium, including radioactive Cs-137, strontium and other radionuclides from radwastes. CSTs are a new class of inorganic ion exchanger invented by . . .
P/SNL—5029	CRYSTALLINE SILICOTITANATES	The objective of this project is to produce engineered form material of crystalline silico-titanates that is suitable for highly alkaline, column ion-exchange applications that selectively remove cesium, strontium, and other radionuclides.
P/SNL—5031	EVALUATION OF GROUT TECHNOLOGY	This program is to address the use of innovative grouts to form subsurface barriers for use in remediation of aqueous waste radiochemical contained in underground storage tanks at Hanford and elsewhere. The DOE has over 200 such underground . . .
P/SNL—5134	METAL EMISSIONS MONITOR	This project provides funds to develop and demonstrate a continuous monitor to measure metal emissions in the effluent from DOE waste treatment processes. This monitor is based on technique called Laser-Speak-Emission-Spectroscopy, which has been . . .

Table B-2. DOE Projects with Unknown Determination (continued)

Project ID	Project Title	Project Description
P/SNL—5334	ROBOTICS TECH DVLPMT PROG	Research and advanced development of intelligent systems and technologies for characterization and remediation of underground storage tanks and buried waste sites; automation of analytical chemistry laboratories; waste facility operations; waste . .
P/SNL—5335	ROBOTIC DEVELOPMENT FOR INTEGRATION DEMONSTRATION	Development and demonstration of intelligent systems and technologies in the area of Tank Waste Characterization and Tank Waste Remediation to support the Underground Storage Tank Integrated Demonstration program.
P/SNL—5649	DOE MIXED WASTE CHARACTERIZATION	—
P/SNL—5714	INTERNATIONAL SEPARATIONS	A small contract involving Russian separations technologies was established between SNL, SAIC, and the Khlopin Radium Institute in the summer of 1992. The work, directed towards the application of the cobalt dicarbollide process to the . . .
P/SNL—66931202	LANDFILL CHARACTERIZATION SYSTEM TECHNICAL DEMONSTRATION AT KAFB MIXED WASTE SITE	Demonstrate innovative, efficient site characterization approach.
P/SNL—83900813AA	CHEMICAL KINETICS OF SCWO	Establish the effect of water on the thermodynamic properties of reactants.
P/SNL—83921007	KINETIC MECHANISMS FOR SCWO (SERDP)	Fundamental studies of oxidation reactions in supercritical water.
P/SNL—8968	USE OF DU IN STRG/SHPPNG CASKS	The objective of this program is to evaluate viable options for future use and subsequent disposal of depleted uranium in the storage, shipping, and repository emplacement of vitrified high-level waste. The program will produce a number . . .
P/SNL—8982	ADVANCED TECHNOLOGY DEVELOPMENT	The purpose of this task is to provide DOE with the capability to conduct technical investigations of transportation systems, to design and develop packagings for radioactive materials, and to investigate promising new technologies and materials that . . .
P/SNL—W6227	RADIOLOGICAL CRITERIA FOR DECOMMISSIONING	Develop and implement software for evaluating decommissioning impacts.

Table B-2. DOE Projects with Unknown Determination (continued)

Project ID	Project Title	Project Description
P/SRTC—9500132002	CESIUM EXTRACTION TESTING	The task provided technical support to all DOE projects that were evaluating the performance of the resorcinol/formaldehyde resin. Two projects, the SKID demonstration at Savannah River and Cs Removal CPU Studies at Oak Ridge, are performing . . .
P/SRTC—9500132003	SPECTROMETER SYSTEMS FOR ONLINE MONITORING	Large quantities of hazardous waste have been generated and stored at DOE sites and at industrial facilities throughout the United States and the world. A number of processes are under consideration to reduce waste volume and transform the waste to . . .
P/SRTC—9500141005	SOL-GEL INDICATOR PROGRAM CHARACTERIZATION	Environmental Management activities at the DOE sites have created a need for continuous monitoring of contaminants in ground and surface water. Contaminants include toxic metals (U, Cr, Pb, Hg, Cd, etc.) And organics . . .
P/SRTC—9500141013	CHARACTERIZATION AND MONITORING FOR THE MAGNE	The Savannah River Site has been selected to host testing, development, and evaluation of technologies for in situ remediation of groundwater contaminated with metals. As part of the in situ technology demonstration a subsurface barrier will be . . .
P/SRTC—9500141014	IN SITU INORGANIC REMEDIATION OF GROUNDWATER	The Savannah River Site has been selected to host testing, development, and evaluation of technologies for in situ remediation of groundwater contaminated with metals. The first technologies to be demonstrated under this program are the . . .
P/SRTC—9500141015	MAG-SEP PROCESS CHEMISTRY SUPPORT	The Savannah River Site has been selected to host testing, development and evaluation of technilogies for in situ remediation of groundwater contaminated with metals. The first technologies to be demonstrated under this program are the Magnetic . . .
P/SRTC—9500141016	IN SITU CHARACTERIZATION OF HAZARDOUS SOIL	Prompt Gamma Neutron Activation Analysis for safe, real-time, in situ characterization of soil constituents at SRS has been investigated. A field study focused on use and validation of this technology for in situ characterization of . . .

Table B-2. DOE Projects with Unknown Determination (continued)

Project ID	Project Title	Project Description
P/SRTC—9500141019	BIOREMEDIATION OF TOXIC METALS	The principal objective of this task is to evaluate, with the aid of an industrial partner, Frisby Technologies, Inc., A new bioremediation scheme involving filter media consisting of biomass embedded in a hydrophilic foam matrix. Novel features of . . .
P/SRTC—9500141021	IN SITU MONITORING CAPABILITY	The Department of Energy spends large amounts of money to show compliance with environmental regulations by monitoring processes and the surrounding environment. The promulgation of more stringent regulations increases the amount of effort and . . .
P/SRTC—9500142005	HIGH TEMPERATURE DEMONSTRATIONS	This task provided laboratory and demonstration high-temperature vitrification units which could be utilized to demonstrate the effectiveness of remote, high temperature vitrification on existing low-level mixed, hazardous commercial and Department . . .
P/SRTC—9500142007	NOBILE METAL RECLAMATION	This task investigated the reclamation of noble and commercial metals from commercial and government electronic components using advanced process chemistry and available melter technology while immobilizing (via vitrification) hazardous materials . . .
P/SRTC—9500142008	VITRIFICATION PROCESS DEMO	Significant concentrations of transuranic elements are stored within the DOE Complex. Many actinide isotopes are highly radioactive and have long half-lives. Vitrification of transuranic bearing waste streams would encapsulate these species within a . . .
P/SRTC—9500142009	VITRIFICATION DEMO (FIELD SCALE)	The primary objective of this task was to demonstrate a stabilization treatment on an actual mixed hazardous and radioactive waste stream. The selected technology of choice was vitrification using processes developed at the Savannah River Technology . . .
P/SRTC—9500142015	REMOVAL AND DESTRUCTION OF ASBESTOS WASTE	During the next three decades, the Department of Energy will spend a significant portion of its budget decontaminating and decommissioning facilities within the weapons complex. These D&D activities will generate significant amounts of asbestos . . .

Table B-2. DOE Projects with Unknown Determination (continued)

Project ID	Project Title	Project Description
P/SRTC—9500144107	RADIOACTIVE POLYCHLORINATED BIPHENYL WASTE	The objective of this task was to identify and demonstrate an innovative technology for the destruction of polychlorinated biphenyl in radioactively contaminated solid waste. The program included developing techniques and protocol for safe . . .
P/SRTC—9500153002	PENETROMETER FOR SITE CHARACTERIZATION	This task developed a tool for Expedited Site Characterization described in needs statement CM-1 and investigated commercialization of that tool. Environmental Restoration and Waste Management activities at DOE sites . . .
P/SRTC—9502010101	REACTOR WASTE DECONTAMINATION	Developed needed analytical methods and analyzed special radioactive samples associated with characterization of radioactive waste for disposal from reactor areas and miscellaneous radioactive samples from the decontamination facility. Provided . . .
P/SRTC—9502010103	ANALYTICAL DEVELOPMENT FOR L-REACTOR WASTE	Developed needed analytical procedures and provided special analyses to Reactor Waste Management for analysis of radioactive waste handling samples from the L-Area Reactor Facilities.
P/SRTC—9502010104	ANALYTICAL DEVELOPMENT FOR P-REACTOR WASTE	Developed needed analytical procedures and provided special analyses to Reactor Waste Management for analysis of radioactive waste handling samples from the P-Area Reactor Facilities.
P/SRTC—9502010105	ANALYTICAL DEVELOPMENT FOR D2O AREA	Provided analytical development and specialized analyses to Reactor Waste Management for analysis of radioactive waste handling samples, and for samples from the D-20 Area Facilities. Conducted analyses of contaminated heavy water and resin . . .
P/SRTC—9502010307	ANALYTICAL INSTRUMENTS FOR TRITIUM OPERATIONS	Developed on-line instrumentation for use in facility processing radioactive gas; troubleshot developed instrumentation (for mercury and moisture measurements). Performed special analyses on operations and research samples contaminated with . . .
P/SRTC—9502010403	FISSILE WASTE MONITOR ANALYTICAL DEVELOPMENT	Developed methods/instrumentation for assay of special waste items for special nuclear material content; troubleshoot radioactive waste box monitor.

Table B-2. DOE Projects with Unknown Determination (continued)

Project ID	Project Title	Project Description
P/SRTC—9502010405	F-AREA ANALYTICAL SAMPLES	Developed and applied analytical methods for analysis of special radioactive waste samples from miscellaneous F-Area facilities.
P/SRTC—9502010603	SALTSTONE FACILITY ANALYTICAL DEVELOPMENT	Provided development and troubleshooting of analytical methods needed for operation of the saltstone storage vaults at the Savannah River Site. The saltstone process contains low-levels of radioactivity, but potentially could be highly contaminated . . .
P/SRTC—9502011101	SOLID RADIOACTIVE WASTE CHARACTERIZATION	Developed chemical and nuclear analytical methods to characterize solid radioactive waste. Developed waste acceptance plans based on facility characteristic radioactivity survey and analysis schemes.
P/SRTC—9502A10401	F-CANYON PLUTONIUM ANALYTICAL DEVELOPMENT	Developed analytical methods, on-line methods/instruments, including improved monitors for radioactive contaminated liquid and gaseous effluents; troubleshot monitors developed for the Nuclear Materials Stabilization Program F-Canyon plutonium . . .
P/SRTC—9502A10402	FB-LINE PLUTONIUM ANALYTICAL DEVELOPMENT	Developed analytical methods, on-line methods/instruments, including monitors for radioactive liquid and gaseous effluents; troubleshot monitors developed for the Nuclear Materials Stabilization Program (FB-line) for processing plutonium. Developed . . .
P/SRTC—9502A10411	H-CANYON SNM STABILIZATION ANALYTICAL	Developed analytical methods and on-line methods/instruments, including monitors for radioactive liquid and gaseous effluents; troubleshot monitors developed for the Nuclear Materials Stabilization Program (H-canyon) for processing plutonium and . . .
P/SRTC—9502A10412	PLUTONIUM 238 ANALYTICAL DEVELOPMENT	Developed analytical methods and on-line methods/instruments including improved monitors for liquid and gaseous effluents. Developed analytical methods to improve analyses for Pu-238 impurities. Performed R&D for the SRTC Analytical Laboratories . . .
P/SRTC—9502A10602	ANALYTICAL METHOD DEVELOPMENT FOR DWPF	This activity provided for the development of analytical methods needed for operation of the Defense Waste Processing Facility (DWPF) at the Savannah River Site and for analysis of R&D and special radioactive samples from the DWPF operations.

Table B-2. DOE Projects with Unknown Determination (continued)

Project ID	Project Title	Project Description
P/SRTC—9502B10602	CHEMICAL TECHNOLOGY SUPPORT FOR DWPF	Provided technical expertise to the SRTC Defense Waste Process Technology Section, specifically, assisting in developing an understanding of Cu catylyst behavior for in-tank processing of high-level waste.
P/SRTC—9508010401	AMERICIUM/CURIUM STABILIZATION	Americium/Curium (Am/Cm) solution is currently being stored at SRS. Per Defense Nuclear Facilities Safety Board (DNFSB) recommendation 94-1, this solution must be put in a form more suitable for safe interim storage. Several options were considered, . . .
P/SRTC—9508C10402	BAGLESS TRANSFER SYSTEM DESIGN	To aid in compliance with the Department of Energy's long term plutonium storage criteria, a system has been developed that will safely package plutonium while avoiding the use of plastic bags and other organic materials. These materials, used in . . .
P/SRTC—9602103008	FAST GAS ANALYZER SYSTEM	It is desired to be able to measure gaseous isotopic distribution in a fast moving stream. A laser Raman system was previously developed to accomplish this task. In FY-1996 this system was tested against a process mass spectrometer system. The mass . . .
P/SRTC—9602105001	ANALYTICAL METHOD FOR TOTAL ALPHA AND TOTAL PU MEASUREMENTS	A technique was developed to measure total alpha and total Pu in matrices where the beta activity and salt content of the samples can be very high. Historically, high salt content in waste samples has degraded prepared alpha plates to the extent that . . .
P/SRTC—9602105002	LIQUID SCINTILLATION COUNTING METHOD FOR HIGH SALT CONTENT RADIOACTIVE WASTE SAMPLES.	As a consequence of waste vitrification efforts underway at the Savannah River Site, a need has arisen to obtain gross alpha and gross beta values in samples with high salt contents, and high beta activities. A less time-consuming method than the . . .
P/SRTC—9602105003	SEPARATION AND MEASUREMENT METHOD OF I-129 IN NUCLEAR WASTE	The method to measure I-129 in nuclear wastes separates anion complexes of sulfur, iodine, and phosphorus radioisotopes to enable measurement by radiochemical techniques. This method involves ion chromatographic separation of the anion complexes from . . .

Table B-2. DOE Projects with Unknown Determination (continued)

Project ID	Project Title	Project Description
P/SRTC—9602105005	REMOTE TURBIDITY METER	A commercially available turbidity meter was modified to accept fiber optic input. The custom instrument is used to measure turbidity in radioactive waste sludge settling studies performed in a shielded cell. The modification allows the measurement . . .
P/SRTC—9602106001	RADIOACTIVE SAMPLE VIAL INSERT	A sample vial insert was designed and tested to allow sampling of a radioactive slurry stream with a standard sampler but with reduction of volume such that all of the sample can be analyzed without introducing dilution and washing errors. The insert . . .
P/SRTC—9602106003	ON-LINE NITRITE AND BENZENE ANALYZER SYSTEM	A real-time analyzer system was developed to analyze a radioactive liquid process stream for nitrite and benzene concentration. The nitrite is measured in the liquid phase; the benzene is air-purged into the vapor phase and measured. Both are . . .
P/SRTC—9602108004	SOL GEL SENSOR FOR CHROMIUM	A fiber optic sensor for measurement of chromium in solution was developed by incorporating 5-Diphenylcarbazide indicator in a sol gel matrix. The sol gel indicator is coated on the end of a fiber optic lens and chromium concentration is measured by . . .
P/SRTC—9602108005	SOL GEL SENSOR FOR URANIUM	A fiber optic sensor for measurement of uranium in solution was developed by incorporating Arsenazo II indicator in a sol gel matrix. The sol gel indicator is coated on the end of a fiber optic lens and uranium concentration is measured by absorption . . .
P/SRTC—9602111001	WET CHEMICAL OXIDATION AS AN ALTERNATIVE TO INCINERATION	Further development was performed in FY-1996 on nitric-phosphoric acid oxidation of organics. Extensive work was completed in the area of acid off-gas handling in which researchers studied nitric acid depletion rates, ways to remove HCl from the off-gas, . . .
P/SRTC—9603108000	FISSILE MATERIALS DISPOSITION - PLUTONIUM VITRIFICATION	SRTC provides the technology for vitrification of excess plutonium (Pu) in a glass waste form that meets non-proliferation objectives. Glass formulations have been developed that demonstrate the feasibility of incorporating 10 weight percent . . .

Table B-2. DOE Projects with Unknown Determination (continued)

Project ID	Project Title	Project Description
P/SRTC—9608010105	AUTOMATED DIGITAL RADIOGRAPHY INSPECTION SYSTEM	During FY-1996, the Automated Digital Radiography Inspection System was used to verify the quality of circumferential welds of new, replacement storage drums. Code quality DR images were produced, evaluated, and stored for 1800+ drums (over) . . .
P/SRTC—9608011102	SOLID WASTE VERIFICAITON BY DIGITAL RADIOGRAPHY	During FY-1996, the WSRC Solid Waste Engineering department requested SRTC to provide a . . .
P/SRTC—9608013000	HORIZONTAL SUBSURFACE BARRIER DEVELOPMENT	Technology is being developed and demonstrated to place a horizontal subsurface barrier selectively along a fracture plane with minimal or no disruption of the soil over burden. The barrier material is selected so as to limit the penetration of . . .
P/SRTC—9608013100	ELECTRO-CHEMICAL DESTRUCTION OF NITRATES AND NITRITES	An electro-chemical reactor is being utilized to evaluate the capability to decompose nitrates and nitrites in waste streams. Nitrate and nitrite compounds tend to reduce the effectiveness of glass as a waste disposal medium. Pilot studies have been . . .

APPENDIX C

Treatability Study Database Analysis

Table C-1. Treatability Studies with Yes Determination

TS-ID No.	Technology Name	Technology Description
TS-IN003	Decontamination	—
TS-LL001	Filtration	There were three general categories of wastes treated during the filtration treatability study: oils, mixed chlorosolvents, and coolant wash waters. However, LLNL considered each container of waste used in the treatability study a separate waste stream due to the variability of the processes which generated the waste. All three categories of wastes contained radioactive isotopes (less than 1 millicurie per drum) consisting of U-238, D-38, and/or Th-228. Some of these wastes also contained low concentrations of other metals including copper, beryliium, chromium, and nickel and/or zinc. The oils consisted of mechanical vacuum pump oils, diffusion pump oils, and part machining oils. These oils were generally a mixture of machine lubricants (tramp oils) and cutting fluids contaminated with water, chlorosolvents, metals, and radioactive isotopes. In some cases there was a significant amount of water (45-% by volume) mixed in with the oils. The chlorosolvents were degreasers, polishing aand cutting fluids contaminated with water and metals. These chlorosolvents consisted of trichloroethylene (TCE) ; 1,1,1-trichloroethane (TCA); dichloroethylene (DCE); and perchloroethylene (PCE).
TS-LL002	Filtration	LLNL rented a membrane filtration system from Golder Associates in April 1995 specifically for the purpose of performing these studies. Samples of coolant wash waters from one 55 gallon container were used to begin the filtration treatability study. The coolant wash waters were primarily cutting fluids consisting of water and ethanolamines, contaminted with oil, chlorosolvents, metals and radioactive isotopes. These wastes are approximately 95% water (by volume). The samples contained radioactive isotopes (less than 1 millicurie per drum) consisting of U-238. The samples also contained low concentrations of other metals including lead and beryllium.
TS-SA004	Neutralization/ Stabilization	Products produced by Fluid Tech, Inc. (Aqueous, Pedrosed, Aquaset II, Petroset II, Aquaset-H, and Petroses-H) are used to solidify small-volume liquid mixed wate streams that contain low levels of radio-activity (principal radionulcides are Co-60, Cs-137, H3, and uranium isotopes). The silica-based solidification compounds are uniqually suited (the SNL/NM waste streams) because they can be used to solidify mixtures of organic and aqueous liquids.
TS-WV001	Cement Solidification	The purpose of cement solidification is to solidify the decontaminated liquid waste into a waste form suitable for storage and disposal as Low-Level Class C waste which meets the stability requirements of 10CFR61 and the NRC Branch Technical position on Waste Form Qualificiation.

Table C-2. Treatability Studies with Unknown Determination

TS-ID No.	Technology Name	Technology Description
TS-LA001	Electorchemical Treatment	This process electrochemically extracts metals from metalic solutions and deposits the metals on cathodes, oxidizes cynaides to cyanates, and oxidizes organic compounds to carbon dioxide, water, and other nontoxic organic compounds.
TS-LL003	Stabilization	Stabilization treatability studies are continually conducted to evaluate the effectiveness of different stabilizing media in binding the toxic contaminants of various mexied wastes. The stabilizing media includes clays, dithiocarbonates and thiocarbonates.
TS-OR002	Ion Exchange	—
TS-OR003	Soil Mixing	—
TS-OR008	Grouting	—
TS-RF003	Microwave Solidification	—
TS-RF006	Tablet Pressing	—
TS-RF010	Microwave Solidification	—
TS-RF013	Cementation	—
TS-RF019	Cementation	—
TS-RF020	Solidification/Stabilization	—
TS-SA002	Electrokinetics	Porous ceramic-cased electrodes are installed in contact with soil within a small test cell (21.5cm X 15.3cm X 1.9cm). A small DC current is passed through the soil which causes chromate ions to migrate to and through the porous ceramic casing at the anode where it is purged out of the electrode assembly.
TS-SR006	Incinerator Ash Vitrification	—
TS-SR007	Waste Vitrification	—
TS-SR008	Cement Stabilization	—
TS-WV002	Supernatant Decontamination	A new class of inorganic ion exchangers called Crystalline Silicotitanates (CST), were invented by researchers at Sandia National Laboratories and Texas A&M University. The materials exhibit a high selectivity for the ion exchange of cesium, strontium, and several other radionuclides from highly alkaline solutions containing molar concentrations of Na+. Sandia and UOP teamed under a Cooperative Research Development Agreement (CRADA) to develop an engineered form of CST suitable for column ion exchange use. The selectivity and stability of the CSTs have made them candidates for treatment of DOE high level radioactive waste tank supernatants.
TS-YP002	Final Waste Form I	—
TS-YP004	MWTF Leaching Study	—