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WASTE DISPOSAL
AND PROCESSING
(TID-4500)

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RADIOACTIVE WASTE PROCESSING AND DISPOSAL

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An Annotated Bibliography of Selected Literature

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ABSTRACT

These selected 1,012 annotated references supplement TID-3311, TID-3311 (Suppl. 1), TID-3555, and TID-3555(Suppl. 1). Unclassified report and journal literature from February 1965 through July 15, 1966 is covered. The references have been subject-arranged and provided with numerous cross references. Author, corporate author, and report number-availability indexes are included.

INTRODUCTION

The protection of man from radioactive wastes presents a new and unique challenge in that the hazardous nature of these wastes can not be dissipated by simple chemical reactions.

Storage that permits decay of radiation-emitting isotopes, dilution of radioisotope-laden wastes to permissible limits of concentration, and immobilization of these wastes are the acceptable and necessary processes of waste disposal.

Originally, wastes were stored in large underground tanks. The costs involved, anticipated volumes, and necessity for long-term storage soon indicated that more economical and practical methods of disposal were required.

Present research is directed toward waste disposal methods that will provide maximum safety with minimum costs. Volume reduction combined with ultimate disposal of the concentrates in abandoned salt mines appears to offer the best solution to this enigma. Large volumes of liquid wastes are being released into subterranean formations while solids are incinerated, baled, or sealed in concrete prior to land or sea burial.

Recovery of industrially and medically useful fission products from wastes before disposal continues to reduce both the over-all expense of disposal and the radiation levels of the wastes.

The many references in this compilation and in its complementing predecessors, TID-3311, TID-3311 (Suppl. 1), TID-3555, and TID-3555(Suppl. 1), indicate the great effort being expended to protect humanity while developing a useful new industry.

REFERENCES

GENERAL

COMMERCIAL UTILIZATION

1 (HW-SA-3080(Rev.)) THE FISSION PRODUCTS. C. A. Rohrmann (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). May 8, 1964. Contract AT(45-1)-1350. 17p. (CONF-635-1)

From AEC Atomic Industrial Forum, Washington, D. C., May 1964.

Radioisotopes as potential heat sources are discussed briefly. Of the usable radioisotopes, four are fission products: ^{144}Ce, ^{137}Cs, ^{147}Pm, and ^{90}Sr. The separation of these fission products from wastes at Hanford is discussed, and the results of an engineering study of the economics of such separation in the future are summarized. The processes for carrying out the separation and purification are described. Separation of other isotopes from power reactor fuel reprocessing is also considered.

2 (ORNL-P-1944) PREPARATION OF MEGACURIE QUANTITIES OF ^{137}Cs SALTS AND SOURCE COMPOUNDS AT OAK RIDGE NATIONAL LABORATORY. Lewis, R. E.; Ottinger, C. L. (Oak Ridge National Lab., Tenn.). [1965]. Contract W-7405-eng-26. 13p. (CONF-660305-4). Dep. mn. CFSTI $1.00 cy, $0.50 mn.

From American Nuclear Society Topical Meeting, Augusta, Ga.

Processes used for recovery, purification, and conversion of megacurie quantities of ^{137}Cs to salts and source compounds are reviewed with special emphasis on the conversion of cesium alum to cesium carbonate by the cesium tetraoxalate process. Subsequent conversion of cesium carbonate to source compounds is discussed.

3 (RL-SA-59) LARGE-SCALE PROCESSING AND SOURCE PREPARATION OF SEPARATED FISSION PRODUCTS. Beard, S. J.; Smith, P. W. (General Electric Co., Richland, Wash. Hanford Atomic Products Operation); Cochran, J. S. (Isochem Inc., Richland, Wash). Dec. 6, 1965. Contract AT(45-1)-1851. 22p. (CONF-660305-3). Dep. mn. CFSTI $1.00 cy, $0.50 mn.

For Presentation at American Nuclear Society Topical Meeting, Augusta, Ga.

Fission product recovery activities currently being conducted at Hanford are discussed, and the processes used are described, and future plans for encapsulation of fission products in the commercial FPCE to be built at Hanford are outlined. Production plant flowsheet performance is discussed for the individual fission product recovery processes. Full level plant tests of improved fission product recovery and purification flow sheets are reviewed.

4 (TID-7698(Pt.I)(p.1-16)) THE FISSION PRODUCTS. C. A. Rohrmann (Hanford Atomic Products Operation, Richland, Wash.).

Of the 12 or 14 radioisotopes which have potential as heat sources, four are fission products (^{90}Sr, ^{137}Cs, ^{144}Ce, and ^{147}Pm). The properties of these radioisotopes and fission products are tabulated. The Hanford Waste Management Program is described briefly. The cost results of the engineering study for the Hanford Isotope Plant are discussed. The separation of isotopes from irradiated power reactor fuels is considered.

5 ON THE HEAT RELEASE FROM HIGHLY RADIOACTIVE SOLIDS IN RELATION TO THE PROBLEM OF WASTE DISPOSAL AND UTILIZATION. P. V. Zimakov, B. S. Kolychev, V. V. Kulichenko, and Yu. P. Martynov. At. Energ. (USSR), 18: 428-31(Apr. 1965). (In Russian)

An attempt was made to utilize the heat released from buried highly radioactive waste by melting vitreous preparations and retaining the necessary temperature in the burial container. Descriptions are given of an installation designed for electromodeling the heat release from melted highly-radioactive materials and for determining the temperature dependence in the burial container on the quantity and specific activity of the buried block. Graphs are shown of the amount of fixation material which can be melted in the burial container containing 1 m^3 melted fixation mixture with the initial activity of 30,000 C/dm^3. It is shown that at the given temperature in the burial ground of highly active blocks it is possible to melt also large quantities of low activity calcinated wastes. The properties of melted highly active fission concentrates can be used for various purposes such as isotopic sources of power.

6 A NEW AND IMPROVED METHOD OF GENERATING ELECTRICITY. (to Yardney International Corp.). British Patent 1,010,479. Nov. 17, 1965. Priority date July 12, 1961, United States.

A method is described for generating electricity in which a pair of electrodes are placed in a liquid electrolyte, a radioactive material is used to establish a higher radiation level in one electrode than the other to generate an emf between the electrodes, and the electrodes are connected to an external circuit. The electrodes may be placed in a stream of water containing radioactive material, or the radioactive material may be placed in a container that constitutes one of the electrodes and the filled container placed in a stream of water. In either case, the electrode having the lower radiation level must be protected. Radioactive wastes may be used to generate electricity by this method.

BIBLIOGRAPHIES, SURVEYS, BOOKS, AND PATENTS

See also the following references:

6, 79, 81, 86, 87, 123, 182, 238, 354, 355, 629, 979

7 (A/CONF.28/P/868) ADVANCES IN RADIOACTIVE WASTE MANAGEMENT TECHNOLOGY—ITS EFFECT ON THE FUTURE U.S. NUCLEAR POWER INDUSTRY. W. G. Belter (Atomic Energy Commission, Washington, D. C.). 18p.

Prepared for the United Nations Third International Conference on the Peaceful Uses of Atomic Energy, 1964.

A discussion is presented on the advances in treatment and disposal of low- and high-level wastes, and their probable effects on the future nuclear power industry. Ion exchange and foam separation techniques are being applied to low-level wastes on an engineering pilot plant scale. The disposal of intermediate-level wastes into shale by hydraulic fracturing is being investigated. Calcination and fixation of high-level wastes are discussed. The feasibility of storing high-level wastes in salt mines and bedrock caverns is being studied. Engineering evaluation studies relative to the nuclear power economy in the future, indicate waste treatment and ultimate storage costs of ~0.03 mill/kwh, less than 1% of the cost of nuclear power in a 4 mill/kwh economy. (D.L.C.)

8 (AED-C-01-12) AUSGEWAEHLTES SCHRIFTTUM NACH SACHGEBIETEN. BAUWESEN. BESEITIGUNG RADIOAKTIVER ABFALLSTOFFE. TEIL I. (Selected Literature According to Subject Field. Construction Engineering. Disposal of Radioactive Waste Material. Part I). (Zentralstelle fuer Atomkernenergie-Dokumentation, Gmelin-Institut fuer Anorganische Chemie und Grenzgebiete, Frankfurt am Main (West Germany)). 1965. 142p. Dep. mn.

Abstracts of 127 papers dealing with various aspects of the disposal of radioactive waste material are presented.

9 (AED-C-01-13) AUSGEWAEHLTES SCHRIFTTUM NACH SACHGEBIETEN. BAUWESEN. BESEITIGUNG RADIOAKTIVER ABFALLSTOFFE. TEIL II. (Selected Literature According to Subject Field. Construction Engineering. Disposal of Radioactive Waste Material. Part II). (Zentralstelle fuer Atomkernenergie-Dokumentation, Gmelin-Institut fuer Anorganische Chemie und Grenzgebiete, Frankfurt am Main (West Germany)). 1965. 99p. Dep. mn.

Abstracts of 84 papers on various aspects of radioactive waste disposal are presented.

10 (CEA-Bib.-59) QUELQUES ASPECTS DE RADIOHYDROLOGIE. Etude Bibliographique. (Some Aspects of Radiohydrology. Literature Survey). Bittel, Robert (Commissariat a l'Energie Atomique, Saclay (France). Centre d'Etudes Nucleaires). Sept. 1965. 46p. Dep. mn.

Various aspects of hydrological problems arising from the presence of radioisotopes in rivers between the point of disposal and the places at which the water is used by man are reviewed. 151 references are cited.

11 REVIEW OF NEW DEVELOPMENTS IN THE FIELD OF LOW AND INTERMEDIATE RADIOACTIVE WASTE DISPOSAL IN THE REPUBLIC OF SOUTH AFRICA. Barbour, R. A. (South African Atomic Energy Board, Pelindaba). Vienna, International Atomic Energy Agency, 1965, Preprint No. SM-71/63(B), 9p. (CONF-651202-10). ORAU.

From Symposium on Practices in the Treatment of Low and Intermediate Level Radioactive Waste, Vienna.

A very general description of all the waste processing and disposal facilities at Pelindaba, South Africa is presented. The treatment of all levels of solid and liquid wastes is described.

12 A REVIEW OF RESEARCH AND DEVELOPMENT FOR TREATMENT OF LOW AND INTERMEDIATE LEVEL RADIOACTIVE WASTES. Thomas, K. T. (Atomic Energy Establishment, Trombay, India). Vienna, International Atomic Energy Agency, 1965, Preprint No. SM-71/63(A), 12p. (CONF-651202-44). ORAU.

From Symposium on Practices in the Treatment of Low and Intermediate Level Radioactive Waste, Vienna.

This paper summarizes and reviews the work being done in connection with the treatment and disposal of low and intermediate level radioactive effluents. The following aspects of the waste disposal problem are briefly summarized: chemical treatment; ion exchange; foam separation; atomization suspension technique; evaporation; disposal of TBP/diluent wastes; fixation in concrete, asphalt, or other solid media, and gas cleaning studies. In all aspects, the emphasis is on using materials locally available in India for the processing.

13 (HW-81481) WASTE MANAGEMENT PROGRAM—CHEMICAL PROCESSING DEPARTMENT. S. J. Beard, R. J. Kofoed, J. J. Shefcik, and P. W. Smith, comps. (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Mar. 24, 1964. Contract AT(45-1)-1350. 59p.

During the nineteen years of Hanford operation, waste management practices have included the dispersal of gaseous wastes in the atmosphere, the burial of solid wastes in the ground, the discharge of liquid wastes of low and intermediate radioactivity level to the ground, and the accumulation and confinement of liquid wastes of high radioactivity level in the underground tanks. During the past several years, the Chemical Processing Department has formulated a program to achieve significant improvements in the handling of high activity level radioactive wastes. This program has the objectives of assuring the safe confinement of the radioactive materials for centuries, minimizing the long-term cost of the storage and surveillance operations, and making selected fission products available for beneficial use. The improvement program has been approved in principle by the Atomic Energy Commission and is now being implemented. If the program is carried out as planned, it is expected that all high activity level liquid wastes will be solidified. The most radioactive wastes will be processed for cesium and strontium removal and packaging before the remainder is solidified. Processing of wastes will be on a current basis by about 1975. (auth)

14 (HW-83609) PROGRESS IN FISSION-PRODUCT RECOVERY AT HANFORD. A SUMMARY OF HANFORD LABORATORIES ACHIEVEMENTS IN THIS PROGRAM UNDER GENERAL ELECTRIC, 1954-1964. R. L. Moore (General Electric Co., Richland, Wash. Hanford Atomic Products Operation). Nov. 1964. Contract AT(45-1)-1350. 28p. Dep.(mn); $2.00(cy), 1(mn) CFSTI.

Development of processes for Cs, Sr, Ce, Pm, and Tc recovery is summarized. Discussions are included con-

cerning the CSREX process, waste management, and isotope production. A bibliography of 78 references is included. (J.R.D.)

15 (HW-83610) PROGRESS IN RADIOACTIVE RESIDUE PROCESS DEVELOPMENT. A Summary of Hanford Laboratories Achievements in This Program Under General Electric, 1958-1964. A. M. Platt and C. R. Cooley (General Electric Co., Richland, Wash. Hanford Atomic Products Operation). Nov. 1964. Contract AT (45-1)-1350. 36p. Dep.; $2.00(cy), 1(mn) CFSTI.

Results are summarized for studies on high-level waste calcination, hot cell pilot plants, waste solidification, and methods for intermediate-level waste treatment. (J.R.D.)

16 (HW-SA-3279) THE MANAGEMENT OF HIGH-LEVEL RADIOACTIVE WASTES. Allison M. Platt (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Nov. 14, 1963. Contract AT(45-1)-1350. 19p.

A survey is presented of current practices of USAEC and its contractors for managing high-level radioactive wastes. The future waste problems associated with a growing nuclear power industry are discussed, and the progress of research and development activities on solidification of wastes is summarized. (D.L.C.)

17 (JAERI-4025) PURIFICATION OF HELIUM COOLANT. Keiichi Oshima, Keiji Naito, and Keiichi Nishida (Japan Atomic Energy Research Inst., Tokyo). Feb. 1963. 41p. (In Japanese)

The literature from 1955 to 1961 on the purification of helium coolant, especially on the separation of fission gases, is reviewed. The principle of gas purification, the separation of radioactive and nonradioactive gases, and purification systems for helium coolant are covered. 184 references. (D.C.W.)

18 (NP-15267) WASTE MANAGEMENT RESEARCH ABSTRACTS NO. 1. (International Atomic Energy Agency, Vienna (Austria)). 1965. 75p. Dep. mn; IAEA Free.

Abstracts are presented on national waste management research projects which were completed in 1963–64 by Czechoslovakia, Germany, Italy, Japan, Sweden, United Kingdom, USA, and USSR. The abstracts are arranged into the following categories: gaseous waste treatment and disposal, liquid waste treatment, solid waste treatment, storage and discharge studies, and economic studies. (D.L.C.)

19 WASTE MANAGEMENT. Blomeke, John O.; Roberts, J. Tom (Oak Ridge National Lab., Tenn.). Ann. Rev. Nucl. Sci., 15: 151-74(1965). (ORNL-P-1169).

A survey is presented of waste management problems associated with the fuel processing requirements of the nuclear power industry, which includes a review of existing techniques, together with the needs and prospects for new methods.

20 (ORNL-TM-1289) RECENT DEVELOPMENTS IN TREATING LOW AND INTERMEDIATE LEVEL RADIOACTIVE WASTE IN THE UNITED STATES. Blanco, R. E.; Alkire, G. J.; Davis, W. Jr.; Godbee, H. W.; Irish, E. R.; King, L. J.; Mercer, B. W.; Roberts, J. T.; Yee, W. C. (Oak Ridge National Lab., Tenn.). Dec. 1, 1965. Contract W-7405-eng-26. 52p. (CONF-651202-11). Dep. mn. CFSTI $3.00 cy, $0.50 mn.

From Symposium on Practices in the Treatment of Low and Intermediate Level Radioactive Waste, Vienna.

Recent research and development work in the treatment of low- and intermediate-level radioactive wastes is surveyed and brief economic evaluations of new treatment methods are included. The methods discussed include a scavenging-precipitation ion exchange process for treating low level waste; a scavenging-precipitation foam separation process; a process for treating process condensate wastes containing ammonia and organic materials; and a process for incorporation of intermediate level wastes in asphalt.

21 (SB-448(Rev.)) AIR POLLUTION AND PURIFICATION. OTS Selective Bibliography. (Office of Technical Services, Washington, D. C.). [nd]. 26p.

Three-hundred sixty-four references are presented to reports dealing with air pollution by fumes, industrial gases, smokes, and aerosols. A special section of Be air contamination is included. References to two air pollution bibliographies are included. (C.H.)

22 THE PROBLEM OF FISSION PRODUCTS. ELIMINATION OR RECOVERY? Francoise Laveissière and André Raggenbass (CEN, Saclay, France). Euratom Bull., 3: No. 4, 12-16(Dec. 1964).

The problem of fission-product formation in reactors is reviewed. Possible solutions — long-term storage, storage through conversion to solid form, separation of specific isotopes for storage or use, and uses as radiation or heat sources — are surveyed. (M.J.T.)

23 (TID-8541, pp 149-248) RADIOACTIVE WASTE MANAGEMENT. Belter, W. G. (Division of Reactor Development and Technology (AEC), Washington, D. C.); Pearce, D. W. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.).

A review of highlights of recent waste-management experience in the U. S. and abroad is presented. Advances in technology that have occurred during the past three to four years were summarized.

24 ION REMOVAL AND ITS APPLICATION TO RADIOACTIVE EFFLUENT TREATMENT. D. Pepper (Atomic Energy Research Establishment, Harwell, Berks, Eng.). Brit. Chem., Eng., 9: 510-16(Aug. 1964).

Methods available for removing ions from solution are reviewed. The mechanism of each method is briefly described and the possibility of applying the processes to radioactive effluent treatment is discussed, the costs of some effluent treatment schemes being estimated. (auth)

25 METHODS FOR THE TREATMENT OF ATOMIC WASTES. T. V. Arden and J. Pilot. J. Inst. Public Health Engrs., 62: 267-82(1963).

In a review of methods of treatment for low-activity atomic wastes, methods for the decontamination by coagulation and filtration and by ion-exchange resins are described. Both methods were inefficient in the removal of anionic activity. An experimental electrodialysis plant at Harwell is expected to produce an effluent containing 1 ppm or less of dissolved solids. (UKAEA)

26 A REVIEW OF THE LITERATURE OF 1961 ON WASTEWATER AND WATER POLLUTION CONTROL. D. A. Okun, C. M. Weiss, T. A. Alspaugh, et al. J. Water Pollution Control Federation, 34: 525-64(June 1962).

The literature of 1961 on water pollution control is reviewed. Emphasis is placed on methods for the treatment

of industrial wastes including radioactive wastes. 360 references. (C.H.)

27 WASTE MANAGEMENT: TECHNOLOGICAL ADVANCES AND ATTITUDES OF SAFETY. W. G. Belter, D. E. Ferguson, and F. L. Culler. Nucl. News, 7: No. 10, 94-7(Oct. 1964).

A summary is given of the Geneva Conference papers on waste management including the technological advances that were reported and the changes in safety attitudes that have occurred. (C.E.S.)

28 ARTIFICIAL AND NATURAL RADIOISOTOPES IN THE MARINE ENVIRONMENT. J. Mauchline and W. L. Templeton (Marine Station, Millport, Scotland and United Kingdom Atomic Energy Authority, Windscale Works, Eng.). Oceanogr. Mar. Biol. Ann. Rev., 2: 229-79(1964).

A review is presented on the content, measurement, behavior, and source of artificial and natural radioisotopes in the marine environment. Approximately 600 references are included.

29 (TID-3910(Suppl.2)) TERRESTRIAL AND FRESHWATER RADIOECOLOGY. A Selected Bibliography (Supplement 2). Alfred W. Klement, Jr. and Vincent Schultz (Division of Biology and Medicine, AEC). Mar. 1964. 125p.

A list is presented of 1280 references to the literature on terrestrial and fresh water radioecology. Emphasis is placed on publications issued since Feb. 1963. (C.H.)

30 (TID-3910(Suppl.3)) TERRESTRIAL AND FRESHWATER RADIOECOLOGY. A Selected Bibliography— Supplement 3. Alfred W. Klement, Jr., and Vincent Schultz (Division of Biology and Medicine (AEC), Washington, D. C.). Feb. 1965. 119p. Dep.; $2:50(cy), 2(mn) CFSTI.

A list is presented of 1207 publications in the field of radioecology, including the biological aspects of radioactive fallout, waste disposal, terrestrial radioecology, and freshwater radioecology. An attempt was made to reference publications related to the field or laboratory studies of wild species of plants and animals with respect to radiation effects or metabolic studies involving radionuclides. Complete information concerning reference sources and brief content notes are included. (C.H.)

31 BACKGROUND INFORMATION ON ATOMIC POWER WASTE HANDLING. John F. Hogerton, James G. Cline, Robert W. Kupp, and Charles B. Yulish. New York, Atomic Industrial Forum, 1965. 50p. $1.50.

The handling and disposal of the radioactive matter routinely formed in the course of atomic power generation is considered. On-site handling of wastes and effluents, shipment of radioactive materials from the power plant, processing of spent fuel, and ultimate disposal of radioactive wastes are discussed. An appendix summarizes pertinent background information on radioactivity and radiation protection. (M.C.G.)

32 A COLLECTION OF PAPERS ON NUCLEAR SCIENCE AND ENGINEERING. Inam-ur-Rahman. Lahore, Atomic Energy Centre, 1963. 497p.

Nuclear propulsion of ships and aircraft, analysis of nuclear reactor accidents, pressurized water reactors, interaction of neutrons with different elements, disposal of radioactive wastes, fusion as a source of energy, fast reactors, and selection of suitable site for power reactor installation in Pakistan are discussed in the 8 papers presented. (C.E.S.)

33 DEZAKTIVALITSIYA SBROSNYKH VOD RADIOKHIMICHESKIKH LABORATORII I EKSPERIMENTAL'NYKH ATOMNYKH REAKTOROV. (Decontamination of Waste Liquid from Radiochemical Laboratories and Experimental Atomic Reactors). A. A. Khonikevich. Moscow, Atomizdat, 1964. 110p.

A review is given of problems in decontaminating weakly active waste liquids. The effects of chemical and radiochemical compositions on the efficiency of decontamination are discussed. Installations and devices used for decontamination and their technological design and performance are also considered. 107 references. (R.V.J.)

34 LOW-LEVEL RADIOACTIVE WASTES— THEIR HANDLING, TREATMENT, AND DISPOSAL. Conrad P. Straub (Robert A. Taft Sanitary Engineering Center, Cincinnati). 1964. 444p. $1.50(GPO).

This book describes techniques for the handling, treatment, and disposal of low-level wastes to keep radiation exposure to a practicable minimum. Discharge to water, soil, and air is considered, and the removal of radioactivity by water treatment processes, biological processes, chemical processes, etc., is discussed. Finally, public health implications are considered. (D.L.C.)

35 MANAGEMENT OF RADIOACTIVE WASTES. Mawson, C. A. Van Nostrand Nuclear Science Series. Princeton, N. J., D. Van Nostrand Company, Inc., 1965. 203p. $6.95.

Information is included on general principles of waste management; sources and nature of radioactive wastes; waste management in the uranium industry; treatment of gaseous effluent; storage and fixation of liquid wastes; treatment of liquid wastes; processing of solid wastes; storage in geological formations; ground disposal; disposal into sea, lakes, and rivers; management of small quantities of radioactive waste; monitoring and control; and political and legal considerations. Information on atmospheric dispersion is appended. (P.C.H.)

36 MANAGEMENT OF RADIOACTIVE WASTES. p.166-80 of "Fundamental Nuclear Energy Research, 1964." Washington, D. C., Atomic Energy Commission, 1964.

Studies of rivers and subsurface hydrology at Hanford, Oak Ridge, and Idaho Falls are described. Oceanographic and meteorological studies of release of radioisotopes to the atmosphere and to oceans are in progress. Treatment of low- and high-activity wastes is described. Waste disposal in geologic formations is detailed. Evaluations are carried out of various waste management methods. (T.F.H.)

37 RECENT ADVANCES IN AEROSOL RESEARCH. A BIBLIOGRAPHICAL REVIEW. C. N. Davies. A Pergamon Press Book. New York, The Macmillan Company, 1964. 87p.

Research progress on aerosols is reviewed. Emphasis is placed on contributions to pure science with little mention of applications. Topics covered include: acoustic, ultrasonic, and showk-wave effects; adhesion of particles; chemical reactions; coagulation and diffusion; combustion; concentration, particle size, and shape; electric properties; evaporation and condensation; filtration; generators; laminar flow and impingement; nucleation and growth; optics of aerosols; photophoresis; radioactivity; sampling instruments; sedimentation and fluid resistance; thermophoresis and diffusiophoresis; and turbulent flow and deposition of aerosols.

38 IMPROVEMENTS IN OR RELATING TO THE DETECTION OF FISSION PRODUCTS. Murray Vincent Wilson (to Plessey Co. Ltd.). British Patent 937,663. Sept. 25, 1963. Filed July 27, 1959.

A method and apparatus are described for detecting fission products carried in a gaseous phase and measuring their quantity by electrostatically precipitating the daughter products of the gaseous fission product materials onto an inactive liquid surface in the presence of a gas phase and then transferring the liquid medium to an activity counting device shielded from the active gas phase. The liquid surface may be in the form of charged drops, charged jets, or a surface film covering a suitably shaped electrode. In one preferred form of the apparatus the electrode is a dished porous plate with a central bore through which the liquid is led into a shielded counting chamber. The liquid is delayed in this chamber for a suitable period, and secondary decay of the precipitated activity is measured. Proportional counters, scintillation counters, or an assembly of Geiger tubes may be used for activity measurement. (A.G.W.)

39 IMPROVEMENTS IN METHODS OF SOLIDIFYING WATERY ATOMIC WASTE. Rudolf Alberti. British Patent 938,211. Oct. 2, 1963. Priority date Apr. 2, 1960, Germany.

In the method described radioactive waste water is introduced without stirring into a finely divided hydraulic binder contained in a mold, the mixture thus formed is allowed to set hydraulically, and the block is subjected to a finishing treatment for increasing its resistance against lixiviation and atmospheric and mechanical influences. The hydraulic binder may consist of a refractory cement either alone or together with suitable aggregates in a granular condition. The waste water may be introduced into the binder by tubes suspended in the mold. The tubes have a funnel-like upper end, apertures at fixed distances along their length, and are open at the lower end. The finishing treatment may consist of baking in an oven after the block has set or introducing an impregnating material through the tube used to insert the waste water. The impregnating materials may consist of water-repellent substances in a melted condition or in the form of solutions in a volatile organic solvent or emulsions, for example, liquid self-hardening synthetic resins such as casting resins or paraffin, lacquers, synthetic fat-amino or silicone compounds.

40 DEVICE FOR THE REMOVAL OF DANGEROUS PRODUCTS. Jean Desroches and Francis Thubert (to Commissariat a l'Energie Atomique). British Patent 938,311. Oct. 2, 1963. Priority date Nov. 17, 1960, France.

The device described is intended for carrying away the radioactive waste products resulting from manipulations effected in sealed enclosures. A disposal can on a wheeled trolley can be positioned beneath a chamber communicating with the enclosure from which the products are to be removed. A detachable bag is fixed inside the disposal can, and an airtight hose connects the chamber and the disposal can. When the disposal can is filled with waste products, it is sealed with an airtight cover. Means are also provided for closing the chamber and the manipulation enclosure in such a manner that the filled vessel is removed without admission of air into the sealed enclosure. No external contamination is produced. The device contains no metallic parts in position to be attacked by corrosive fumes from the sealed enclosure. (A.G.W.)

41 IMPROVEMENTS IN OR RELATING TO THE DISPOSAL OF RADIOACTIVE WASTE. Preston Lenell Gant, D'Arcy Adriance Shock, and John Dean Sudbury (to Continental Oil Co.). British Patent 940,254. Oct. 30, 1963. Priority date Feb. 24, 1961, United States.

A method of disposing of liquid radioactive waste in subterranean locations is patented. A well bore is drilled through an impermeable rock formation, and the rock formation is fractured at a depth to provide sufficient shielding for the radioactive waste. A stream of radioactive waste treated with a solidifiable material is injected into the fracture, preceded and followed by the solidifiable material free of radioactive waste to seal the waste in the rock formation. The solidifiable material consists of clay minerals, lime, and basic metal hydroxide. Check wells are located in the vicinity of the well bore to determine the extent and direction of the fracture. (M.P.G.)

42 RECOVERY OF RHODIUM FROM FISSION PRODUCTS. (to Engelhard Industries, Inc.). British Patent 941,985. Nov. 20, 1963. Priority date Mar. 4, 1959, United States.

A process was developed for the separation of rhodium from fission products in waste streams from hot uranium reprocessing plants. The stable rhodium is isolated by first separating its parent Ru^{103}. The aqueous waste is treated in such a manner as to distill out the ruthenium tetroxide, which is reduced to ruthenium hydrated oxide or metallic ruthenium and allowed to stand for about one year. Its 40-day half life results in virtually complete conversion to Rh^{103} in this period. After the elapse of a year of storage, the residual ruthenium is removed by distillation, leaving the rhodium present as metal free of long-lived radioactive impurities. (M.C.G.)

43 RECOVERY OF RHODIUM FROM FISSION PRODUCTS. (to Engelhard Industries, Inc.). British Patent 941,986. Nov. 20, 1963. Priority date Mar. 4, 1959, United States.

A process for recovering rhodium from fission products was developed. An aqueous solution of the fission products is treated so as to precipitate rhodium as a sulfide. The rhodium sulfide is redissolved and converted to a base-soluble complex rhodium salt. The pH of the solution is adjusted to the alkaline range to remove base-insoluble constituents, after which the filtrate is converted to chloride. Ruthenium and other volatile fission products are removed by distillation, and the rhodium chloride is reduced to rhodium metal. (M.C.G.)

44 IMPROVEMENTS IN OR RELATING TO PLANT FOR TRAPPING RADIOACTIVE FISSION PRODUCTS. Samuel Brittan Hosegood (to United Kingdom Atomic Energy Authority). British Patent 955,345. Apr. 15, 1964. Filed Aug. 31, 1961.

A process plant is designed for removing fission products from a gaseous coolant coming from a nuclear reactor. The plant comprises an absorbent material, e.g. charcoal, through which the gas is forced to pass and means for transferring heat from the gas to a condensible gas, e.g. nitrogen, circulated in a closed boiling/condensing circuit which includes a condensing coil located outside

the radioactive field. Double containment is provided for the fission products, and the risk of ozone formation by irradiation of oxygen impurity in nitrogen is minimized. (D.L.C.)

45 IMPROVEMENTS IN OR RELATING TO THE MANUFACTURE OF PRODUCTS CONTAINING RADIOACTIVE MATERIALS. Guy Lefillatre, Jean Rodier, and Jean Scheidhauer (to Commissariat à l'Energie Atomique). British Patent 959,751. June 3, 1964. Priority date Dec. 6, 1961, France.

A process is given for the production of solid products containing radioactive wastes. The process comprises mixing an aqueous radioactive mud with a fluidified bitumen in the presence of a wetting agent to eliminate water, mixing the resulting bituminous mastic containing the mud solids in coated form, and pouring the mastic to form solid plastic blocks. Before mixing, the mastic may be heated to 100°C to distill off completely the residual water. Several examples of the process are given. (D.L.C.)

46 IMPROVEMENTS IN THE STORAGE TREATMENT OF RADIOACTIVE SOLUTIONS CONTAINING MOLYBDENUM. Claude Sombret (Commissariat à l'Energie Atomique). British Patent 968,763. Sept. 2, 1964. Priority date Sept. 12, 1961, France.

A process is given for treating solutions containing Mo and fission products to form a solid insoluble product. The process comprises evaporating the solution, calcining the residue to oxides, mixing the powder with the components of a phosphate glass, and heating the mixture to form the glass. Alternatively, the solution may be mixed with the components of the glass in phosphoric acid solution prior to evaporation. The process may be applied to solutions derived from the solvent extraction of irradiated Mo–U fuel elements. Several examples of the process are given. (D.L.C.)

47 METHOD FOR THE PRODUCTION OF GLASS. Bo Kurt Wilhelm Simmingskold (to Aktiebolaget Atomenergi). British Patent 975,718. Nov. 18, 1964. Priority date Mar. 17, 1962, Sweden.

By the process described, radioactive waste solutions such as those resulting from processing of used reactor fuel elements may be converted into a form of glass suitable for storing for an unlimited time. Diatomaceous earth or other solid silica material of equivalent porosity is stirred into the solution and elements from the solution are absorbed on it. The silica material is then separated, dried, and melted. The silica material may be used in powdered form or in the form of stones produced by use of a binder material such as clay, and insoluble ingredients may be mixed with the silica material before the absorption step. (A.G.W.)

48 IMPROVEMENTS IN OR RELATING TO RADIOACTIVE MATERIALS EXTRACTION PROCESSES. Geoffrey James Chetewoode Nash (to United Kingdom Atomic Energy Authority). British Patent 985,400. Mar. 10, 1965. Filed Dec. 30, 1960.

A process and apparatus are described for the extraction of volatile radioactive isotopes of the elements having atomic numbers 38, 51, 52, 53, 55, or 56 in aerosol form from a stream of carrier gas. The apparatus includes a duct connected to a source of isotopes in vapor phase, a bed of extractant material in the flow path, and means for introducing the natural form of the element upstream of the extractant material. A plurality of dispensers are provided outside the duct and each is arranged to supply a vapor of the natural isotope of the different elements in the duct to effect exchange of the natural and radioactive isotopes. A bed of active charcoal serves as the extractant.

49 WATERPROOF NUCLEAR RADIATION SCREENING SUBSTANCE. Baumgartner, Kurt; Strahle, Erwin. British Patent 1,004,248. Sept. 15, 1965. Priority date Mar. 13, 1962, Germany.

A bitumen-bound asphalt composition in the form of blocks or sheets can be used for screening against α, β, and γ radiation and neutrons. The composition includes up to 98% of at least one heavy-atom substance such as barium sulfate, lead oxide, thorium oxide, or bismuth oxide. The composition is made by dispersing the heavy-atom substance in melted, solvent-liquefied, or emulsified bitumen. It can be used for floor, ceiling, or wall coverings, to form containers for radioactive wastes, or to solidify radioactive wastes by binding with the solid substance.

50 PROCESS FOR THE DESTRUCTION OF CELLULOSE-CONTAINING MATERIAL CONTAMINATED WITH RADIOACTIVE MATERIAL. Jacobson, Cecil Felix; Larsen, Ib (to Atomenergikommissionen). British Patent 1,007,212. Oct. 13, 1965. Priority date Sept. 24, 1963, Denmark.

A process for the destruction of cellulose-containing material contaminated with radioactive material by means of concentrated sulfuric acid consists of suspending the material in cold concentrated sulfuric acid and conducting the suspension down below the surface of concentrated sulfuric acid maintained at a temperature of 300–330°C. A catalyst, such as selenium, is added to promote oxidation of the cellulose-containing material to CO_2 and H_2O. According to the process the destruction proceeds without the formation of large volumes of gas and smoke that entrain the radioactive material as suspended particles.

51 A NEW GLASS CONTAINING RADIOACTIVE WASTE OXIDES. Grover, John Roger; Jones, Alan David (to United Kingdom Atomic Energy Authority). British Patent 1,019,373. Feb. 2, 1966. Filed Aug. 10, 1962.

A glass having a melting point in the range of 850 to 1100°C consists essentially of glass forming oxides and from 20 to 50% by weight of waste oxides resulting from the treatment to remove the U and Pu from irradiated metallic U nuclear fuel having a burnup in excess of 3000 Mwday/ton. The glass forming oxides are selected from the oxides of Ca, B, Na, P, Si, Pb, Mo, Zn, Al, and Fe. The glass is formed by melting the oxides in a steel containing vessel. After the glass is formed, the vessel may be sealed and the entire body disposed of as a means of waste disposal. The sealed glass containing vessel may be used also as a radiation source providing a mixed spectrum of gamma rays substantially free of beta rays or alpha particles.

52 DEACTIVATION OF RADIOACTIVE WASTE LIQUIDS. Vladmir Vesely, Jiri Napravnik, and Jindrich Jansa. Czech Patent 109,734. Jan. 15, 1964.

The radioactive waste liquid is extracted by means of colloidal sorbent (bentonite clay 2 g/l) operating in an inactivation system of 2 to 10 stages. The waste is fed into the system from one side, while the colloidal sorbent suspension is introduced periodically, alternating from the opposite side. In each deactivation stage the radioactive waste is stirred with the sorbent coming from the next higher deactivation stage and the suspension is subjected to coagulation in an electric field. After separation of the solid and liquid phases the liquid is conducted to the higher stage, while the mud of coagulated sorbent, meanwhile thickened by sedimentation, is led into the lower deactivation stage; the process is repeated in each stage of the

system. After a 6-stage treatment, approximately 99% of ^{90}Sr, ^{90}Y, ^{137}Cs, ^{137}Ba, ^{106}Ru, and ^{106}Rh is removed. Public Health Eng. Abstr., 45: (Feb.-Mar. 1965).

53 IMPROVEMENTS RELATING TO GRIPPING MEANS FOR PERFORMING REMOTE CONTROLLED GRIPPING OF LOADS, ESPECIALLY RADIOACTIVE LOADS. (to Commissariat à l'Energie Atomique). French Patent 1,337,489. Aug. 5, 1963. Filed Aug. 3, 1962.

For the handling of a container for radioactive waste a gripping device, adapted to be lowered over such a container, was developed. Radially movable lockets of the device may cooperate with an inwardly turned edge of the container. A hollow electromagnet works together with its movable core linked to the lockets by a system of pivotally mounted levers. The core movement provoked by excitation of the electromagnet against the action of a return spring controls the simultaneous withdrawal movement lockets. (IIB)

54 METHOD AND DEVICE FOR THE SEPARATION OF GAS MIXTURES BY THE ACTION OF CENTRIFUGAL FORCES. Hans-Georg Noeller. German Patent 1,154,793. Sept. 26, 1963. Filed Jan. 4, 1956.

The method described comprises the steps of injecting the gas tangentially into a vessel with a circular cross section, of choosing the jet velocity, the gas pressure within the vessel, and its length in such a way that a circular flow is established, and of extracting the mixture enriched in heavy molecules from the circumference and the mixture depleted from the center of the vessel. The vessel is rotated by the action of the injected gas. (Gmelin Inst)

55 METHOD AND DEVICE FOR THE DISPOSAL OF WATERY RADIOACTIVE WASTE. Rudolf Alberti. German Patent 1,166,392. Mar. 26, 1964. Filed Apr. 2, 1960.

In the method described, a compact solid product is produced from the waste solution. The method comprises the steps of pouring the watery radioactive waste into a container filled with alumina cement, of allowing the saturated mixture to set, and of impregnating the compact block with a liquid synthetic plastic material which plasticizes afterwards. The device described features a container filled with cement into which a system of practically vertical tubes is inserted. The liquid waste is poured into these tubes from above, and as these tubes are provided with perforations the waste leaks into the cement. An essential feature of the invention is that a cement with polydispersed grain size distribution can be used. (Gmelin Inst.)

56 METHOD FOR THE DISPOSAL OF RADIOACTIVE WASTE MATERIALS. Guenther Friese. German Patent 1,173,998. July 16, 1964. Filed Oct. 10, 1961.

Radioactive waste can be disposed of by insertion into a waterproof container provided with protective walls of a hydraulic binding agent. The walls of the container are made of set oil shale remains. A waterproof foil is inserted into these walls and soldered or welded all over. (Gmelin Inst.)

57 METHOD FOR THE SOLIDIFICATION OF AQUEOUS ATOMIC WASTE. Alberti, Rudolf. German Patent 1,199,414. Aug. 26, 1965. Filed Mar. 18, 1961.

The method consists of introducing radioactive liquid from below into a container filled with hydraulic binding agent, of impregnating the block of radioactive liquid and binding agent with melts, solutions, or emulsions of paraffins, lacquers, or silicones, or aqueous solutions of inorganic, solidifying substances like water glass. (Gmelin Inst.)

58 METHOD FOR REMOVING RADIOACTIVE SUBSTANCES IN WATER. Kurita, Haruo (to Kurita Ind. Co., Ltd.). Japanese Patent 1965-4720. Mar. 13, 1965. Filed Aug. 24, 1962.

An efficient low-cost method using a modified natural clay mineral is proposed for removing radioactive substances in colloidal state from ordinary waste water. The modified clay is prepared by boiling natural clay mineral, i.e., montmorillonite, in a concentrated aqueous solution of sodium hydroxide, and drying the clay with or without washing. After the radioactive substances are adsorbed from waste water by the exchange reactions, the powder is separated from the water.

59 METHOD FOR REMOVING RADIOACTIVE SUBSTANCES IN WATER. Kurita, Haruo (to Kurita Industrial Co., Ltd.). Japanese Patent 1964-11560. June 9, 1965. Filed Aug. 24, 1962.

A new method for simultaneously removing radioactive ions and colloidal substances from water employs an alkali-treated clay of a double-layer structure and another alkali-treated clay of a triple-layer structure. It was found that a clay, treated with a heated alkaline substance, of a double-layer structure (e.g., Kaoline or Halloysite) adsorbed selectively divalent radioactive ions (e.g., ^{90}Sr), and that a heated-alkali treated clay of a triple-layer structure (e.g., Montmorillonite or Vermiculite) adsorbed selectively monovalent radioactive ions (e.g., ^{137}Cs).

60 METHOD OF TREATING RADIOACTIVE WASTES. Thomas H. Milliken, Jr., Jack H. Krause, and Joseph J. Donovan (to Air Products and Chemicals, Inc.). U. S. Patent 3,101,998. Aug. 27, 1963. Filed Sept. 19, 1958.

A process is described for separation of radioactive cesium, strontium, and ruthenium from nitrate waste liquor solutions to a degree whereby such solutions may be discarded or further processed in normal patterns with simplified techniques. The waste solution is first treated to form the normally adsorbable bulk ions other than Cs, Sr, and Ru into a sol having no appreciable affinity for adsorption on ion exchange materials. Additives suitable for formation of the sol include oxides or hydrous oxides of Al, Ti, Fe, Th, or Zr, metal organic compounds of the same metals such as Al isopropylate, or oxycarbonates. The treated solution is then subjected to ion exchange with a siliceous acid-form ion exchange material, and the adsorbed Cs, Sr, and Ru are recovered. (A.G.W.)

61 RADIOACTIVE WASTE DISPOSAL. Marshall L. Spector (to M. W. Kellogg Co.). U. S. Patent 3,110,557. Nov. 12, 1963. Filed Sept. 10, 1959.

A process is described for the incorporation of radioactive wastes containing metal oxides in a water-unleachable solid. In the process, the waste is reacted, in the presence of a solid metal oxide, with a base metal which has a higher exothermic heat of oxide formation per oxygen atom than that of the metal oxide. The reaction is initiated by heating the mixture, and the waste, metal oxide, and elemental metal form a solid complex mass which is insoluble and unleachable. The waste may contain Cs and/or Sr, while the metal oxide may be an oxide of Mg, Fe, or an oxide mixture found in haematite, magnetite, and ilmenite. The elemental metal may be Al, Si, Zr, or Mg.

62 SCAVENGING OF RADIOACTIVE AEROSOLS. Jan Rosinski and Donald K. Perle (to U. S. Atomic Energy Commission). U. S. Patent 3,113,007. Dec. 3, 1963.

A process of decontaminating an atmosphere from suspended radioactive particles by introducing silicon tetrafluoride whereby the particles precipitate and are removed, is described. (AEC)

63 METHOD AND MATERIALS FOR DISPOSING OF RADIOACTIVE WASTE. Henry P. Beerman (to Victor Comptometer Corp.). U. S. Patent 3,116,131. Dec. 31, 1963. Filed Aug. 1, 1960.

A method is given for isolating radioactive wastes contained in waters or vapors. The method comprises impregnating the wastes into granules or pellets of fusible, absorbent ceramic material by absorption, adsorption, or capillary condensation; drying the granules; and heating the granules to glaze or fuse their exterior surfaces into a continuous vitreous film that prevents leaching of the trapped wastes. The ceramic material comprises two major components, 80 to 90% of expanded cellular mineral absorbent (vermiculite, obsidian, pitchstone, pumice, etc.) and 10 to 20% of a fusible bonding material (bentonite, kaolin, etc.). Also, 10 to 25% of a desiccant (calcium oxide, magnesium oxide, etc.) may be used to permit capillary condensation of wastes entrained in a vapor. A method is given for making the ceramic material. (D.L.C.)

64 SUPPRESSION OF RUTHENIUM VOLATILIZATION IN EVAPORATION AND CALCINATION OF RADIOACTIVE WASTE SOLUTION. W. E. Clark and H. W. Godbee (to U. S. Atomic Energy Commission). U. S. Patent 3,120,493. Feb. 4, 1964.

A method of suppressing ruthenium volatilization in the evaporation and calcination of nitric acid fission-product solutions obtained in the reprocessing of irradiated reactor fuels is described. Phosphite or hypophosphite ions are provided in the solution to prevent volatilization of ruthenium and thus to decrease the radioactive contamination of off-gas equipment. These ions are converted to phosphate, which lowers the melting point of the resulting solids and enhances the formation of a glass-like product. Fission-product waste solutions are converted by this process to a stable, insoluble solid form suitable for long-term storage. See also NSA 17: 17618. (AEC)

65 PROCESS FOR RECOVERY OF STRONTIUM VALUES FROM FISSION PRODUCT WASTE SOLUTIONS. D. E. Hroner and R. P. Wischow (to U. S. Atomic Energy Commission). U. S. Patent 3,122,414. Feb. 25, 1964.

An improvement in the recovery of strontium-90 from fission-product waste solutions by solvent extraction with an organic solvent containing di(2-ethylhexyl)phosphoric acid is reported. Iron values, normally present as a corrosion product in fission-product solutions, are complexed by addition of tartrate ions to control iron interference with extraction of strontium. The complexed fission-product solution is first contacted with the organic phase to extract strontium and rare-earth fission products. Strontium is then separated from the rare earths by stripping and extracting the rare earths with the same organic phase at a low pH. (AEC)

66 PROCESS FOR THE PRODUCTION OF SOLID PRODUCTS CONTAINING RADIOACTIVE WASTE MATERIAL AND PRODUCTS OBTAINED BY THIS PROCESS. Guy Lefillatre, Jean Rodier, and Jean Scheidhauer (to Commissariat à l'Energie Atomique). U. S. Patent 3,142,648. July 28, 1964. Priority date Dec. 6, 1961, France.

A process is given for solidifying aqueous radioactive muds. The process comprises mixing the mud with a fluidified agent to form two phases, eliminating the water, distilling off the residual water from the bitumen, and pouring the bitumen to form solid plastic blocks. The bitumen may be fluidified by high temperature or a nonpolar solvent, and the wetting agent may comprise 1 to 20% by weight of the butumen. Two examples of the process are given in which potassium and calcium naphthenates were used as wetting agents. (D.L.C.)

67 METHOD OF DEHYDRATING AND INSOLUBILIZING AN AQUEOUS NUCLEAR REACTOR WASTE SOLUTION. W. E. Winsche and M. W. Davis, Jr. (to U. S. Atomic Energy Commission). U. S. Patent 3,152,984. Oct. 13, 1964.

A method for disposing of radioactive waste is described, which is inexpensive, convenient, and safe. The waste solution is mixed with sulfur, the mixture heated to a temperature sufficient to dehydrate it and drive off volatile components and decomposition products, then it is heated to a higher temperature sufficient to insolubilize the other waste components, and finally the residual mixture is solidified. The radioactive waste material is thereby encapsulated within a solid water-repellent matrix, which is virtually immune to degradation by the radiation of the waste material. (AEC)

68 DECONTAMINATION OF VOLATILE RADIOACTIVE EFFLUENTS. Marshall L. Spector (to Pullman Inc.). U. S. Patent 3,153,566. Oct. 20, 1964. Filed Aug. 28, 1961.

In the process for the decontamination of volatile radioactive effluents during the calcination of reactor wastes, the radioactive wastes are calcined to the corresponding oxides at 300 to 600°C, and the oxides of the volatile radioactive components such as Ru and Ce are drawn off through a cartridge located at the top of the calcining zone and packed with a solid mixture of a metallic oxide and an elemental metal. The metallic oxide is selected from the group consisting of Co, Cr, and Fe; and the elemental metal from the group consisting of Al, B, Zr, and Si. The solid mixture in the cartridge may be withdrawn and heated for fixation or passed into the fixation zone of the apparatus along with the solid oxide wastes from the calcining zone. In the fixation zone the oxides are heated to a temperature of 500 to 1200°C to chemically combine the oxides with a mixture of the same type as that used in the absorption cartridge into a solid fused mass for convenient disposal. (B.L.M.)

69 STRONTIUM RECOVERY PROCESS. George Jansen, Jr., Gerald L. Richardson, Allison M. Platt, and Lane A. Bray (to U. S. Atomic Energy Commission). U. S. Patent 3,154,500. Oct. 27, 1964.

A process was developed for recovering strontium from aqueous waste solutions by adding ethylenediaminetetraacetic acid, extracting into di-2-ethylhexylphosphoric acid, stripping the di-2-ethylhexylphosphoric acid with an aqueous citric acid solution, adsorbing on a cation exchange resin, washing the resin with a complexing agent for elu-

tion of co-adsorbed rare earths, zirconium, niobium, iron, lead, and ruthenium, and desorbing the strontium with an aqueous sodium nitrate solution. (AEC)

70 METHOD OF TREATING RADIOACTIVE WASTE. Lane A. Bray and Earl C. Martin (to U. S. Atomic Energy Commission). U. S. Patent 3,158,577. Nov. 24, 1964.

A method is described for reducing the concentrations of nitric acid and nitrous and nitrate ions in a nuclear fuel waste solution. Disaccharide is added to the solution in the ratio of one mole to from ~17 to 22 moles of nitric acid and nitrous and nitrate ions. The solution is heated from about 85°C to the boiling point for at least four minutes, and thereafter from between 50°C and the boiling point until the disaccharide is destroyed. (AEC)

71 METHOD FOR PURIFYING RADIOACTIVE WASTE LIQUID. Walter L. Hayden, Jr., and Frank J. Dzierzanowski (to Minerals and Chemicals Philipp Corp.). U. S. Patent 3,167,504. Jan. 26, 1965. Filed Oct. 4, 1961.

A method is described for purifying an aqueous waste liquid containing a low level of radiocesium in the form of a dissolved salt of a mineral acid by absorbing the cesium with a synthetically produced zeolite. The zeolite containing 4 to 14 moles of SiO_2 and from 1 to 2 moles of Al_2O_3 per mole of Na_2O was obtained by precipitating oxides of Na, Al, and Si in intimate association with finely divided particles of pre-precipitated SiO_2. The contacting of the aqueous waste liquid with the zeolite was carried out at ambient temperatures for a time ranging from about ½ to 48 hr. (B.L.M.)

72 TREATMENT OF BY-PRODUCT URANIUM MATERIAL TO RECOVER PURIFIED CONCENTRATED URANIUM VALUES. Townend, Robert V.; Wilkalis, John E. (to Allied Chemical Corp.). U. S. Patent 3,175,879. Mar. 30, 1965. Filed July 8, 1960.

A process is reported for recovering U values from U waste materials containing a mixture of U oxides and V and Mo compounds as impurities. The waste solutions are reacted with a dilute sulfuric acid below 50% concentration, preferably 15–25%, in an amount in excess of that required to react with all the U in the waste. Sufficient MnO_2 is added to convert all the U(IV) to U(VI) at a temperature in the range of 50 to 140°C to form the soluble UO_2SO_4, and the insoluble materials are separated. Ferrous sulfate in an amount equal to at least two moles of ferrous sulfate per mole of UO_2SO_4 and an excess of HF are added to the solution at a temperature in the range of 20 to 100°C to precipitate the UF_4 hydrate and to form soluble ferric sulfate. The UF_4 hydrate is separated from the reaction mixture and dried under an atmosphere of gaseous HF at a temperature 100 to 200°C. The dried UF_4 hydrate is converted to anhydrous UF_4 by heating at a temperature of 300 to 600°C in an atmosphere of gaseous HF or in an atmosphere of an inert gas at a temperature of not more than 400°C. (B.L.M.)

73 KRYPTON–XENON SEPARATION FROM A GAS MIXTURE. Dennis, Wolcott (to Air Reduction Co., Inc.). U. S. Patent 3,191,393. June 29, 1965. Filed Dec. 30, 1959.

A method and apparatus are provided for recovering Kr and Xe from gaseous mixtures such as atmospheric air or process streams containing the desired elements. The apparatus includes a primary rectification column, a concentrating column, and means for transferring an enriched liquid fraction from the primary column to the concentrating column. Condensing means connected at the top of the concentrating column receive the enriched liquid fraction from the primary column where the fraction is vaporized by heat transfer from the condensing means to form an enriched evaporated fraction, forming reflux in the concentrating column by condensation of vapor. Hydrocarbons are removed from the enriched fraction by catalyzing means, and the purified fraction is then fed into the concentrating column as feed. The reboiling means in the concentrating column is heated by the low boiling effluent from the primary column so that the Kr–Xe rich liquid accumulates in the concentrating column and is then transferred to a simple batch still for fractional distillation to separate the two. The reboiled vapor from the concentrating column is returned to the primary column. The batch still has a top condenser refrigerated by low boiling effluent from the primary column and a reboiler heated by warmed up low boiling effluent from the primary column. The fractionally distilled product is withdrawn from the top of the condenser in accord with the temperature at the top.

74 DRYING PROCESS AND APPARATUS THEREFOR FOR REMOVING SOLIDS FROM LIQUID MIXTURES. Oetjen, Georg-Wihelm; Nerge, Wilhelm; Eilenberg, Hanns; Fauser, Friedrich (to Leybold-Anlagen Holding A.G.). U. S. Patent 3,205,588. Sept. 14, 1965. Priority date Oct. 21, 1960, Germany.

A process for converting filterable radioactive waste products included in a liquid into a permanent form capable of being stored and transported consists of first filtering the liquid within an evacuable drying chamber to separate the waste products from the liquid in the form of a filter residue. The residue is then vacuum freeze dried in situ forming a product that may be easily handled. The apparatus for carrying out the process is described.

75 METHOD OF SEALING REFRACTORY VESSEL CONTAINING RADIOACTIVE WASTES. Heinemann, Heinz; Kraus, John W.; Spector, Marshall L. (to Pullman Inc.). U. S. Patent 3,213,031. Oct. 19, 1965. Filed Aug. 28, 1961.

A method is described for sealing a refractory vessel containing a solid fixation mixture of a radioactive component and non-radioactive components. The fixation mixture is fed through an entrance port of a vessel. After the entrance port is sealed, the vessel is submerged in a sealant mixture contained in a second larger container. The sealant mixture is made up of a metal oxide selected from the group consisting of oxides of Cr, Co, and Fe and an elemental metal selected from the group consisting of Al, B, Zr, and Si. The sealant mixture is heated to its ignition temperature to provide a seal around the vessel that is resistant to erosion and thermal degradation.

76 RECOVERY OF STRONTIUM VALUES FROM SULFATE-CONTAINING WASTE SOLUTIONS. Davis, W.; Yee, W. (to U. S. Atomic Energy Commission). U. S. Patent 3,218,123. Nov. 16, 1965.

A method is presented for recovering strontium values from sulfate-containing waste solutions. A nitric acid solution containing strontium values and values of other metals which precipitate as sulfates in the precipitation of strontium nitrate is contacted with a stoichiometric excess of ferric ions for formation of ferric sulfate. The nitric acid concentration is then raised to between 55 to 60 weight per cent whereby ferric sulfate is selectively precipitated. The precipitate is subsequently separated from the solution and the strontium values recovered from the aqueous solution by precipitation as strontium nitrate from greater than 18 molar nitric acid solution at a temperature of 50° to 75°C.

ECONOMICS

See also the following references:

4, 7, 18, 19, 24, 25, 31, 34, 36, 87, 92, 99, 106, 108,

116, 119, 120, 130, 170, 209, 210, 213, 222, 235, 238, 244, 247, 597, 680, 681, 682, 685, 689, 717, 774, 794, 819, 876, 878, 880, 979, 980, 983, 994, 997, 1005, 1009

77 (ORNL-TM-1441) AN ECONOMIC ANALYSIS OF HIGH-LEVEL WASTE MANAGEMENT FOR FLUIDIZED-BED VOLATILITY PROCESSING OF POWER REACTOR FUELS. Suddath, J. C.; Blomeke, J. O. (Oak Ridge National Lab., Tenn.). Apr. 1, 1966. Contract W-7405-eng-26. 20p. Dep. mn. CFSTI $1.00 cy, $0.50 mn.

Costs were estimated for management of major wastes from the fluidized-bed volatility processing of Zircaloy- and stainless-steel-clad UO_2 reactor fuels. Management alternatives consist of encapsulating the wastes without additional treatment, encapsulation following preparation of glassy solids, and encapsulation of the fission products as a glass following their separation from the inerts by a phosphoric acid leach. In all cases, the final products are shipped to a salt mine for final disposal. It was concluded that there is not a clear-cut economic incentive to make glasses or to leach the fission products, but that if these conversions are desirable for greater safety in handling and shipment, a substantial credit from cheaper management of the products can be applied against the cost of the treatment step. Also, more economic waste management cannot be cited as an advantage of volatility over aqueous processing in the fluidized-bed method for low-enrichment fuel.

78 (RL-SEP-496) WASTE MANAGEMENT PROGRAM. CHEMICAL PROCESSING DEPARTMENT. REVIEW OF COSTS AND INCENTIVES. Beard, S. J.; Smith, P. W. (General Electric Co., Richland, Wash. Hanford Atomic Products Operation). May 25, 1965. Contract AT(45-1)-1350. 12p. Dep.(mn); $1.00(cy), 1(mn) CFSTI.

Currently projected costs for the Waste Management Program are reviewed, and current estimates are given of the costs for the two alternative programs, perpetual tank storage and interim storage and solidification. The incentives for the fractionization and solidification approach are discussed. (D.L.C.)

79 (UJV-1493) LIKVIDACE RADIOAKTIVNICH ODPADU Z VYHORELEHO JADERNEHO PALIVA. EKONOMICKY PREHLED. (Disposal of Radioactive Waste from Irradiated Nuclear Fuel: Economic Survey). Kourim, V.; Korsova, M. (Ustav Jaderneho Vyzkumu, Ceskoslovenska Akademie Ved, Rez). [1965]. 76p. Dep. mn.

Data published from 1953 to 1964 on the economics of waste disposal are compiled. The work is chiefly from the U.S.A. and Great Britain and is primarily concerned with the economics of the fuel cycle in nuclear power plants and particularly with the processing of high-activity-level wastes for regeneration of burned-up nuclear fuel. Proposed megawatts of installed power for 14 countries in 1970 and 1980 are presented. The methods of disposal are organized under the headings: tank storage; chemical separation, with special attention to the Hanford program; fixation in glass; calcination, including pot calcination; and subterranean storage. Both process and economic data are given for each heading. 56 references.

GENERAL TREATMENT AND DISPOSAL METHODS

See also the following references:

1, 11, 16, 36, 77, 78, 166, 169, 174, 237, 238, 239, 250, 257

80 (ACNP-65544(Vol.I)) LA CROSSE BOILING-WATER REACTOR—SAFEGUARDS REPORT FOR OPERATING AUTHORIZATION. VOLUME I. (Allis-Chalmers Mfg. Co., Bethesda, Md. Atomic Energy Div.). July 1965. Contract AT(11-1)-850. 303p. Dep. mn; CFSTI $7.00 cy, $2.00 mn.

Descriptions of the reactor and site are presented along with descriptions of the primary cooling system and auxiliary systems including waste disposal. Design features of the containment building are also described along with the power plant.

81 (A/CONF.28/P/86) IMPROVEMENTS IN THE TREATMENT OF RADIOACTIVE RESIDUES. Gérald Wormser, Jean Rodier, and Emmanuel de Robien (France. Commissariat à l'Énergie Atomique, Paris). May 1964. 15p.

Processes and facilities for the treatment of low- and medium-activity effluents from radioactive sources in France are discussed. (R.E.U.)

82 (A/CONF.28/P/88) NEW INDUSTRIAL FACILITIES OF THE COMMISSARIAT A L'ENERGIE ATOMIQUE FOR THE PROCESSING OF LIQUID AND SOLID RADIOACTIVE WASTES. Pierre Cerre, Emile Mestre, and Jean Bourdez (France. Commissariat à l'Energie Atomique, Paris). May 1964. 16p.

Descriptions are given of the principle, operation, and performance of facilities used in France for processing low- and medium-activity liquids and solids. (D.L.C.)

83 (A/CONF.28/P/187) TECHNIQUES AND ECONOMICS OF RADIOACTIVE WASTE TREATMENT. T. G. Hughes (United Kingdom Atomic Energy Authority. Production Group, Windscale, Sellafield, England, D. W. Clelland, and B. F. Warner. May 1964. 15p.

A description is given of the systems used at Windscale for the treatment of radioactive liquid wastes arising from the two fuel reprocessing plants. High-, medium-, and low-activity wastes are considered. The effectiveness of the systems are discussed, and the improvements that have been made are described. (D.L.C.)

84 (A/CONF.28/P/282) THE NUCLEAR SAFETY RESEARCH AND DEVELOPMENT PROGRAM IN THE UNITED STATES. J. A. Lieberman (Atomic Energy Commission, Washington, D. C.), H. L. Hamester, and P. Cybulskis. 15p.

85 (A/CONF.28/P/775) A REVIEW OF RESEARCH WORK ON RADIOACTIVE WASTE TREATMENT CARRIED OUT IN THE NUCLEAR RESEARCH INSTITUTE OF THE CZECHOSLOVAK ACADEMY OF SCIENCES. L. Berak, Z. Dloughy, F. Kepak, J. Napravnik, J. Ralkova, J. Saidl, L. Schejbalova, V. Vesely, and J. Zaruba (Ceskoslovenska Akademie Ved. Ustav Jaderneho Vyzhumu, Rez). May 1964. 10p.

Developments in decontamination and disposal of wastes are described, including facilities and processes for decontaminating effluents, sorbents for radioactive particles, fixation and solidification of concentrates, and a scintillation counter for weakly active waters. (D.C.W.)

86 (A/CONF.28/P/794) MANAGEMENT OF RADIOACTIVE WASTE AT TROMBAY (A SURVEY REPORT).

R. V. Amalraj, A. R. Chinoy, J. D. Sehgal, and K. T. Thomas (India. Atomic Energy Establishment, Trombay). May 1964. 19p.

The Atomic Energy Establishment Trombay houses both industrial and research set-ups, including such diverse facilities as production units for uranium, thorium, and plutonium; fuel element fabrication shop; laboratories; and research reactors. The radioactive wastes produced vary widely over a spectrum of activity levels and chemical nature. The magnitude of the problem is presented, and the manner in which it has been solved or is proposed to be solved is dealt with. Strict limits are put on the discharge of radioactivity at Trombay, owing to the low water renewal rate of Bombay Harbor Bay, the existence of fishing and salt production industries dependent on the bay waters, the nearness of the center to the city of Bombay, and the high water table and heavy monsoon rains in the area. Various factors were considered for evolving a suitable waste management philosophy at Trombay. Details of these considerations and short descriptions of existing and proposed waste treatment plants and facilities including methods of handling radioactive sludges and management of high level wastes from the fuel reprocessing plant are given. Research work being carried out for improvement of present techniques and development of new methods of treatment and disposal is summarized.

87 (A/CONF.28/P/869) U.S. OPERATIONAL EXPERIENCE IN RADIOACTIVE WASTE MANAGEMENT (1958-1963). W. G. Belter (Atomic Energy Commission, Washington, D. C.). 16p.

Prepared for the United Nations Third International Conference on the Peaceful Uses of Atomic Energy, 1964.

Highlights of U.S. waste management experience in the past six years are summarized. The experiences include those of research laboratories, nuclear production facilities, and nuclear power plants. Among the methods of waste disposal discussed are tank storage, land burial, and sea disposal. The costs of radioactive waste management are described. (D.L.C.)

88 (AECL-2233) CONTAMINATION CONTROL AND WASTE DISPOSAL IN HIGH-LEVEL ISOTOPE OPERATIONS AT AECL. A. F. Trott (Atomic Energy of Canada Ltd., Chalk River (Ontario)). Apr. 1965. 18p. (CONF-650604-4). $0.50(AECL).

From International Symposium on Working Methods in High-Activity Hot Labs., Grenoble, France.

Ten years of operating experience handling multi-kilocurie quantities of ^{60}Co and other isotopes in a plant near the center of the city of Ottawa and remote from radioactive waste disposal facilities is reviewed. Contamination control, including ventilation design, continuous effluent air monitoring and filtration, continuous effluent water monitoring, in-cell monitoring and contamination control, liquid and dry waste processes and removal from hot cells, and the removal, packaging, and transportation of waste from the plant site to burial are described. (auth)

89 (AP-Note-442) PROTOTYPE SKID-MOUNTED RADIOACTIVE LIQUID WASTE DISPOSAL SYSTEM. P. F. Santoro and O. P. Shellar (Alco Products, Inc., Schenectady, N. Y. Nuclear Power Engineering Dept.). July 3, 1962. Contract AT(30-1)-2639. 41p. Dep.(mn); $2.00(cy), 1(mn) CFSTI.

A description is given of the prototype skid-mounted radioactive liquid waste disposal system with sufficient detail about the process components and auxiliary systems and equipment to allow understanding of operating procedures. Operating procedures are provided to enable operation of the unit by NPFO personnel while processing radioactive liquid wastes at Ft. Belvoir. The present system incorporates four techniques of reducing liquid waste activity to MPC levels (1×10^{-7} μC/cc): neutralization and/or precipitation, centrifugation, evaporation, and demineralization. The four operations are discussed as an integrated system. A detailed procedure for operation of the hydroclones is included since this equipment was installed after the training program was conducted at Schenectady on operation of the system. Recommendations for modification to the present design which have arisen during fabrication and non-radioactive testing done at Alco-Schenectady, are included. (auth)

90 WASTE FRACTIONIZATION PLANT /WAFRAX/ (Engineering Materials). (General Electric Co., Richland, Wash. Hanford Atomic Products Operation). (CAPE-1269). 104 drawings.

Engineering drawings of the WAFRAX Plant and associated equipment used for the fractionization of waste are included. The facility has a "hot side" that consists of 40 cells and canyon equipment and a "cold side" that contains the storage facilities and aqueous make-up equipment. Facilities to remove ^{90}Sr and ^{137}Cs from current and stored wastes are located in 35 of the 40 cells. Cesium recovery equipment, including ion exchangers, is located in cells 17 thru 21. Strontium or mixed Cs and Sr packaging facilities are located in cells 14 and 15. Facilities are provided in cells 9, 10, 16, and 22 thru 25 for the collection and disposal of liquid process waste, cell drainage, steam condensate from process vessels, off-gases, and organic waste. Pressure vessels and tanks are located on the "cold side" to store nitric, formic, and acetic acids; carbon dioxide; anhydrous ammonia; ammonia; carbonate; sodium hydroxide; ethylenediamine tetraacetic acid; and organic solvents. Operations and maintenance are remotely controlled. (For details, see TID-4100).

91 (CRIO-1194) RADIOACTIVE LIQUID WASTE HANDLING AND DISPOSAL AT CHALK RIVER. N. F. Murray (Atomic Energy of Canada Ltd. Chalk River Nuclear Labs., Chalk River, Ont.). Jan. 1964. 20p. (AECL-1975)

A description is given of the active liquid waste handling procedures and ultimate disposal methods used at the Chalk River Nuclear Laboratories. Experience of the past eight years, including estimates of the amounts of waste handled, is outlined. (auth)

92 (DPSPU-65-30-30) HANDLING OF LOW AND MEDIUM LEVEL LIQUID WASTE AT SAVANNAH RIVER PLANT. Girdler, R. M. (Du Pont de Nemours (E. I.) and Co., Aiken, S. C. Savannah River Plant). Sept. 1965. Contract AT(07-2)-1. 18p. (CONF-651202-1). Dep. mn; CFSTI $1.00 cy, $0.50 mn.

For Presentation at Symposium on Practices in the Treatment of Low and Intermediate Level Radioactive Waste, Vienna.

A discussion is presented of the various means used to control the radioactivity in liquid wastes which may reach the environment by way of drainage to the Savannah River: seepage basins, distillation, ion exchange on synthetic zeolite, tank storage, and aging and settling. The costs of each method are given.

93 (IEA-102) WASTE DISPOSAL PROBLEMS AT THE INSTITUTO DE ENERGIA ATOMICA. Costa Rodrigues, Jose Maria (Instituto de Energia Atomica, Sao Paulo (Brazil)). Aug. 1965. 28p. (CONF-631104-9). Dep. mn.

From the Meeting on the Utilization of Research Reactors, Sao Paulo, Brazil.

A preliminary evaluation of the radioactive liquid waste problem at the Instituto de Energia Atômica is presented. Several types of wastes produced at present, or expected in the near future, are described or tentatively evaluated. The natural drainage system of this area is also briefly described and the annual flowrates indicated. Dispersion of radioactive liquid wastes into the available near-by streams was examined and the construction of a treatment plant is suggested.

94 (HW-77163) CGC-981 PROCESS DESIGN ENGINEERING WASTE MANAGEMENT PROGRAM — PHASE II. H. L. Caudill, H. D. Haberman, and J. W. Kolb (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Mar. 31, 1963. Contract AT(45-1)-1350. 98p.

Phase II of the Waste Management Program involves a demonstration of a process system for packaging the long-lived fission products as a concentrated, low-volume waste. The process includes precipitating Cs in the feed solution, adsorbing and immobilizing the Cs and/or Sr on inorganic zeolite ion-exchange resin in containers, sealing the containers, and routing the process raffinates to the waste tank farm complex. The process engineering work includes the construction of a packaging cell, modifications of building air exhaust system, and relocation and modification of existing process vessels. (D.L.C.)

95 (HW-SA-3575) DISCUSSION OF RADIOACTIVE WASTE MANAGEMENT PRACTICES. G. E. Backman (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). June 5, 1964. Contract AT(45-1)-1350. 18p. (CONF-708-1)

From Lecture on Radioactive Waste Management, Portland, Ore., June 1964.

The main consideration in radioactive waste management is to prevent undue radiation exposure as economically as possible. The two disposal concepts of concentration-containment and dilution-dispersion are discussed for gaseous, liquid, and solid wastes, respectively. (D.L.C.)

96 HANFORD ISOTOPES PLANT. C. A. Rohrmann. Isotopes Radiation Technol., 2: 99-123(Winter 1964-65). (HW-SA-3607)

The Hanford Waste Management Program produces large quantities of crude and purified concentrates of ^{90}Sr, ^{137}Cs, ^{144}Ce, and ^{147}Pm. A design was developed and costs estimated for an isotope processing plant, which would be capable of high-volume production of encapsulated heat-source forms of ^{90}Sr, ^{137}Cs, ^{144}Ce, and ^{147}Pm. Drawings of the conceptual plant are presented. The waste management process will produce, by solvent extraction and ion exchange, two streams of interest: an aqueous stream containing a mixture of Sr and Cs at about 98% recovery and an aqueous stream containing the rare earth isotopes including cerium and promethium, with 90 and 80% recovery, respectively. Methods for processing these further are described, and flowsheets presented. Alternative products and processes are discussed. (M.C.G.)

97 (IDO-12040) RADIOACTIVE WASTE DISPOSAL DATA FOR THE NATIONAL REACTOR TESTING STATION. Joe H. Osloond (Idaho Operations Office (AEC), Idaho Falls. Health and Safety Div.). April 7, 1965. 36p. Dep.(mn); $2.00(cy), 2(mn) CFSTI.

A summary is presented of the data pertaining to the disposal of radioactive waste at the National Reactor Testing Station. The volumes of liquid, gaseous, and solid wastes are reported with the concentration and total amounts of contained nuclides given. Detailed information is listed for the years 1962 through 1964. Yearly summary totals are listed for the years since 1955. (auth)

98 (IDO-16925(p.119-29)) WASTE MANAGEMENT AT THE NATIONAL REACTOR TESTING STATION. Denzel K. Jenson (Phillips Petroleum Co. Atomic Energy Div., Idaho Falls, Idaho and Idaho Operations Office, AEC).

Waste management at the National Reactor Testing Station (NRTS) is carried out to prevent harm of any type from coming to any person from the wastes generated by activities at the NRTS. High-level liquid waste, waste calcining facilities, solid waste, gaseous waste, and low-level liquid waste are discussed. Both concentration and confinement and dilution and dispersion are used in waste management.

99 (JAERI-1048) TEST OPERATION AND COST ESTIMATION ON THE RADIOACTIVE WASTE TREATMENT PLANT AT JAERI. Sadahiro Sakata, Masukuni F. Ito, and Nobu Mitsuishi (Japan Atomic Energy Research Inst., Tokyo). June 1963. 55p. (In Japanese)

Performance data on test operations conducted through April 1960 and treatment cost estimates concerning the Radioactive Waste Treatment Plant at JAERI (cf. JAERI-1021) are given. Test operations were first conducted with nonradioactive materials and then with real radioactive wastes from JAERI. The flocculation and ion exchange unit, evaporation unit, and incineration unit were examined. The performance data showed that the facilities fulfilled their specifications and functions expected from the initial planning. Considering that the plant is the first of the kind in Japan, some peculiarities and operational troubles were also pointed out from the view point of radioactive waste treatment. Brief cost estimates on each facility have been carried out with two purposes as follows: (1) to find out the optimum natures of wastes for each treatment process and, consequently, to serve as a reference of the selection of a suitable treatment process for a specified waste; (2) to suggest practical ways for realization of more economical waste treatment. For the liquid treatment, it is shown that solid content is one of the most important indices in this system. Some reasonable combinations of various ion exchanges and evaporations were examined to clarify their optimum regions in connection with solid content. For the solid processing, incineration is more favorable than direct storage. (auth)

100 (JPRS-34927) METHODS OF COMBATING RADIOACTIVE POLLUTION. Robev, Stefan. Translated from Priroda, No. 5, 14-17(1965). 8p. (TT-66-31365). CFSTI $1.00 cy, $0.50 mn.

Disposal of increasing accumulations of radioactive materials is discussed. Methods for converting radioactive refuse to the solid state, the most convenient form for safe disposal, are evaluated. Decontamination of food, drinking water, areas, and surfaces is considered.

101 (JUL-261-DE) WASSERWIRTSCHAFT UND ABWASSERBEHANDLUNG DER KERNFORSCHUNGSANLAGE JUELICH IN DEN JAHREN 1960–1962. (Water Economy and Waste Water Treatment of the Juelich Nuclear Research Installation in the Years 1960–1962). Schulze-Rettmer, R.; Rattay, K. H. (Kernforschungsanlage, Juelich (West Germany)). Apr. 1965. 30p. Dep. mn.

The water economic relations at the research institution were reviewed. The methods used for the monitoring and treatment of domestic sewage, waste waters from laboratories and workshops, and waste waters from hot laboratories are reported. The methods used for the monitoring of drinking water, ground water, rain water, and rivers and lakes are described. The analytical monitoring of boiler water is also reported. (J.S.R.)

102 (K-1651) REMOVAL OF LOW-LEVEL RADIOACTIVE WASTES BY A SANITARY WATER TREATMENT PROCESS. Schultz, N. B. (Oak Ridge Gaseous Diffusion Plant, Tenn.). Apr. 12, 1966. Contract W-7405-eng-26. 13p. Dep. mn. CFSTI $1.00 cy, $0.50 mn.

The cleanup factors that were obtained across a potable water treatment plant using normal treatment methods are reported; the average gross beta radioactivity was reduced by 15% and the strontium beta concentrations by 6%. A summary and evaluation of six months data obtained from daily field samples of the raw water supply, as well as the finished water product, is included. Sample analysis included determinations for gross beta radioactivity and radiostrontium. The raw water radioactivity concentrations were less than 25% of the maximum permissible concentrations in water recommended by the National Committee on Radiation Protection.

103 (LA-DC-7249) OPERATIONAL PRACTICES IN THE TREATMENT OF LOW AND INTERMEDIATE LEVEL RADIOACTIVE WASTES—ARGONNE AND LOS ALAMOS LABORATORIES, U.S.A. Emelity, L. A.; Christenson, C. W. (Los Alamos Scientific Lab., Univ. of California, N. Mex.); Kline, W. (Argonne National Lab., Ill.). [1965]. Contract W-7405-eng-36. 18p. (SM/71-24; CONF-651202-3). Dep. mn; CFSTI $1.00 cy, $0.50 mn.

For Presentation at Symposium on Practices in the Treatment of Low and Intermediate Level Radioactive Waste, Vienna.

Several fundamental differences exist between waste disposal practices at Argonne and Los Alamos, but the basic "concentrate and contain" philosophy is identical. There is also complete agreement between the laboratories as to the primary responsibility for protection of operating personnel and the public in pursuing the established practices. Environmental surveys provide witness to the fact that precautionary attitudes are successful.

104 (NP-8519) EUROCHEMIC WASTE TREATMENT PROPOSAL. M. Lung, F. Marcus, H. H. Ph. Moeken, and A. Redon (European Company for the Chemical Processing of Irradiated Fuels, Mol, Belgium). Sept. 21, 1959. 10p. (ETR-69)

Flowsheets are presented for the disposal of sanitary, cold, warm, and hot waste. Methods of final disposal of hot waste are discussed. (J.R.D.)

105 (NP-13876) RAPPORT SUR L'EXERCICE, 1962. (Report on Work, 1962). (Centre d'Energie Nucleaire, Brussels). 71p.

The activities of the center are summarized. Reactor operation and research, dosimetry, reactor materials, fuel element development and processing, industrial research and development, solid state physics, nuclear physics, transuranium elements, radiobiology, safety operations, waste treatment, and mathematical services are covered. (D.C.W.)

106 (ORNL-3627(p.73-114)) WASTE TREATMENT AND DISPOSAL. (Oak Ridge National Lab., Tenn.).

The waste treatment and disposal development program was designed to develop a comprehensive waste management system for nuclear wastes, including their final disposal, and to estimate the cost of this operation. The effective, economic management of radioactive effluents is a prerequisite to the natural growth of a nuclear power industry. Two methods are being developed for the conversion of high-level radioactive wastes (HLW) to solids, the Potcal (pot calcination) and Potglass (fixation of wastes in glassy solids) processes. The calcined or glassy solids are contained in 88-in.-long 5- to 24-in.-dia. stainless steel pipes, which are designed to be closed by welding prior to shipment and ultimate disposal. Since there are no off-gases, future surveillance requirements during ultimate disposal should be negligible. Glassy solids, rather than porous calcines, are desirable because of their higher thermal conductivity and high degree of insolubility. These properties should result in an increase in economy and safety. A major remaining program objective is to determine costs of calcination and fixation in glassy solids and the interrelation of cost and safety. Low-level radioactive wastes (LLW), such as cooling water, canal water, floor-drain water, and other very low-salt-content waters, are treated to remove radionuclides prior to their discharge to the environment. General treatment methods being developed for this purpose include the scavenging-precipitation ion-exchange process and the scavenging-precipitation foam separation process. An improved scavenging-precipitation ion-exchange process was successfully tested in a 600-gph pilot plant, where ORNL process waste was decontaminated to a level less than 2.0% of the maximum permissible levels for human exposure. The improvements reduced the estimated treatment cost from 78¢ to 66¢ per 1000 gal for a 750,000 gal/day plant. The scavenging-precipitation foam separation process will be tested in the pilot plant. These processes were developed to serve as high-capacity, bulk-treatment operations to achieve the desirable economics inherent in this type of processing, compared with the costs of specialized treatment processes where batching operations, many analyses, and high labor requirements lead to high unit costs. Intermediate-level radioactive wastes (ILW), such as residues from processing LLW, evaporator concentrates, and concentrates from the second and third solvent extraction cycles from processing nuclear fuels and other high-salt-content wastes, can be processed by incorporation in asphalt or concrete prior to burial in a controlled area. Work on incorporation in asphalt is promising (low leach rate, high volume reduction, easily shipped) and is continuing. Economic and safety analyses on the HLW management systems show that the cost of conversion of wastes to solids and their storage in ultimate disposal sites, such as a salt mine, will be well within economic limits. These studies, conducted jointly with the Health Physics Division, will be extended to other waste-solidification systems in the future to provide a sound basis for national and international comparison of alternative systems and to serve as a guide for the direction of the experimental work. (auth)

107 (ORNL-tr-143) TREATMENT OF RADIOACTIVE EFFLUENTS AT THE ENRICO FERMI NUCLEAR RESEARCH CENTER OF THE MILAN POLYTECHNIC INSTITUTE. Alessandro Facchini. Translated for Oak Ridge National Lab., Tenn., from Acqua Ind., Nos. 7-8(May-Aug. 1960). 22p.

The radioactive waste treatment methods used at the E. Fermi Nuclear Center of the Milan Politecnico are presented. The liquid wastes are concentrated by evaporation in an evaporator which is described; the combustible solid wastes are incinerated and the gas is filtered. (auth)

108 (RFP-638) AQUEOUS RADIOACTIVE WASTE TREATMENT PLANT AT ROCKY FLATS. Ryan, E. S.;

Vance, J. N.; Maas, M. E. (Rocky Flats Div., Dow Chemical Co., Golden, Colo.). Oct. 25, 1965. Contract AT(29-1)-1106. 19p. (SM-71-43; CONF-651202-4). Dep. mn; CFSTI $1.00 cy, $0.50 mn.

From Symposium on Practices in the Treatment of Low and Intermediate Level Radioactive Waste, Vienna.

The operation of an aqueous radioactive waste treatment plant at Rocky Flats is described, including details of construction, flow charts, operating data, and problems encountered. Plans for additional processing of the effluent are also discussed.

109 TECHNOLOGY OF RADIOACTIVE WASTE MANAGEMENT AVOIDING ENVIRONMENTAL DISPOSAL. Report of a Panel Held in Vienna, February 12-16, 1962. (International Atomic Energy Agency, Vienna). Technical Reports Series No. 27. 1964. 151p. (STI/DOC/10/27). $3.00; Sch 63,-; 18s.stg; F.fr.12,-; DM 10.50 (IAEA).

The present radioactive waste management practice is related to total containment. Studies were made to determine how closely total containment might be approached using present technology and to assess the effects of research and development work underway. Topics covered include: collection and segregation, treatment and storage, and research and development. Both theory and operational experience are considered. (48 references). (M.C.G.)

110 (SZS-2) MATERIALIEN DER JAHRESTAGUNG DES DEUTSCHPOLNISCHEN ARBEITSKREISES ZU PROBLEMEN DER ERFASSUNG, DES TRANSPORTS, DER BEARBEITUNG UND DER ENDGULTIGEN BESEITIGUNG RADIOAKTIVER ABFÄLLE. (Materials from the Yearly Meeting of the German-Polish Working Group for Problems of Fixation, Transport, Processing, and Final Disposal of Radioactive Waste). (Staatliche Zentrale für Strahlenschutz, Berlin (East German)). June 1964. 47p. Dep.(mn)

The problems encountered in Poland, the USSR, Hungary, Bulgaria, and East Germany on the collection, transport, processing, disposal of radioactive waste and the solutions adopted are reported in the papers presented at the annual meeting. (J.S.R.)

111 (TID-19941(p.1-18)) THE RESIN-IN-PULP URANIUM EXTRACTION PROCESS. MINES DEVELOPMENT COMPANY, EDGEMONT, SOUTH DAKOTA. E. C. Tsivoglou, D. C. Kalda, and J. R. Dearwater.

A field study was carried out on liquid wastes from the extraction of U from its ore in the Edgemont refinery using the resin-in-pulp extraction process. The process is analyzed in detail. The study of the wastes was conducted with particular attention to the radium balance throughout the mill and in the wastes. The results show that, although the wastes contain ~0.6 mg radium/ton ore processed or a total of ~300 mg/day, 99.8% of the radium remained in undissolved form and was effectively retained in the sands and slimes tailings pond. The waste disposal practice and monitoring of wastes at other mills are discussed. (D.L.C.)

112 (TID-19941(p.19-36)) THE ACID LEACH-SOLVENT EXTRACTION URANIUM REFINING PROCESS. I. GUNNISON MINING COMPANY, GUNNISON, COLORADO. S. D. Shearer, C. E. Sponagle, J. D. Jones, and E. C. Tisvoglou.

The results of an in-plant survey of the Gunnison U refinery and its wastes are reported. The acid leach solvent extraction process is analyzed, and the resulting liquid waste was characterized at several sampling stations. The results show that two waste constituents, Ra^{226} and arsenic, are present in potentially hazardous quantities, but that Ra^{226} stayed mainly in the undissolved state through the entire process. The flow records of Gunnison River are analyzed for the dilution available for possible future effluent discharges. (D.L.C.)

113 (TID-19941(p.37-54)) THE ACID LEACH-SOLVENT EXTRACTION URANIUM REFINING PROCESS. II. CLIMAX URANIUM COMPANY, GRAND JUNCTION, COLORADO. J. B. Cohen, C. E. Sponagle, R. M. Shaw, J. D. Jones, and S. D. Shearer.

An in-plant survey was conducted at the U refinery at Grand Junction, Colorado. The mill process and the waste pond arrangement are described. Samples were taken from the mill process for analysis of the gross activities and Ra^{226} concentrations. Waste disposal is discussed. (D.L.C.)

114 (TID-19941(p.55-71)) THE CARBONATE LEACH URANIUM EXTRACTION PROCESS. I. HOMESTAKE-NEW MEXICO PARTNERS COMPANY, GRANTS, NEW MEXICO. J. B. Cohen, H. R. Pahren, and M. W. Lammering.

Results are reported of an in-plant survey of a U refinery using the carbonate leach extraction process for the separation of U from its ore. The process is described. The survey results are presented in the form of activities and Ra^{226} concentrations; they were derived from field data, laboratory analyses, and flow balances. Waste disposal is discussed. (D.L.C.)

115 (TID-19941(p.73-91)) THE CARBONATE LEACH URANIUM EXTRACTION PROCESS. II. HOMESTAKE-SAPIN PARTNERS, GRANTS, NEW MEXICO. H. R. Pahren, M. W. Lammering, and J. Hernandez.

Results are reported of an in-plant survey of the Homestake-Sapin mill which uses the carbonate leach extraction process. The process is described. The results are presented in the form of activities and Ra^{226} concentrations, and the discrepancies in the data are discussed. Finally, the waste disposal is considered. (D.L.C.)

116 (TID-21786) [AN ECONOMIC STUDY OF THE C.E.N. RADIOACTIVE WASTE TREATMENT PLANT.] (Division of International Affairs (AEC), Washington, D. C.). [nd]. Contract AT(30-1)-3348. 68p. Dep.(mn); $3.00(cy), 2(mn) CFSTI.

Economics of the C.E.N. waste treatment plant, which handles the radioactive wastes originating in the operation of the C.E.N. reactors and laboratories, the Eurochemic plant, and the Euratom laboratories, are given. Waste treatments, concentrated waste storage, and ultimate treated waste disposal are included. Data on activities and monthly running expenses are tabulated. (P.C.H.)

117 (TID-21861) HALLAM NUCLEAR POWER FACILITY MONTHLY OPERATING REPORT NO. 32, MARCH 1965. (Consumers Public Power District, Hallam, Nebr.). Contract AT(11-1)-513. 77p. Dep. (mn); $3.00(cy), 2(mn) CFSTI.

A summary of plant operating history is presented along with summaries of plant operation and performance during the report period. Summaries are also included on plant testing, maintenance, modifications, moderator transfers, core transfers, radiation control, and operating procedure additions. (J.R.D.)

118 (TID-22168) PATHFINDER ATOMIC POWER PLANT. Monthly Operating Report No. 1, March-December 1964. Bjeldanes, M. N. (Northern States Power Co., Minneapolis, Minn.). Contract AT(11-1)-589. 46p. Dep. mn; CFSTI $2.00 cy, $0.50 mn.

Plant operations from the initial core loading until the end of 1964 are summarized. Core loading and performance, startup tests, cooling-water chemistry, radiation-monitor calibration, waste disposal, and training activities are covered. (D.C.W.)

119 (TID-22427) WASTE TREATMENT PLANT ACCOUNTING PROCEDURE, APPLICATION. Hoffmann, E.; Van Lint, J-M.; Vande Poel, M. (Societe Belge pour l'Industrie Nucleaire, Brussels). Nov. 1965. Contract 218/US. 37p. Dep. mn; CFSTI $2.00 cy, $0.50 mn.

Economics of running the Mol treatment plants on behalf of the C.E.N. are reported. All prices are quoted in Belgian francs. Two kinds of figures are quoted. In one table actual figures of the December 1964 monthly running expenses are given. Although the other figures quoted are not considered as actual, they do give a very accurate idea of running expenses averaged over the last five years.

120 (TID-22428) WASTE TREATMENT PLANT MANAGEMENT AND ACCOUNTING PROCEDURE. Hoffmann, E.; Van Lint, J-M.; Vande Poel, M. (Societe Belge pour l'Industrie Nucleaire, Brussels). Nov. 1965. Contract 218/US. 104p. Dep. mn; CFSTI $4.00 cy, $0.75 mn.

A waste treatment plant is described together with the way in which its accounting is carried out and yields the information required for efficient running and possible improvements. The information is given to show how a treatment cost is determined and understood.

121 (UCRL-14320) WASTE MANAGEMENT PRACTICES AT LAWRENCE RADIATION LABORATORY, LIVERMORE Kvam, Donald J. (Lawrence Radiation Lab., Univ. of California, Livermore). Sept. 28, 1965. Contract W-7405-eng-48. 18p. (CONF-651202-2). Dep. mn; CFSTI $1.00 cy, $0.50 mn.

From Symposium on Practices in the Treatment of Low and Intermediate Level Radioactive Waste, Vienna.

The waste management program has the functions of reducing waste generation and disposing of radioactive wastes. Retention tank waste, carboy wastes, and solid wastes are considered. The waste plant, dry waste container, waste packaging facility and waste press are described.

122 (WASH-1052) NUCLEAR SAFETY RESEARCH AND DEVELOPMENT PROGRAM. Summary Report. (Division of Reactor Development, AEC). Nov. 1964. 93p. Dep.; $0.50(GPO).

Basic and applied research activities in reactor safety including reactor kinetics, fuel meltdown phenomena, fission product release, and reactor containment are summarized. Work on reactor kinetics is being devoted mainly to Special Power Excursion Reactor Test (SPERT). The characteristics of reactors and research in the SPERT program are discussed along with work on metal-water reactions, metal ignition and burning, fission products, reactor containment, and fast reactor safety. The programs in field testing are aimed at providing ways of demonstrating the safety and reliability of terrestrial and aerospace nuclear systems. Activities in these programs are summarized in sections on terrestrial systems, aerospace systems, and nuclear safety and recovery. Effluent control research is directed toward safe management and disposal of radioactive wastes and toward control of environmental radioactivity. Activities in these areas are summarized in sections on environmnetal studies, waste management, and gaseous and aerosol effluents. A discussion concerning the direction of nuclear analysis and evaluation projects is included. (J.R.D.)

123 MANAGEMENT OF RADIOACTIVE WASTES. Atomics, 18: 8-16(May-June 1965).

Various means used by AEC installations for the treatment and disposal of radioactive wastes are discussed. Treatments by fluidized bed calcination, fixation in glass, and long term storage or ultimate disposal are discussed separately. (P.C.H.)

124 A SMALL INSTALLATION FOR THE TREATMENT OF RADIOACTIVE SEWAGE. Schulze-Rettmer, R.; Rattay, K. H. (Kernforschungsanlage, Julich, Ger.). Atompraxis, 6: 320-2(June 1965). (In German)

For digesting the radioactive sewage effluent from a radiological hospital, a small sewage-treatment device was installed in which the biological material of the waste water was chemically oxidized. This was desirable in order to assist the decontamination process. After homogenizing the solids, the sewage was warmed to 80°C and potassium peroxodisulfate was added. The addition of acid was advantageous. The efficiency was similar to that of good biological sewage treatment plants. The amount of sludge occurring in the waste water was small. Both sludge and water may afterwards be decontaminated in a further treatment system. The oxidized components were sterilized, and after neutralization were no longer toxic. (auth)

125 THE TREATMENT OF BIOLOGICALLY LOADED RADIOACTIVE SEWAGE. R. Schulze-Rettmer and K. H. Rattay (Kernforschungsanlage, Julich, Ger.). Atompraxis, 10: 74-9(Feb. 1964). (In German)

The usual systems for the treatment of biologically loaded radioactive sewage (fecal effluents of a radiological hospital) are discussed, and disgestion of the biological material prior to decontamination is described. For this purpose ordinarily biological sewage purification is indicated. Since this treatment has some disadvantage when small amounts of sewage are involved, chemical oxidation is suggested. Peroxodisulfate as an oxidant is best suited because of its effectiveness and the effluents are not toxic. (auth)

126 ROLE OF WASTE MANAGEMENT AT EUROCHEMIC. Marcus, F. R. (Eurochemic Co., Mol, Belg.). Atompraxis, 12: 50-3(Jan. 1966).

More waste streams of different compostions are created in the multipurpose plant than in any other known processing plant in operation. A major task of Eurochemic is considered to be to demonstrate that waste management can be done in a simple, safe, and economic manner despite the unfavorable plant location and the extremely low discharge limits. The different waste groups and their handling, the principles for design of equipment, and the development work to solve the problem of decladding waste that is specific to Eurochemic are discussed.

127 THE ATOMIC ENERGY MARKET IN EUROPE. PART I. FEDERAL REPUBLIC OF GERMANY. Winnacker, K. Atomwirtschaft, 10: 442-6(Sept. 1965). (In German).

The development of nuclear energy in the Federal Republic of Germany is reviewed. The market for the reactor industry, based on the projected electrical needs up to 1980, is considered. The development of industry dealing with the fuel cycle and the utilization of radiation did not develop as rapidly as the construction of power reactors. Installations, however, for the reprocessing of reactor fuels are under construction. The disposal of radioactive waste is in

the study stage in the Federal Republic. The long range view of radioisotopes in medicine, biology, agriculture, and technology indicates that they will have a considerable economic significance. (J.S.R.)

128 THE ATOMIC ENERGY MARKET IN EUROPE. PART IX. NORWAY. Moeller, F. Atomwirtschaft, 10: 474(Sept. 1965). (In German).

The nuclear market in Norway is composed of the requirements of equipment laboratories, etc. by the Institutt for Atomenergi, that is operating three research reactors – JEEP-I, the Halden Heavy Water Boiling Reactor, and the NORA Research Reactor. A plutonium extraction plant and a waste treatment plant are also in operation. The design and proposal for a 65,000 ton tanker was developed in cooperation with Sweden. The study indicated that a balance in economy could be achieved with a reactor. The main occupation of the Norwegian industry in the field of radioisotope application is laboratories and instruments for use in handling the radioisotopes. (J.S.R.)

129 SPECIAL DEVICES OF THE KRB NUCLEAR POWER PLANT. VI. WASTE TREATMENT. Queiser, H. (Allgemeine Elektricitaes-Gesellschaft, Frankfurt am Main); Mueller, A. Atomwirtschaft, 10: 624-6(Nov. 1965). (In German).

Radioactive wastes of the KRB are treated in the processing installation and released into the Donau after dilution to at least 10^{-7} mc/cm^3. Concentrates and spent resins are stored. Gaseous wastes are taken to the off-gas installation in which the free hydrogen and oxygen constituents are catalytically burned. The usual radioactive off-gases are allowed to decay before they are released to the stack.

130 RADIOACTIVE WASTE PROBLEMS IN THE PROCESSING OF IRRADIATED FUELS. Walrave, W. K. A. Atoomenergie Haar Toepassingen, 7: 135-41(July 1965). (In Dutch).

In the processing of irradiated fuels large quantities of radioactive fission products are released. The matter of storage of the waste presents both economic and safety aspects. In the descriptions given of various types of waste, emphasis is placed on concentrated solutions and on the methods of storing these solutions in solid form. (tr-auth)

131 THE IRRADIATED MATERIALS WORKSHOP (A.M.I.) AT THE CHINON POWER PLANT. Garaud, M. J. Bull. Inform. A.T.E.N. (Ass. Tech. Energ. Nucl.), No. 55, 37-49(Sept.-Oct. 1965). (In French).

The irradiated materials workshop consists essentially of installations for control of irradiated materials and of installations for the treatment of waste and for decontamination. Each of these is described.

132 RADIOACTIVE WASTE DISPOSAL. PART II. E. F. Gloyna (Univ. of Texas, Austin), S. D. Stoddard, and C. W. Christenson. Ceram. Age, 80: No. 7, 24-31(July 1964).

Details on ion exchange processes with montmorillonite clay, pot calicination, spray calcination, fluidized-bed calcination, and rotary-ball-kiln calcination of high level radioactive wastes are discussed. (P.C.H.)

133 THE HANFORD PROGRAM FOR MANAGEMENT OF HIGH-LEVEL WASTE. Tomlinson, Roy E. (General Electric Co., Richland, Wash.). Chem. Eng. Progr., 60: Symp. Ser. No. 53, 2-19(1964).

Process wastes containing fission products have been stored as alkaline slurries in underground tanks. A program is proposed to decrease the mobility of the stored materials. Wastes that self-heat significantly will be fractionized into long-lived heat emitters and residual salt waste. The cesium and strontium heat emitters will be stored separately in high-integrity containers, which can be made available for beneficial use on request; residual waste will be combined with other low-heating waste and evaporated to salt cakes in existing underground tanks. An obsolete canyon-type building will be reactivated and revised to conduct the chemical processing operations required. Solvent extraction will probably be used to isolate cesium, strontium, and cerium from acid waste and dissolved sludges. Cesium will be removed from the stored alkaline sludges supernatant solutions by adsorption on a zeolite. (auth)

134 THE CENTER AT THE HAGUE. Duboz, M. Energie Nucl., 7: 228-45(June 1965). (In French)

The Center at The Hague which is being built by the C. E. A. in the north-west part of the Cotentin Peninsula, is to be used for the treatment of irradiated fuels coming from nuclear power stations belonging to the Électricité de France. Descriptions are given of the site, arrangement of the buildings, and processes that will be used in the irradiated fuel treatment plant for the following operations: mechanical de-canning, chemical peeling, dissolving, extractions for the elimination of fission products, separation of uranium from plutonium, and treatment of plutonium (purification, preparation of the metal using oxalate, fluoride, and extraction by calciothermal method). Special attention is given to the treatment of radioactive effluents, the design of active buildings, and to the controls: process and health. (auth)

135 NEW DEVELOPMENTS IN THE MANAGEMENT OF LOW- AND INTERMEDIATE-LEVEL RADIOACTIVE WASTES. Harold Bernard (Atomic Energy Commission, Washington, D. C.). Purdue Univ., Eng. Bull., Ext. Ser., No. 112, 389-408(Mar. 1963).

Recent developments in the treatment of low- and intermediate-level radioactive wastes are reviewed. It is pointed out that such wastes are usually associated with large volumes of water and that treatment techniques should remove hazardous radionuclides simultaneously with the more abundant stable chemicals or have a specificity for the hazardous radionuclides in spite of the presence of stable constituents. Examples of waste treatment facilities described include a chemical treatment plant at Oak Ridge National Laboratory that consists of a lime-soda ash water softening plant with clay added to scavenge the Ce from the water while the parent process removes Sr and the disposal of large volumes of radioactive effluent into the ground through a pond and well at the National Reactor Testing Station. The monitoring of both systems is discussed. Research studies on ion exchange technology, recycling effluent to the process plant, mineral exchange between fission products and natural minerals, foam separation, deep underground disposal, the use of the Higgins Contactor, evaporation and sludge storage, and freeze clarification are reviewed. (C.H.)

136 THE EUROCHEMIC REPROCESSING PLANT AT MOL. PART II. R. O. Brooks. Euro Nucl., 2: 85-7(Feb. 1965).

A description is given of the treatment of effluent, storage of high level waste, general services building, research laboratory pilot hall, and health and safety program at the Eurochemic reprocessing plant. (L.B.S.)

137 RESEARCH CENTRES IN EUROPE – NO. 10. RISOE. Euro Nucl., 2: 497-504(Oct. 1965).

Studies at the Risö research center are discussed. Reactor development, facilities for handling radioactive materials, waste treatment, metallurgical studies, chemistry research, theoretical studies, research in plasma physics, applied radiation studies, and the linear accelerator are described.

138 THE TREATMENT AND DISPOSAL OF LIQUID RADIOACTIVE WASTES FROM MEDICAL AND INDUSTRIAL USERS AND RESEARCH CENTERS. R. H. Burns (Atomic Energy Research Establishment, Harwell, Berks, Eng.). Foederation Europaeischer Gewaesserschutz, Informationsbl., No. 10, 37-47(May 1964).

Potential hazards in the treatment and disposal of radioactive wastes are assessed. Some of the radioisotopes in use in industry and medicine are listed and the liquid wastes arising from these uses are considered. The advantages and disadvantages of both concentration and containment and controlled dispersal to the environment are discussed. Legislation for control of waste disposal is reviewed. Methods used at individual sites and at central treatment and disposal centers are described. (M.C.G.)

139 RADIATION CONTROL AT THE NATIONAL INSTITUTE OF RADIOLOGICAL SCIENCES. Ishikawa, Tomokiyo (National Inst. Radiol. Sci., Chiba, Japan). Genshiryoku Kogyo, 10: 58, 68-71(Dec. 1964). (In Japanese).

140 THE DESIGN OF A RADIOACTIVE WASTE STORE. R. Oliver and F. J. Stubbs (Oxford Univ.). Health Phys., 10: 69-71(Feb. 1964).

Arrangements for the storage and disposal of radioactive waste and the design of a central Radioactive Waste Store for use by all science departments at Oxford University are described. The scheme is described for liquid and solid wastes and animal carcasses. (P.C.H.)

141 PLANT FOR TREATMENT OF RADIOACTIVE WASTES OF THE CZECHOSLOVAK INSTITUTE OF NUCLEAR RESEARCH. Jenicek, A.; Kortus, J. Czech. Heavy Ind., No. 3, 9-14(1964).

A central station of comparatively high capacity was erected for treatment of low-level effluents from various sources, chiefly from decontamination of surfaces of laboratory equipment contaminated in radiochemical work. The station was designed for processing 3.6 m^3 of effluents per hour. It was assumed that the maximum specific activity would be 10^{-5} C/l with an average activity lower by about two orders of magnitude. The radioactive solutions are mixed with precipitants and, in coagulation, the predominant part of the activity is retained in flakes. The sludge is placed in asphalt-lined metal barrels. Waste waters with higher activity or those not efficiently deactivated by a chemical method are evaporated. Radiation protection procedures are described. (M.C.G.)

142 PROBLEMS OF RADIOACTIVE WASTES BEING SOLVED IN THE INSTITUTE OF NUCLEAR RESEARCH OF THE CZECHOSLOVAK ACADEMY OF SCIENCES. Lubomir Berak, Zdenek Dlouhy, Frantisek Kepak, Jiri Napravnik, Jarmila Ralkova, Jaroslav Saidl, Ludmila Schejbalova, Vladimir Vesely, and Josef Zaruba (Inst. for Nuclear Research, Czechoslovak Academy of Sciences, Rez). Jaderna Energie, 11: 9-11(Jan. 1965). (In Czech)

Investigations of radioactive waste disposal are reviewed. Some methods developed and proved are briefly described. Several of them became a basis for pilot plant and full operation equipment design and construction. (auth)

143 ACTION GUIDES FOR RADIOACTIVE EFFLUENTS FROM PRTR. Richard F. Foster (General Electric Co., Richland, Wash.), George E. Backman, Norman G. Wittenbrock, and John M. Skarpelos. J. Sanit. Eng. Div., 89: No. SA6, 17-35(Dec. 1963).

The Plutonium Recycle Test Reactor (PRTR) and the existing guides for the control of radioactive effluents and for emergency operations are discussed. (C.H.)

144 STATUS OF RADIOACTIVE LIQUID WASTE MANAGEMENT IN THE UNITED STATES. Walter G. Belter and Harold Bernard (U. S. Atomic Energy Commission, Washington, D. C.). J. Water Pollution Control Federation, 35: 168-85(Feb. 1963).

Current radioactive waste disposal practices illustrate a high degree of safety throughout the various phases of the fuel cycle. The Atomic Energy Commission also supports a vigorous research and development program in the many and varied aspects of waste management. The research and development projects, besides attempting to solve particular problems at operating sites, also endeavor to develop performance criteria which can be utilized to determine future radioactive waste management practices.

145 NUCLEAR WASTE MANAGEMENT IN CANADA. C. A. Mawson (Atomic Energy of Canada Ltd., Chalk River, Ont.). J. Water Pollution Control Federation, 35: 1055-65 (Aug. 1963).

A review of waste management is presented, including disposal to sail and disposal in a sewer system. Disposal methods at Chalk River are emphasized. It is noted that the majority of Chalk River waste goes to the Ottawa River or decays on its way. The rise in radioactivity in this River associated with plant operation is much less than the natural radioactivity present. (J.R.D.)

146 EXPERIENCES IN THE TREATMENT OF LOW-LEVEL RADIOACTIVE EFFLUENTS IN THE KERNFORSCHUNGSZENTRUM, KARLSRUHE DURING 1961 AND 1962. H. Krause (Kernforschungszentrum, Karlsruhe, Ger.). Kerntechnik, 6: 97-104(Mar. 1964). (In German)

Decontamination facilities are described and results from the first two years of operation are presented. Decontamination factors of 5 were reached in the treatment of low level radioactive laboratory effluents by chemical precipitation. In this case the ratio of sludge to decontaminated water was 1 : 120. Slightly higher active waters were treated by ion exchange. The decontamination factor reached now an average of 30 while the volume reduction ratio was 1 : 80. The regeneration waters from the ion exchangers were concentrated together with certain laboratory effluents in raising film evaporator yielding a solution of 40% which was then fixed on concrete. The total amount of radioactive effluents treated was 5300 m^3. (auth)

147 STUDY ON TREATMENT OF RADIOACTIVE LIQUID WASTE. K. Hirose (Tokyo Univ.), M. Sago, et al. Kuki Chowa, Eisei Kogaku, 37: 423-43(June 1963). (In Japanese)

Various methods for the treatment of low level liquid waste from radioisotope handling facilities such as hospitals, laboratories, and industrial factories were investigated. A new method was devised to determine the radioactivity of river water and laboratory effluents. A

new type of holding tank was designed for the storage of low level liquid waste. Experiments were undertaken, moreover, on various ion exchange and adsorption methods for the removal of radioactive materials from the liquid waste. (auth)

148 HANDLING AND BURYING OF RADIOACTIVE WASTES IN HUNGARY. Daniel Voedroes (Fővárosi Közegészségügyi Állomás, Budapest). Nepegeszsegugy, 42: 315-18(Oct. 1961). (In Hungarian)

The proper disposition of radioactive lab wastes is discussed, and practices and procedures laid down by the Ministry of Health are outlined. It is pointed out that radioactive wastes should be separated from other ordinary waste material and deposited in special containers. The contaminated wastes should be separated as to types of isotope contamination and whether they can be incinerated or not. Contaminated wastes which have a radioactive half life of less than 30 days can be stored at the lab site, and, after proper checks and measurements, can be disposed of with ordinary refuse. Contaminated clothing should be labeled and stored in a shielded area, and after the necessary decontamination time, can be reused. Contaminated wastes having a half life of 30 days to five years should be stored separately and liquid wastes should be placed in glass or polyethylene containers, which are then placed in a metal container to avoid spills in case of breakage. Long-term wastes, those having a half life of five years or longer, should be stored separately. Wastes contaminated by ^{14}C should always be stored separately. Radioactive wastes of all types should be dated and labeled as to type of isotope causing contamination, circumstances of contamination, the isotope's chemical characteristics, total activity, and solvents present. Various methods of concentrating the wastes, such as incineration and subsequent storage of ashes, are also discussed. Boiling down, silting, and sealing certain isotopes in concrete are discussed as alternative possibilities. (BBB)

149 THE WASTE PROCESSING PLANT OF THE HOT AND ISOTOPE LABORATORY. P. Graf, G. Maeder, and E. von Ruethi (EIR, Wurenlingen, Ger.). Neue Tech., 5: 520-6(Sept. 1963). (In German)

The liquid waste from the laboratory can be completely inactive, weakly radioactive, or highly radioactive. The weakly radioactive waste is processed by chemical precipitation and ion exchange. The processing installation and its monitoring system are described. (J.S.R.)

150 THE REMOTE-CONTROLLED PURIFICATION PLANT FOR ACTIVE LIQUID WASTES. P. Graf (EIR, Wurenlingen, Ger.), W. Griner, E. von Ruethi, and H. Studer. Neue Tech., 5: 527-31(Sept. 1963). (In German)

The installation for the purification of the highly radioactive liquid wastes by filtration, ion exchange, and evaporation is described. The decontaminated water is led into the processing system for the weakly radioactive wastes. The monitoring system and the electropneumatic remote-control system of the installation are described in some detail. (J.S.R.)

151 EURATOM 1964—REVIEWS AND PROJECTION. P. Baehr. Neue Tech., 6: 243-9(Oct. 1964). (In German)

The present state of the principal research programs of Euratom is briefly summarized. The reactor program, studies on controlled thermonuclear reactions, processing of fuel elements, and treatment of radioactive waste products are discussed. The adaptation of the second five-year program is considered. (J.S.R.)

152 HUMBOLDT BAY REACTOR OPERATING EXPERIENCE. Carroll, J. C.; Schuyler, J. O. (Pacific Gas and Electric Co., San Francisco). Nucl. Safety, 6: 441-51; 467(Summer 1965).

Design and operation of the reactor are described. Information is included on equipment performance, testing program for uprating, waste disposal, water chemistry, and maintenance and refueling experience.

153 WHAT HAPPENS TO RADIOACTIVE WASTE IN SWITZERLAND AND OTHER COUNTRIES? K. Eschle. Schweiz. Tech. Z., 60: 969-73; 989-95(Nov. 1963). (In German)

After a description of the origin of the main types of radioactive waste (preparation of nuclear fuels in reactors, reprocessing of used fuel slugs, use of radioisotopes in laboratories, hospitals and industry), modern methods of handling and storing radioactive waste are reviewed. Solid, liquid, and gaseous wastes are dealt with separately, and each of these groups is broken down into at least three subgroups according to activity (low, medium or high). Finally radioactive waste in Switzerland is given special attention. (Occupational Health Safety Abstr., 2: (1964))

154 TREATMENT INSTALLATION FOR RADIOACTIVE WATERS OF THE C.E.N. AT MOL (BELGIUM). P. Dejonghe. Techn. Sci. Munic., 58: 429-34(Dec. 1963).

Treatment of radioactive waste effluents displaying low, moderate, or high radioactivity is discussed. The treatment installations considered employ double chemical treatment followed by ion exchange; this treatment permits elimination of 99% of the β emitters and even more of the α emitters. The treatment concentrates are filtered after congelation and then dispersed in a hot asphalt bath. Such a mixture contains approximately 45% hard sediment and is virtually insoluble. (Public Health Eng. Abstr., 44: No. 8, Aug. 1964.)

155 SHIPPINGPORT ATOMIC POWER STATION DISPOSITION OF RADIOACTIVE WASTE. Suchoza, B. P. (Westinghouse Electric Corp., Pittsburgh). Trans. Amer. Nucl. Soc., 8: 605-6(Nov. 1965).

156 THE TRANSPORT OF RADIOACTIVE MATERIALS. Zaadnoordijk, H. Ziekenhuiswezen, 36: 373(Oct. 1963). (In Dutch)

About 20 radioactive isotopes are now used in medical work, and the problem of disposal of radioactive samples must be considered. A therapeutic dose of ^{131}I will be excreted to the extent of 60 to 70%, and the radioactive waste must be disposed. Suggestions for the disposal of gas, liquid, and solid samples are given. Laboratories should have measurement devices to determine activity levels, so that dilutions to allowable concentrations may be made. Areas should be set aside for short-lived isotopes to decay, while those of longer life should be disposed of in protected areas. Excrement of a radioactive nature may be stored in a separate well-ventilated section, where inspection of the area is carried out. Solid materials (glassware, instruments, capsules, linen) are stored in an isolated area until the activity level is so low that they can be handled and cleaned. Extremely active material of long half life should be handled by a central organization. Cadavers and animals may be sealed in plastic bags and maintained in the deep freeze until the level is safe. Burial of solid wastes is suggested and rules for the retraction from use

of exposed instruments are mentioned. Gas samples may be treated according to rules of the I.C.R.P.; filters are suggested. (BBB)

157 METHODS OF REMOVAL OF RADIOACTIVE WASTE FROM ISOTOPE LABORATORIES IN POLAND. A. Wasilewski and Z. Ksiazak (Central Laboratory for Radiological Protection, Warsaw). Symp. Tech. Zagadnien Ochrony Promieniowaniem, Warsaw, 446-55(1962). (In Polish)

The present situation in Poland with respect to the disposal of radioactive wastes is reported from the viewpoint of radiation monitoring of isotope laboratories. The possibilities for the decontamination of some liquid and solid wastes in some areas are discussed. (tr-auth)

158 PERERABOTKA I ZAKHORONENIE RADIOAKTIVNYKH OTKHODOV LABORATORII. (Processing and Disposal of Radioactive Waste from Laboratories). E. Ya. Spitsyn. Moscow, Atomizdat, 1965. 132p.

Problems in the processing, transportation, and disposal of radioactive waste from laboratories and industries handling radioactive materials are discussed. Methods of disposal used in the USSR and abroad are analyzed. (R.V.J.)

159 RADIOAKTIVNYE OTKHODY URANOVYKH ZAVODOV. (Radioactive Waste from Uranium Plants). Bakhurov, V. G.; Lutsenko, I. K.; Shashkina, N. N. Moscow, Atomizdat, 1965. 150p.

A brief review is given of uranium ore characteristics and uranium ore technology. Storage, purification, and waste disposal techniques during uranium ore processing are discussed. Methods for selective analysis of small amounts of radioactive elements are described. (R.V.J.)

160 REMOVAL OF LOW-LEVEL RADIOISOTOPES FROM WASTE WATER BY AEROBIC METHODS OF TREATMENT. Charles Hillman Lawrence. Thesis, Gainesville, Fla., Univ. of Florida, 1963. 175p.

The removal of low-level radionuclides from waste water by a complete sewage treatment plant operating under field conditions was evaluated. A pilot-size treatment plant employing primary sedimentation, trickling filtration, and lagooning was utilized. These processes were subjected to a series of eight experiments involving phosphorus-32, iodine-131, potassium-42, cerium-141 and -144, iron-59, cobalt-58, strontium-89, and mixed fission products. The results indicated that primary sedimentation was relatively ineffective in removing colloidal and dissolved radionuclides from sewage. The effectiveness of the trickling filtration process was limited to the ability of the filter to remove the radioactive components from the main waste stream and to concentrate them in the solids leaving the filter in a settleable form. Consequently, the treatment efficiency of the over-all process was controlled by the removal accomplished by secondary sedimentation. Uptake studies indicated that the removal of low-level radionuclides within the filter resulted from the concentration of these materials by the zoogleal slimes present throughout the unit. Lagoon studies indicated that the long detention time and intimate algal-bacterial contact provided by this process were effective in diluting, delaying and, in most cases, substantially removing "slugs" of low-level radioactive materials. The basin had an effective detention time of two days and achieved the following removals: approximately 90 per cent for cerium-141 and -144 and iron-59; 69 per cent for strontium-89; 50 to 60 per cent for phosphorus-32, cobalt-58, and mixed fission products; and less than 1 per cent for iodine-131. The mechanisms indicated to be effective in the removal of radioactive materials by lagooning were precipitation by chemical, physical, and biological means and ion exchange reactions involving the suspended radionuclides and the stable materials retained in the basin. Trivalent materials in their colloidal form and divalent elements biologically significant in trace amounts were removed most effectively. (A.G.W.)

161 REVIEW OF RESEARCH AND DEVELOPMENT WORK AT EUROCHEMIC. R. Rometsch (Eurochemic Co., Mol, Belg.). p.207-18 of "Aqueous Reprocessing Chemistry for Irradiated Fuels." Paris, European Nuclear Energy Agency, 1963.

In the first few years at Eurochemic emphasis was placed on development work connected with the plant project, such as adaptation of flowsheets, determination of missing data, testing of assumptions, and gaining technological experience. The results of flowsheet studies for the multi-purpose dissolver, the various extraction cycles, and waste handling are briefly reviewed. Among the possibilities investigated in this category are: a zirconium and Zircaloy dissolution process, by means of buffered solutions of hydrofluoric acid; the development of a plutonium separation and purification process using tertiary amines, and precipitation from organic solutions; and the use of quaternary ammonium salts for plutonium separation. Screening investigations have recently been started for processing fuel coated with graphite, carbon, or silicon carbide. (P.C.H.)

162 SOME CONSIDERATIONS IN THE DESIGN AND ENGINEERING OF ACTIVE EFFLUENT TREATMENT PLANTS. Spalding, E. G.; Ayre, D. J. Paper No. [9] of Anglo-Spanish Nuclear Power Symposium. Vol. 2. 30p. [Risley, Eng., United Kingdom Atomic Energy Authority, 1965].

The nature of the effluent treatment problem is considered in reference to discharge limits and probable arisings of solid, liquids, and gaseous wastes from various types of nuclear facilities. Emphasis is given to research establishment and power reactor wastes. Treatment methods are discussed, which include segregation, storage, and dilution; filtration; floc treatment; ion exchange; and evaporation. A brief outline is given on the application of treatment methods; emphasis is given to aqueous wastes, but mention is made of nonaqueous arisings. The design and construction of effluent treatment plants are also considered. Discharge limits are tabulated for some nuclear facilities.

163 THE WASTE DISPOSAL SYSTEM FOR THE MH-1A NUCLEAR POWER PLANT FOR THE SS WALTER F. PERRY. T. S. Weissman (Martin-Marietta Corp. Aerospace Div., Baltimore). p.313-29 of "Progetti ed Esperienze nel Settore della Propulsione Navale Nucleare." Rome, CNEN, 1964.

The processing and disposal of liquid and gaseous wastes from the 10,000-kwe pressurized-water power plant designed for the SS Walter F. Perry are described. The liquid wastes are handled either by dilution with the sea water condenser flow or holdup followed by evaporation and dilution. Gaseous wastes are diluted with the ventilation air for the containment vessel. Silver nitrate is used to remove the iodine from the gases collected. Solid wastes are stored in shielded shipping containers. (J.S.R.)

164 WATER POLLUTION POTENTIAL OF AIR POLLUATION CONTROL DEVICES. Gartrell, F. E. (Tennessee Valley Authority, Chattanooga). pp 387-98 of Air Pollution. Volume 2. New York, Academic Press, 1962.

Interrelations between air and stream pollution are examined, and air pollution control facilities with stream pollution control features are considered along with a review of water quality standards. Many waste materials considered as air pollutants of major concern are likewise objectionable for many water uses, when present in even fairly low concentrations. This is particularly true of wastes involving fluorides (from uranium ore processing) and radioactive materials. Examples are cited of fluoride and radioactive wastes from atomic energy installations, and information as to how stream pollution control features have been incorporated in air pollution control facilities is revealed. One example referred to is the Feed Materials Production Center located at Fernald, Ohio, which produces uranium and thorium metal. This plant is located on the Miami River, which provides a means for disposal of liquid wastes. Of the problems of waste disposal at this plant, the biggest is with fluorides. The plant processes result in excess hydrofluoric acid used in the process, hydrofluoric acid generated by the process, and metal fluoride by-products. For air pollution control, off-gases containing HF are scrubbed before discharge to the atmosphere. Insoluble fluorides are removed from scrubber systems by filtration and stored in the ground in a pit on the site prepared specially to prevent ground water contamination. Neutralized soluble fluorides from scrubber systems are discharged with other treated wastes to the river. All liquid wastes are discharged to a central sump from which they are discharged at a controlled rate dependent on concentration of contaminants and river flow. By careful regulation of waste discharge to stream flow, the amount of fluoride discharged is kept well below the permissible level of 1.2 ppM fluoride. Waste disposal practices at Oak Ridge National Laboratory are also described as illustrative of an integrated radioactive waste disposal program involving air pollution and water pollution control. Wastes are disposed of in four ways: by discharge into the atmosphere through tall stacks, by storage and decay, by fixation in the soil of the waste disposal area, and by discharge to the Clinch River. A minimum amount of radioactive material is discharged to the Clinch River and into the atmosphere. Other Atomic Energy Commission Laboratories (Argonne and Los Alamos) are also considered as examples where air-cleaning devices contribute materials to the liquid waste disposal systems.

ORIGIN AND NATURE OF WASTES

See also the following references:

19, 24, 31, 34, 35, 81, 86, 87, 93, 126, 153, 238, 250, 252, 255, 256, 262, 305, 354, 425, 501, 842

165 (AEET/HP/RWD/4) NATURE OF WASTES. Working Party No. 2. A. R. Chinoy and A. K. Ganguly (India. Atomic Energy Establishment, Trombay). Feb. 5, 1959. 18p.

Various sources of wastes in the nuclear industry are analyzed for radioactivity and other characteristics. Hazards from radiation and chemicals in the wastes are discussed. (31 references). (C.H.)

166 (CEA-tr-A-1490) LE PROBLEME DES DECHETS RADIOACTIFS. (The Problem of Radioactive Wastes). Spitsyn, V. I. Translated into French from Abhandl. Deut. Akad. Wiss. Berlin, Kl. Chem., Geol. Biol., No. 2, 43-61(1960). 58p. Dep. mn; JCL $5.60 fs, $1.94 mf.

Sources of radioactive wastes from industrial and non-industrial installations are discussed and their characteristics are tabulated. The processing of radioactive wastes before burial or storage is summarized, and criticisms of disposal by burial are reviewed. The processing of radioactive wastes at Windscale, Marcoule, and Mol is discussed, and the advantages and disadvantages of methods used by each are emphasized.

167 (NYO-3086-1, Pt.VIII) THE RADIOACTIVITY OF FOSSIL FUELS. Petrow, Henry G.; Cover, Arthur; Eisenbud, Merril (New York Univ., N. Y. Inst. of Environmental Medicine). 2p.

Samples of fly ash and fuel oil residues were obtained from a number of localities: Alabama, Wyoming, Utah, Japan, Eastern Europe, and North Africa. The results to date tend to support the conclusion that fossil fuel plants are a source of low-level atmospheric radioactive contamination.

168 (RT/PROT(65)-12) ASPETTI TECNICO-IGIENISTICI DELLO SMALTIMENTO DEI RIFIUTI RADIOATTIVI IN ITALIA. (Technological-Hygienic Aspects of Radioactive Waste Disposal in Italy). Branca, G. (Comitato Nazionale per l'Energia Nucleare, Rome (Italy)). Apr. 1965. 54p. (CONF-631005-1). Dep. mn.

From 23rd Congress of the Italian National Assn. for Hygiene and Public Health, Catania, Italy.

The most important technical procedures giving rise to liquid, solid, and gaseous radioactive wastes are described. Particular reference is made to typical situations which exist or will be set up in Italy in the near future. Measures taken in the different Italian nuclear research centers to solve the problem of contamined waste treatment and disposal are examined. Finally, a description of the waste discharge philosophy applied in some particular cases is given. This philosophy is established taking into account the environmental characteristics and the possible influence of the radioisotopes dispersed in the ecosystem.

169 RADIOACTIVITY IN OUR VITAL ENVIRONMENT. O. Jaag (Eidgenossisches Technische Hochschule, Zurich). Foederation Europaeischer Gewaesserschutz, Informationsbl., No. 10, 16-32(May 1964). (In English, German, and French)

The influence of radioactivity in our environment, its source, its manifestations, and its intensity are discussed. Purification methods for radioactively contaminated water are described. Methods of analyzing water, grass, hay, and soils for radioisotopes are outlined for use in studying the transport of the radioisotopes by water. Radioactivity control activities in Europe, especially Switzerland, are reviewed. Waste disposal is also considered. (M.C.G.)

170 PROBLEMS RELATIVE TO THE TREATMENT OF RADIOACTIVE WASTES. M. P. Dejonghe (Societe Belge de Chimie Nucleaire, Mol, Belg.). Bull. Inform. Assoc. Belge Develop. Pacifique Energie At., 9: No. 49, 5-19(Apr. 1964). (In French)

A brief review is given of the problems in the treatment of radioactive wastes. The topics discussed are sources of the wastes, processing techniques, migration of radioisotopes in the soil, special problem of highly active wastes, and economic aspects. (J.S.R.)

171 RADIOACTIVE WASTE DISPOSAL. PART I. E. F. Gloyna (Univ. of Texas, Austin), S. D. Stoddard, and

C. W. Christenson. Ceram. Age, 80: No. 6, 63-72(June 1964).

The rapidly growing stockpiles of high-level radioactive wastes derived from processing irradiated nuclear fuels are discussed. Descriptions of the processing techniques, problems involved in ensuring safe storage of the wastes, and some methods of converting liquid wastes to solids are included. In all, 65 references are included in Part I and 2 other parts to follow. (P.C.H.)

172 THE DISPOSAL OF RADIOACTIVE EFFLUENTS. P. Bovard and M. E. Gahinet. Eau, L., 50: 301-6 (May 1963).

Various sources of radioactive contamination of water, including material sources, contributions from nuclear explosions, and industrial waste effluents are discussed. The problems associated with treatment of radioactive wastes are considered, particularly the problems encountered when field disposal involves possible contamination of rivers. (Public Health Eng. Abstr., 43: No. 11, 1963)

173 RADIOACTIVE WASTE: GENERAL CHARACTERISTICS. Iranzo, Emilio (Junta de Energia Nuclear, Madrid). Energia Nucl. (Madrid), 9: 365-76(Sept.-Oct. 1965). (In Spanish).

The origin of radioactive wastes is summarized briefly. The nature of the waste from reactors, fuel processing plants, uranium ores and fabricating plants, radioisotope production, and nuclear accidents is discussed.

174 ON THE TREATMENT OF LIQUID WASTE PRODUCED BY ORGANIC COOLED AND MODERATED REACTORS. G. Branca and S. Gagliardi (CNEN, Rome). Ing. Sanit., No. 6, 2-8(1962). (In Italian)

The contaminated liquid wastes produced by an organic moderated and cooled reactor and its ancillary devices can be divided into two types: organic and aqueous. More specifically, the wastes of the second type consist of the condensation waters of the ejectors of the degassing and purification plants and the waters eliminated by the decontamination systems. The main features of such wastes are examined, with reference to a possible negative influence on the disposal of the radioactive substances. The results of a series of tests performed to ascertain which treatment is more adequate for these wastes are presented.

175 RADIOACTIVE POLLUTION. BIOLOGICAL EVOLUTION. Michon, G. Maroc Med., 43: 167-76(1964).

The origin of radioactive pollution from human industry, methods of emitting radioactive dusts and gases, evacuation of residues in water currents, storage of radioactive elements, and the physicochemical behavior of radioisotopes are discussed. The behavior of radioelements in atmosphere, water, and soil is reviewed. A study of the biological cycle of radioelements, including processes of dilution and concentration in living organisms, and relationships with ambient factors and with factors of animal and vegetal vitality is described. (TCO)

176 WASTE WATER AND NUCLEAR ENERGY. Michon, G. (CEN, Fontenay-aux-Roses, France). Rev. Pathol. Gen. Comp., 64: 583-8(1964).

The various origins of waste water produced in the generation of nuclear energy are studied. Thus far radioactive wastes have not been kept minimal in industry. Physical, chemical, and biological tests permit an assessment of the general aspects of such pollution. Biological studies permit better definition of the risks and improvement of protection of public health, although this is already perfectly satisfactory.

177 RADIOACTIVE PARTICLE IN SEDIMENT FROM THE COLUMBIA RIVER. Norman Cutshall and Charles Osterberg (Oregon State Univ., Corvallis). Science, 144: 536-7(May 1, 1964).

A hot particle containing only fission product radioactivity and comprised of a number of characteristics that make it unusual and its origin uncertain was found in the Columbia River. (auth)

PROGRESS REPORTS

178 (BNL-799(p.26-54)) CHEMISTRY AND CHEMICAL ENGINEERING DIVISION. O. E. Dwyer and R. H. Wiswall (Brookhaven National Lab., Upton, N. Y.).

179 (BNL-799(p.55-63)) RADIATION DIVISION. B. Manowitz (Brookhaven National Lab., Upton, N. Y.).

180 (BNL-841(p.18-42)) CHEMISTRY AND CHEMICAL ENGINEERING DIVISION. O. E. Dwyer and R. H. Wiswall (Brookhaven National Lab., Upton, N. Y.).

181 (BNL-900(p.29-72)) CHEMISTRY AND CHEMICAL ENGINEERING DIVISION. Dwyer, O. E.; Wiswall, R. H. (Brookhaven National Lab., Upton, N. Y.).

182 (BNWL-36) HANFORD RADIOLOGICAL SCIENCES RESEARCH AND DEVELOPMENT ANNUAL REPORT FOR 1964. Pearce, D. W.; Green, J. K. (eds.) (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). Jan. 1965. Contract AT(45-1)-1830. 571p. Dep. mn; CFSTI $8.71 cy, $2.50 mn.

Developments in atmospheric physics, radiological physics, radiological chemistry, chemical effluents technology, and instrumentation are reported. Approximately 110 papers or abstracts are included.

183 (BNWL-36, pp 3.1-122) RADIOLOGICAL CHEMISTRY. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.).

184 (CENRD/HP/62/2) S1C PROTOTYPE REACTOR FACILITY. Environmental Monitoring Report, January-June 1962. Volume III, Number 2. (Combustion Engineering, Inc. Naval Reactors Div., Windsor, Conn.). July 1962. Contract AT(30-3)-519. 9p.

Approximately 45.96 mC of beta radioactivity was released during the period January to June 1962 in liquid waste effluent and dilution water. The average concentration released was 8.7×10^{-6} μC/cc. All liquid waste was released at less than $^1/_{10}$ the permissible concentrations for continuous exposure as specified by the National Committee on Radiation Protection and Measurements for the radionuclides present. The Farmington River was sampled monthly at various locations upstream, and downstream of the Industrial Waste Outlet and at the outlet. The average of the river results indicate values of 3.1×10^{-8}, 1.3×10^{-8}, and 2.3×10^{-8} μC/cc beta at the up-

stream, outlet and downstream locations, respectively. Radioactive fallout is collected on a weekly basis at six locations around the site (on-site) and analyzed for gross beta radioactivity. Radioactive fallout is also collected at eleven off-site locations in neighboring towns and analyzed for gross beta radioactivity. Comparison of the weekly averages of all on-site sampling stations indicated 19.71 mC/mi^2/sample and 20.64 mC/mi^2/sample for all off-site sampling stations. Airborne waste effluent monitored at the point of release for beta radioactivity indicated an average concentration of 4.0×10^{-11} $\mu C/cc$, for the period January to June 1962. This compares with the recommendations of the National Committee on Radiation Protection Standards, 1×10^{-10} $\mu C/cc$, for the radionuclides involved. Air samples taken downwind of the point of release during the period indicates an average of 7.1×10^{-12} $\mu C/cc$ beta. Measurements of radioactivity in the Farmington River, the Industrial waste effluent from the plant, air and fallout, in the S1C Prototype vicinity, during the period January to June 1962 indicated no significant contribution of radioactivity from the S1C Prototype to the environment. Furthermore it can be stated that radioactivity discharged to the environment under controlled conditions from the S1C Prototype, during waste disposal operations, were in full compliance with the recommendations of the National Committee on Radiation Protection and the radiation protection standards established by the Federal Radiation Council. (auth)

185 (EURAEC-202) RESEARCH PROGRAM ON WATER TREATMENT AND RADIOACTIVE RESIDUES. Quarterly Report No. 5, April–June 1961. S. A. Belchim (Societe Belge de Chimie Nucleaire, Brussels). Aug. 1961. Work Performed under United States–Euratom Joint Research and Development Program. 21p. (R-2021). (A translation)

A tracer technique using iodine-131 for measuring the water movements in the underground water-level is being evaluated. Copper rods, sunk into the ground at a known depth and distance from the point of injection, are used as absorbant. The studies of bitumen coating for radioactive residues are investigated, particularly with the aim of finding other types of bitumen and to determine the activity losses through volatilization. The latter are in the region of 10^{-3}% of the insolubilized radioactivity. The construction of the incinerator continues. The study of gas scrubbing by injecting an absorbant is continuing. Decontamination factors of 100 to 200 are achieved. (N.W.R.)

186 (EURAEC-300) RESEARCH ON THE PROCESSING OF RADIOACTIVE WATER AND RESIDUES. Quarterly Report No. 8, January 15–March 31, 1962. (Société Belge de Chimie Nucleaire, Brussels). Work Performed under United States–Euratom Joint Research and Development Program. Contract 048-61-4-RDB. 32p. (R-2113). (A translation)

The elutriation tests of sediments, ashes, and evaporator concentrates with bitumen are evaluated. The climatic data of the Mol site are analyzed in respect of their hydrological aspect. This study has shown that the speed of infiltration of water in the upper layers of the soil may vary from 2.60 to 0.40 m per day, depending upon the degree of humidification of the section. The studies of the unsaturated zone have demonstrated that the infiltration of water in a section gives rise to a mixture between the water already in the soil and the water added. The infiltration of the water takes place by preferential routes and the isotope retention is more marked in the unsaturated medium. The apparatus for the injection of radioisotopes is completed. The groundwater sampling apparatus and the hydraulic model are being prepared. The physico-chemical studies of the ionic occupation of clays and the fixation parameters of sand are evaluated. The fixation parameters of the radioisotopes strontium-89 and cesium-134 are determined. Several incineration tests are being made for the purpose of adapting the incinerator to radioactive waste processing. (N.W.R.)

187 (HW-78388(p.4.1-15)) CHEMICAL RESEARCH AND DEVELOPMENT. (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.).

188 (HW-79140) RESEARCH AND DEVELOPMENT ACTIVITIES FIXATION OF RADIOACTIVE RESIDUES. Quarterly Progress Report, July–September 1963. E. R. Irish, ed. (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Oct. 1963. Contract AT(45-1)-1350. 44p.

The radiant-heat spray calciner in the High-Level Radiochemistry Facility was used for additional experimental studies of full-level Purex 1WW calcination. The unit was operated with ammonia-neutralized Purex waste plus sugar, acidic Purex waste with phosphate added, and with sugar added. Behavior of ruthenium and properties of the melt were observed. A tracer I^{131} run was made in the 8-inch diameter cold calciner to evaluate iodine behavior during calcination. The large Cold Semiworks spray calciner was operated for further evaluation of the calciner capacity and heat-transfer characteristics. Development of glass processes was initiated with installation and operation of a continuous melter for calcine powder. An elaboration on the sugar denitration process for Purex 1WW is provided. Experimental runs were made with a model annular circulator proposed for in-tank solidification of intermediate-level plant wastes to determine heat-transfer characteristics. Details of the engineering work on dehydration of fission product-loaded zeolites were documented. Equilibrium constants and free energies were determined for two exchange reactions on six exchangers. Zeolites with low silica-to-alumina ratios exhibited the highest strontium selectivity. Breakthrough curves for strontium on Linde 4A zeolite were predicted on the basis of plate theory and compared to experimentally determined curves. The calculated and experimental curves show close agreement. Work continued on defining flowsheet conditions for removal of cesium from Purex and Redox wastes. Experimental work is underway to evaluate electrodialysis as a processing technique for treatment of low- and intermediate-level wastes. Pilot plant solvent extraction studies were continued on the development of CSREX and D2EHPA processes for removal of long-lived heat-generating fission products from Purex wastes. Pulse column design details are presented. (auth)

189 (HW-79280(p.4.1-28)) CHEMICAL RESEARCH AND DEVELOPMENT. (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.).

190 (HW-80284(p.4.1-23)) CHEMICAL RESEARCH AND DEVELOPMENT. (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.).

Salt Cycle Process Engineering Development. Test runs of the High Level Radiochemical Facility for processing of irradiated UO_2-PuO_2 fuel from the PRTR were made. Radioactive Residue Process Development. A sample of Redox waste and a fresh quantity of raw Purex waste (not formaldehyde treated) were analyzed. The Redox waste was spray calcined with glass forming additive. The Purex waste was calcined after most of the acid was destroyed by digestion with sugar. A spray calcination run was made with phosphite added to the Purex waste. A further run was made with the Purex waste with cold fission products to simulate high burnup. Large diameter draft tubes, with and without fins, were studied to determine their effects on recirculation rates and feed capacities for the Cold Semiworks Radiant Heat Spray Calciner. Filter blow-back overpressures were measured for various nozzle sizes, venturi shapes, filter permeabilities, and blow-back air flow rate for off-gas filtration for spray calciners. The effects of composition on melting point, phase separation, and solubility were studied for three types of glass systems: Purex-phosphate with phosphate additions; higher phosphate types; and a system containing lithium, calcium, and aluminum additions to reduce the volatilization of sulfates. Work on the process and equipment for the hot cell glass experiment for the conversion of actual high-level wastes to glassy solids continuously was carried out. The capacity of the continuous calciner melter was studied using nonradioactive Redox- and Purex-type wastes; corrosion of the Inconel melter pot was found. The development of the Fuels Recycle Pilot Plant is discussed. Thermodynamic data are given for ion-exchange materials; zeolite, phillipsite, etc. The performance of a three-bed demineralizer for steam-stripped alkaline (Purex tank farm) condensate is demonstrated. (R.E.U.)

191 (HW-80526) RESEARCH AND DEVELOPMENT ACTIVITIES FIXATION OF RADIOACTIVE RESIDUES. Quarterly Progress Report, October–December 1963. E. R. Irish, ed. (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Jan. 1964. Contract AT(45-1)-1350. 38p.

The radiant-heat spray calciner in the High-Level Radiochemistry Facility was used for additional experimental studies of full-level waste calcination. The unit was operated with Redox waste, Purex waste plus sugar, Purex waste with phosphite added, and Purex waste to which cold fission products were added. Large (16 in.) diameter draft tubes, with and without fins, were the subject of a study to determine their effects on recirculation rates and feed capacities. The spray calciner off-gas filter overpressures were measured for various nozzle sizes, venturi shapes, filter permeabilities, and blowback air flow rate. The effect of composition on glass melting points, phase separation, and solubility was studied. Work on process and equipment for the hot cell glass experiment was carried on in cooperation with Brookhaven National Laboratory personnel. Work on the calciner melter was directed toward determining the melter capacity with two types of nonradioactive waste calcines. The use of immersion-type electric heaters in addition to heated air is being examined as an alternate low cost approach to heated air only for in-tank solidification. Thermodynamic data were obtained for three exchange reactions on eight inorganic ion exchangers. Pilot plant studies on intermediate-level waste treatment were continued. However, solids appeared recently in the condensate waste and caused processing difficulties. The solids contained a high concentration of bacteria, which may be due to raw cooling water being directed temporarily to the condensate receiver. Efforts are being made to correct this condition. Work was continued on defining flow sheet conditions for removal of cesium from Purex and Redox stored alkaline wastes. (auth)

192 (HW-83298(p.4.1-25)) CHEMICAL RESEARCH AND DEVELOPMENT. (General Electric Co., Richland, Wash. Hanford Atomic Products Operation).

193 (HW-83320) RESEARCH AND DEVELOPMENT ACTIVITIES, FIXATION OF RADIOACTIVE RESIDUES QUARTERLY PROGRESS REPORT, APRIL–JUNE 1964. A. M. Platt, ed. (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). July 1964. Contract AT(45-1)-1350. 30p.

Membrane filters were shown to be efficient Ru samplers during a spray calcination run with full-level Purex waste. Visual observation of a four-month-old radioactive phosphate glass revealed several small cracks. Results of glass formulation studies indicate that the phosphate is almost universally lost during drying and melting. Assembling and testing of equipment for a hot cell glass experiment is nearly completed. The cold semiworks 18-in. calciner and continuous melter were successfully operated as a composite unit. Laboratory studies of the CSREX process under gamma irradiation show a two-fold decrease in Cs distribution ratio. Continued computer studies to ascertain the most important factors in removal of heat from underground storage tanks have further delineated the effects of cake and soil thermal conductivity and cake thickness. Purex acid and alkaline condensates were treated by an electrodeionization technique to evaluate Ru removal. Ru decontamination factors of 50 and 15 were obtained for the acid and alkaline condensates respectively. Linde AW-500 zeolite was shown to yield minimum fines when mixed vigorously with water. At least three types of cation sites were observed to be present on Type X zeolites. (auth)

194 (HW-84467) RESEARCH AND DEVELOPMENT ACTIVITIES. FIXATION OF RADIOACTIVE RESIDUES. Quarterly Progress Report, July–September 1964. A. M. Platt, ed. (General Electric Co., Richland Wash.). Hanford Atomic Products Operation. Oct. 1964. Contract AT(45-1)-1350. 39p. Dep.; $2.00(cy), 1(mn) CFSTI.

Two runs with full-level radioactive feed were completed in the cooperative BNL-HL Hot Cell Glass Experiment. Both runs were operationally successful and radiologically uneventful. An excellent glass was obtained from the second run. In associated studies on glass formations it was found that the solubility of glass was markedly affected by the thermal history of a sample. Various waste-additive compositions were evaluated as to suitability for use in waste fixation processes involving spray calciners coupled to continuous melters. The systems Purex waste calcine—B_2O_3-CaO and Purex waste—SiO_2Na_2O were studied. Glasses were obtained with both systems; those for the latter system all had fusion temperatures above 900°C. Viscosity, density, and nitrate concentration in the bottoms of a phosphate glass process evaporator operating at a steady state were determined. All three dependent variables were measured as a function of bottoms temperature and phosphate-to-metal ion equivalent ratio. Studies were initiated with a new 14-in.-dia spray calciner in which both calciner-melter coupled and uncoupled studies were conducted with emphasis on Purex type wastes. Since the glass or melt fluxing agents may be readily incorporated into spray calciner liquid feed or added to the continuous melter as a

separate dry chemical, a separate direct feed to the melter was scouted. Results of studies indicate that either method is feasible. Steam stripping of Purex tank farm condensate, following hypochlorite treatment to reduce bacterial activity, produced a relatively clear feed which appears to be satisfactory for thin precoat ion exchange bed treatment. Electrodeionization of alkaline Purex condensate with a resin packed unit removed greater than 95% of the $^{103-106}Ru$ and ^{137}Cs activity present out to 5000 column volumes throughput. Results of laboratory studies indicate that Cs recovered by ion exchange with zeolites from Purex alkaline supernatant waste, can be readily purified by alum crystallization for use in the Hanford Isotopes Plant. The amount and composition of insoluble material formed by evaporation of ammonium carbonate eluate solutions from zeolite ion exchange beds was investigated. Self-diffusion coefficients for Na, Sr, and Cs were determined in the natural zeolites erionite, phillipsite, and clinoptilolite, and the synthetic zeolites Linde 13X and 4A. (auth)

195 (HW-84603) RESEARCH AND DEVELOPMENT ACTIVITIES FIXATION OF RADIOACTIVE RESIDUES. Quarterly Progress Report, October–December 1964. A. M. Platt, ed. (General Electric Co., Richland, Wash. Hanford Atomic Products Operation). Jan. 1965. Contract AT(45-1)-1350. 54p. Dep.; $3.00(cy), 3(mn) CFSTI.

Progress is reported on the following: calcination and solidification studies, waste solidification engineering prototypes, and waste storage and treatment. Individual abstracts were prepared for each topic. (D.L.C.)

196 (IDO-12037) HEALTH AND SAFETY DIVISION ANNUAL PROGRESS REPORT, 1963. Aubrey O. Dodd, ed. (Idaho Operations Office. Health and Safety Div., AEC). Aug. 1964. 97p. Dep.; $3.00(cy), 2(mn) OTS.

197 (IDO-14599) CHEMICAL PROCESSING TECHNOLOGY QUARTERLY PROGRESS REPORT, JULY–SEPTEMBER 1962. J. R. Bower, ed. (Phillips Petroleum Co. Atomic Energy Div., Idaho Falls, Idaho). Dec. 28, 1962. Contract AT(10-1)-205. 43p.

198 (IDO-14616) CHEMICAL PROCESSING TECHNOLOGY QUARTERLY PROGRESS REPORT, JANUARY–MARCH 1963. J. R. Bower, ed. (Phillips Petroleum Co. Atomic Energy Div., Idaho Falls, Idaho). Sept. 9, 1963. Contract AT(10-1)-205. 37p.

199 (IDO-14621) CHEMICAL PROCESSING TECHNOLOGY QUARTERLY PROGRESS REPORT, APRIL–JUNE 1963. J. R. Bower, ed. (Phillips Petroleum Co. Atomic Energy Div., Idaho Falls, Idaho). Jan. 15, 1964. Contract AT(10-1)-205. 42p.

200 (IDO-14625) CHEMICAL PROCESSING TECHNOLOGY. Quarterly Progress Report, July–September 1963. J. R. Bower, ed. (Phillips Petroleum Co. Atomic Energy Div., Idaho Falls, Idaho). Feb. 24, 1964. Contract AT(10-1)-205. 51p.

201 (IDO-14635) CHEMICAL PROCESSING TECHNOLOGY. Quarterly Progress Report, October–December 1963. J. R. Bower, ed. (Phillips Petroleum Co. Atomic Energy Div., Idaho Falls, Idaho). Mar. 1964. Contract AT(10-1)-205. 40p.

202 (IDO-14636) ANNUAL REPORT OF DIVISION ANALYTICAL BRANCH FOR 1963. Ralph C. Shank, et al. (Phillips Petroleum Co. Atomic Energy Div., Idaho Falls, Idaho). June 1964. Contract AT(10-1)-205. 217p. Dep.; $6.00(cy), 3(mn) OTS.

203 (IDO-14638) CHEMICAL PROCESSING TECHNOLOGY QUARTERLY PROGRESS REPORT, JANUARY–MARCH 1964. J. R. Bower, ed. (Phillips Petroleum Co., Idaho Falls, Idaho. Atomic Energy Div.). June 1964. Contract AT(10-1)-205. 34p. Dep.; $0.75(cy), 1(mn) CFSTI.

204 (IDO-14646) CHEMICAL PROCESSING TECHNOLOGY. Quarterly Progress Report, April–June 1964. J. R. Bower, ed. (Phillips Petroleum Co., Idaho Falls, Idaho. Atomic Energy Div.). Jan. 1965. Contract AT(10-1)-205. 38p. Dep.; $2.00(cy), 1(mn) CFSTI.

205 (MLM-1152) MOUND LABORATORY PROGRESS REPORT FOR MARCH 1963. J. F. Eichelberger, G. R. Grove, L. V. Jones, and E. A. Rembold (Mound Lab., Miamisburg, Ohio). Mar. 30, 1963. Contract AT(33-1)-Gen-53. 22p.

206 (MLM-1155) MOUND LABORATORY PROGRESS REPORT FOR APRIL 1963. J. F. Eichelberger, G. R. Grove, L. V. Jones, and E. A. Rembold (Mound Lab., Miamisburg, Ohio). Apr. 30, 1963. Contract AT(33-1)-Gen-53. 43p.

207 (MLM-1266) MOUND LABORATORY PROGRESS REPORT FOR MAY 1965. Eichelberger, J. F.; Grove, G. R.; Jones, L. V. (Mound Lab., Miamisburg, Ohio). May 21, 1965. Contract AT(33-1)-Gen-53. 32p. Dep. mn; CFSTI $2.00 cy, $0.50 mn.

208 (NYO-841-1) AIR CLEANING STUDIES PROGRESS REPORT FOR JANUARY 1, 1963–JUNE 30, 1964. Richard Dennis and Leslie Silverman (Harvard Univ., Boston. School of Public Health). July 1964. Contract AT(30-1)-841. 84p.

Absorption coefficients were determined for ^{85}Kr on activated C at −13, 0, and 25°C. Desorption characteristics were found to be approximately the same as those for the absorption process. Preliminary tests of Pd–Ag diffusion cells for separation rare gases from reactor H_2 bleed streams are described. In studies of foam-capture of fission products it was found that 2.5% lauryl sulfate with dodecanal stabilizer in water can reduce confined aerosolized I_2 concentrations about 20 times. Further work is required on foam compatability with required chemical additives. Tests were made to determine the maximum height to which high-expansion foams can be pumped in high bay areas. A foam column 51 ft high was produced. Work was continued to develop diffusion boards to relieve vessel pressures in reactor accidents and remove particulate contaminants. Data were obtained on the filtering performance of activated C and silica gel packed in Al cellular cores. Fired ceramic and cellulose fiberboard were found to be very efficient in the collection of solid particles. Investigations of materials for use in collection of I_2 and Br from process off-gas were continued. Silver nitrate-saturated cellulose performed well in cleaning devices. Other materials investigated include

Cu foil, silver-impregnated glass, and Al foil. Data are included on I_2 absorption by droplets of aqueous KI and $Na_2S_2O_3$. Work continued on materials which might be dispersed in gases to provide suitable halogen absorption sites. Materials being investigated include $TiCl_4$, activated Al impregnated with $TiCl_4$, MgO, ZrO_2, dioctyl phthalate, and tributyl phosphate. Open field I_2 release studies were conducted which showed that foam systems at present are not suitable for control of short-period outside halogen releases. (J.R.D.)

209 (ORNL-3627(p.73-114)) WASTE TREATMENT AND DISPOSAL. (Oak Ridge National Lab., Tenn.).

The waste treatment and disposal development program was designed to develop a comprehensive waste management system for nuclear wastes, including their final disposal, and to estimate the cost of this operation. The effective, economic management of radioactive effluents is a prerequisite to the natural growth of a nuclear power industry. Two methods are being developed for the conversion of high-level radioactive wastes (HLW) to solids, the Potcal (pot calcination) and Potglass (fixation of wastes in glassy solids) processes. The calcined or glassy solids are contained in 88-in.-long 5- to 24-in.-dia. stainless steel pipes, which are designed to be closed by welding prior to shipment and ultimate disposal. Since there are no off-gases, future surveillance requirements during ultimate disposal should be negligible. Glassy solids, rather than porous calcines, are desirable because of their higher thermal conductivity and high degree of insolubility. These properties should result in an increase in economy and safety. A major remaining program objective is to determine costs of calcination and fixation in glassy solids and the interrelation of cost and safety. Low-level radioactive wastes (LLW), such as cooling water, canal water, floor-drain water, and other very low-salt-content waters, are treated to remove radionuclides prior to their discharge to the environment. General treatment methods being developed for this purpose include the scavenging-precipitation ion-exchange process and the scavenging-precipitation foam separation process. An improved scavenging-precipitation ion-exchange process was successfully tested in a 600-gph pilot plant, where ORNL process waste was decontaminated to a level less than 2.0% of the maximum permissible levels for human exposure. The improvements reduced the estimated treatment cost from 78¢ to 66¢ per 1000 gal for a 750,000 gal/day plant. The scavenging-precipitation foam separation process will be tested in the pilot plant. These processes were developed to serve as high-capacity, bulk-treatment operations to achieve the desirable economics inherent in this type of processing, compared with the costs of specialized treatment processes where batching operations, many analyses, and high labor requirements lead to high unit costs. Intermediate-level radioactive wastes (ILW), such as residues from processing LLW, evaporator concentrates, and concentrates from the second and third solvent extraction cycles from processing nuclear fuels and other high-salt-content wastes, can be processed by incorporation in asphalt or concrete prior to burial in a controlled area. Work on incorporation in asphalt is promising (low leach rate, high volume reduction, easily shipped) and is continuing. Economic and safety analyses on the HLW management systems show that the cost of conversion of wastes to solids and their storage in ultimate disposal sites, such as a salt mine, will be well within economic limits. These studies, conducted jointly with the Health Physics Division, will be extended to other waste-solidification systems in the future to provide a sound basis for national and international comparison of alternative systems and to serve as a guide for the direction of the experimental work.

210 (ORNL-3650(p.1-110)) REACTOR DEVELOPMENT. (Oak Ridge National Lab., Tenn.).

211 (ORNL-3665) APPLIED HEALTH PHYSICS ANNUAL REPORT FOR 1963. J. C. Hart, ed. (Oak Ridge National Lab., Tenn.). Aug. 1964. Contract W-7405-eng-26. 75p.

The waste releases from the Laboratory, atmospheric and liquid, were such that the concentration of radioactive materials in the environs were well below the maximum levels recommended by the NCRP and FRC. There was a reduction in the number of beta curies released via White Oak Creek to Clinch River, 469 curies in 1963 compared with 1436 curies in 1962. Also, there was a reduction in the average concentration of ^{131}I in milk samples processed, 13 pc/liter in 1963 compared with 96 pc/liter in 1962. The concentration of radioactive materials in the air and the background readings for 1963 did not show a significant change as compared with 1962. One employee received an exposure to tritium which resulted in an estimated body deposition of approximately 87 mc. The indicated dose to the whole body was estimated to result in a total dose of about 14 rem. Forty-three unusual occurrences were recorded during 1963 as compared with 55 for 1962. (auth)

212 (ORNL-3830, pp 96-139) WASTE TREATMENT AND DISPOSAL. (Oak Ridge National Lab., Tenn.).

Activities in a program to develop comprehensive waste management systems and to estimate costs of such systems are reported. Information is included on conversion of high-level wastes to solids, disposal of intermediate-level wastes, treatment of low-level wastes, and evaluation of waste management systems.

213 (ORNL-3830, pp 285-98) ASSISTANCE PROGRAMS. (Oak Ridge National Lab., Tenn.).

214 (ORNL-3849, pp 1-42) RADIOACTIVE WASTE DISPOSAL. Parker, F. L. (Oak Ridge National Lab., Tenn.).

Studies of sorption and desorption of iodine on mineral surfaces indicate that some species of iodine, probably I_2, can be actively adsorbed onto such surfaces. The sixth radioactive waste injection of a series was started using a mix containing Kingston fly ash as a substitute for some of the portland cement. After approximately 64,000 gal of waste had been injected, a leak developed in the high-pressure piping in the wellhead cell. The injection well was kept open to determine the bleed-back characteristics of the grout. Preparations are nearly complete for the startup of Project Salt Vault, a demonstration of the disposal of high-level waste solids in salt mines. Operation of the prototype test proved the procedures for measurement of salt flow, temperature, etc., in the demonstration. The field and laboratory work of the cooperative study of the Clinch River was terminated. Sorption tests of three soils from the southeastern United States showed that ion exchange capacity was not the controlling factor in radiocesium sorption by these soils. Desorption with sodium acetate revealed a maximum in the time-desorption relationship, particularly in the subsurface horizons. Solution-dispersion coefficients were measured in Berea sandstone as a function of flow rate. The experimental values of the dispersion coefficients varied as the 1.21 power of the solution velocity. Analysis of the critical pathways of radiation exposure to man resulting from the discharge of radioactive materials to the Clinch River has shown that fish and water consumption are most important. The U. S. Geological Survey cooperated in the preparation of a report on the soils and geology of White Oak Creek Basin. The work of the multiagency steering committee for the Clinch River Study has continued.

215 (ORNL-3868) UNIT OPERATIONS SECTION QUARTERLY PROGRESS REPORT, APRIL–JUNE 1965. Whatley, M. E.; Haas, P. A.; Horton, R. W.; Ryon, A. D.; Suddath, J. C.; Watson, C. D. (Oak Ridge National Lab., Tenn.). Dec. 1965. Contract W-7405-eng-26. 105p. Dep. mn; CFSTI $4.00 cy, $0.75 mn.

216 (ORNL–3900, pp 2–81) REACTOR DEVELOPMENT. (Oak Ridge National Lab., Tenn.).

217 (ORNL-TM-515) FISSION PRODUCTS. Progress Report, October–November 1962. R. E. McHenry, comp. (Oak Ridge National Lab., Tenn.). Apr. 9, 1963. Contract [W-7405-eng-26]. 12p.

Progress is reported on developments in the treatment of Purex waste solutions, the extraction of ruthenium from nitric acid solutions using TBP, the calibration and performance analysis of gamma-source calorimeters, and the adsorption of iodine-131 from aqueous solutions. (B.O.G.)

218 (ORNL-TM-532) FISSION PRODUCTS. Progress Report, December 1962–January 1963. R. E. McHenry, comp. (Oak Ridge National Lab., Tenn.). June 16, 1963. Contract [W-7405-eng-26]. 34p.

Progress is reported on elution of ^{137}Cs from Decalso beds, solvent extraction of cesium, solvent extraction recovery of strontium and rare earths, performance testing of air-pulsed mixer-settler, rare earth fractionation by ion exchange, and removal of radioiodine from hot off-gas system by charcoal adsorption. (M.C.G.)

219 (ORNL-TM-603) WASTE TREATMENT AND DISPOSAL QUARTERLY PROGRESS REPORT, FEBRUARY–APRIL 1963. R. E. Blanco and F. L. Parker (Oak Ridge National Lab., Tenn.). Dec. 16, 1963. Contract W-7405-eng-26. 121p.

Progress made in the development of integrated processes for the treatment and disposal of radioactive wastes resulting from reactor operations and reactor fuel processing is reported. Information is given on high-level waste pot calcination; fixation of high-level waste in phosphate glass; treatment of low-level waste by scavenging, ion exchange, and foam separation; waste disposal in deep wells and natural salt formations; mineral exchange studies; and Clinch River and White Oak Creek Basin studies. (N.W.R.)

220 (ORNL-TM-642) CHEMICAL TECHNOLOGY DIVISION, UNIT OPERATIONS SECTION MONTHLY PROGRESS REPORT, JULY 1963. M. E. Whatley, P. A. Haas, R. W. Horton, A. D. Ryon, J. C. Suddath, and C. D. Watson (Oak Ridge National Lab., Tenn.). Nov. 8, 1963. Contract W-7405-eng-26. 61p.

221 (ORNL-TM-757) WASTE TREATMENT AND DISPOSAL PROGRESS REPORT, MAY–OCTOBER 1963. F. L. Parker and R. E. Blanco (Oak Ridge National Lab., Tenn.). Apr. 1964. Contract [W-7405-eng-26]. 192p.

Progress is reported on developments in waste treatment and disposal in terms of: high-level-waste calcination; low-level-waste treatment; engineering, economics, and safety evaluation; disposal in deep wells; disposal in natural salt formations; Clinch River study; fundamental studies of minerals; and White Oak Creek basin study. (B.O.G.)

222 (ORNL-TM-785) UNIT OPERATIONS SECTION MONTHLY PROGRESS REPORT, SEPTEMBER 1963. M. E. Whatley, P. A. Haas, R. W. Horton, A. D. Ryon, and J. C. Suddath (Oak Ridge National Lab., Tenn.). Aug. 1964. Contract W-7405-eng-26. 34p.

223 (ORNL-TM-798) UNIT OPERATIONS SECTION MONTHLY PROGRESS REPORT, JANUARY 1964. M. E. Whatley, P. A. Haas, R. W. Horton, A. D. Ryon, J. C. Suddath, and C. D. Watson (Oak Ridge National Lab., Tenn.). July 1964. Contract W-7405-eng-26. 61p.

224 (ORNL-3820) APPLIED HEALTH PHYSICS ANNUAL REPORT FOR 1964. (Oak Ridge National Lab., Tenn.). June 1965. Contract W-7405-eng-26. 85p. Dep.(mn); $3.00(cy), 2(mn) CFSTI.

225 (ORNL-TM-823) UNIT OPERATIONS SECTION MONTHLY PROGRESS REPORT, OCTOBER 1963. M. E. Whatley, P. A. Haas, R. W. Horton, A. D. Ryon, J. C. Suddath, and C. D. Watson (Oak Ridge National Lab., Tenn.). Aug. 1964. Contract W-7405-eng-26. 66p.

226 (ORNL-TM-830) WASTE TREATMENT AND DISPOSAL QUARTERLY PROGRESS REPORT, NOVEMBER 1963–JANUARY 1964. R. E. Blanco and F. L. Parker (Oak Ridge National Lab., Tenn.). Aug. 1964. Contract W-7405-eng-26. 129p.

Development of a continuous-melting system for conversion of aqueous waste to glassy solids was continued. A "rising level" fixation test using simulated FTW-65 Purex waste was made. General operation was satisfactory. More than 99% of the mercury was removed from Purex FTW-65 waste solution by passing the solution through a column packed with copper. Gentle agitation of the column or imposition of a small anodic current on the copper increased the volume of waste that could be effectively treated from 76 to about 425 and 800 column volumes, respectively. Volatilization of cesium during calcination of sulfate-deficient Purex waste to 900°C was reduced by addition of aluminum. A program was initiated to define the compositional limits of desirable solid products from fixation of nonsulfate and of sulfate-bearing Purex wastes in phosphate media. Welded type 304L stainless steel exposed in boiling simulated Darex waste solution corroded less after coupling the steel to titanium. It was found that grundite clay is effective for sorbing cesium activity present in cation exchange regenerant waste. It also improved the decontamination factor for cobalt. In foam separation a method for direct, rapid radiochemical analyses by resolution of complex gamma spectra using the method of least squares was developed. Sludge-bed density values were determined to be an inverse function of both the feed flow rate and the sludge-blanket depth. Asphalt, tar, and cement were studied as media in which to disperse simulated waste-evaporator concentrates. Tritium management in wastes from fuel processing is being considered. For studies of disposal in deep wells, the relative behavior of tritiated water and chloride ions as water tracers was investigated. Construction of the injection plant was completed, and the waste pumps and injection pumps were operated to leak test the system. Gages for stress measurements in disposal in natural salt formations were tested. Diffusion coefficients based on a one-dimensional convective solution of the transport

equation were obtained from the tracer tests run in the Clinch River. A special computer program was developed to analyze gamma spectra scans of the bottom-sediment cores. Strontium sorption by clay minerals, alumina, and Conasauga shale was studied. (M.C.G.)

227 (ORNL-TM-836) UNIT OPERATIONS SECTION MONTHLY PROGRESS REPORT, FEBRUARY 1964. M. E. Whatley, P. A. Haas, R. W. Horton, A. D. Ryon, J. C. Suddath, and C. D. Watson (Oak Ridge National Lab., Tenn.). Aug. 1964. Contract W-7405-eng-26. 30p.

228 (ORNL-TM-886) CHEMICAL TECHNOLOGY DIVISION, UNIT OPERATIONS SECTION MONTHLY PROGRESS REPORT, APRIL 1964. M. E. Whatley, P. A. Haas, R. W. Horton, A. D. Ryon, J. C. Suddath, and C. D. Watson (Oak Ridge National Lab., Tenn.). Oct. 1964. Contract W-7405-eng-26. 72p. Dep.(mn); $3.00(cy), 2(mn) OTS.

Work on the sorption of water and CO_2 from streams of He is the basis for a design technique using a computer calculation of a mathematical model for deep beds. The model is based on a sorption mechanism of intracrystalline diffusion found in differential bed studies and validated by tests. In fission product recovery research, pulsed column flooding velocities for extraction, scrubbing, and stripping were run on a proposed flowsheet for the extraction of fission-product Cs from waste solutions with a substituted phenol (BAMBP). In fluoride volatility research, the previous work on the mechanism of UF_6 sorption on NaF pellets was incorporated into differential equations for predicting the performance of deep packed beds. Experimental data from runs with such beds will be compared with computer calculations. The design of an induction furnace for testing large-diameter calciners is described. (auth)

229 (ORNL-TM-936) CHEMICAL TECHNOLOGY DIVISION, UNIT OPERATIONS SECTION MONTHLY PROGRESS REPORT, JUNE 1964. M. E. Whatley, P. A. Haas, R. W. Horton, A. D. Ryon, J. C. Suddath, and C. D. Watson (Oak Ridge National Lab., Tenn.). Nov. 1964. Contract W-7405-eng-26. 32p. Dep.(mn); $2.00(cy), 1(mn) OTS.

230 (ORNL-TM-937) CHEMICAL TECHNOLOGY DIVISION, UNIT OPERATIONS SECTION MONTHLY PROGRESS REPORT, MAY 1964. M. E. Whatley, P. A. Haas, R. W. Horton, A. D. Ryon, J. C. Suddath, and C. D. Watson (Oak Ridge National Lab., Tenn.). Nov. 1964. Contract W-7405-eng-26. 42p. Dep.(mn); $2.00(cy), 1(mn) OTS.

231 (ORNL-TM-1027) CHEMICAL TECHNOLOGY DIVISION, UNIT OPERATIONS SECTION MONTHLY PROGRESS REPORT, AUGUST–SEPTEMBER 1964. M. E. Whatley, P. A. Haas, R. W. Horton, A. D. Ryon, J. C. Suddath, and C. D. Watson (Oak Ridge National Lab., Tenn.). Mar. 1965. Contract W-7405-eng-26. 35p. Dep.(mn); $2.00(cy), 1(mn) CFSTI.

232 (ORNL-TM-1069) CHEMICAL TECHNOLOGY DIVISION, UNIT OPERATIONS SECTION MONTHLY PROGRESS REPORT, OCTOBER 1964. M. E. Whatley, P. A. Haas, R. W. Horton, A. D. Ryon, J. C. Suddath, and C. D. Watson (Oak Ridge National Lab., Tenn.). May 1965. Contract W-7405-eng-26. 27p. Dep.(mn); $2.00(cy), 1(mn) CFSTI.

233 (ORNL-TM-1081) WASTE TREATMENT AND DISPOSAL SEMIANNUAL PROGRESS REPORT, JULY–DECEMBER 1964. F. L. Parker and R. E. Blanco (Oak Ridge National Lab., Tenn.). May 1965. Contract W-7405-eng-26. 168p. Dep.(mn); $5.00(cy), 4(mn) CFSTI.

Solidification tests of highly radioactive waste were completed using a 150-kw furnace. The 20-in.-dia, continuous, horizontal melter operated successfully for 250 hr until a hole formed in the melter as the result of corrosion. The average feed rate for the continuous melter was between 30 and 45 liters/hr for extended times. A computer code was developed for retrieving and correlating the mass of data generated in the program for developing glassy melts for the fixation of high-level waste. Initial read-out was obtained on 356 ORNL melt compositions. Melts containing simulated fission products from a fuel burnup of 10,000 Mwd/t were run in semiengineering-scale continuous melters. In these tests, phosphate volatility was generally <0.5%, and sulfate volatility <5.0%. There were anomalous results with these constituents as well as with Fe and Ru. Of candidate construction materials exposed in a continuous melter at 950°C, only Pt was unattacked. A 50% Cr–50% Ni alloy was relatively resistant and might prove usable in a melter operating at a lower temperature. Inconel ranked third. Stainless steel types 304L, 310, 347, and Carpenter 20SCb, Hastelloys B, C, and F, Ni-o-nel, Nichrome and BMI-HAPO 20 were the other materials tested. Aspects of foam separation in decontamination of low-radioactivity-level process waste water (LLW) were studied. Studies in support of the ORNL foam-separation pilot plant included an evaluation of various gas distributors and an analysis of the dynamics of a foam column having a square cross-section. The pilot plant work on the scavenging-precipitation foam-separation process was terminated. Mixed phosphates present in raw low-radioactivity-level waste (LLW) inhibited Ca precipitation in scavenging-precipitation ion-exchange processes. Use of ALCOA F-1 removed more than 99% of four phosphates tested for more than 3000 bed volumes (BV) (equivalent to 15,000 BV for a feed 1 ppm in phosphate) in downflow operation through a packed bed of solid. Evaluation of a water-recycle flowsheet was made consisting of (1) batch clarification of LLW using zeta-potential-controlled additions of coagulant aid, (2) demineralization of the supernatant through a mixed cation-anion ion exchange column, and (3) treatment with a column of activated C. Overall decontamination factors at 10% conductance breakthrough or 200 BV were estimated to be: >200 for ^{106}Ru, >1300 for ^{60}Co, 300 for ^{137}Cs, 1000 for ^{90}Sr, >600 for ^{144}Ce, and >25 for ^{95}Zr–Nb. Demineralization was accomplished for 175 BV with waste water and with tap water, 210 BV. Asphalt products containing 20 to 60 wt % solids from simulated ORNL waste-evaporator bottoms were leached with water for periods up to 9 months. The leach rate of an asphalt product containing 60 wt % solids was not effected by irradiation to an absorbed dose of 10^8 rads. An estimate of the costs of perpetual tank storage of wastes is being undertaken to establish a basis for comparison with costs for alternative waste-management schemes. Solution-dispersion coefficients were measured in Berea sandstone as a function of flow rate. The experimental values of the dispersion coefficients varied with the 1.22 power of the solution velocity. Preparations for demonstration of disposal of highly radioactive solid wastes in a salt mine are on schedule. Most of the topside facility and mine renovations are complete, and the waste-charging shaft has been installed. Mining of the experimental area is about 67% complete. In the lower 21 miles of the Clinch River, an inventory showed that about 200 curies of radioactivity are in the bottom sediments. Over three-quarters of it is ^{137}Cs. Other principal radionuclides found in the sediment are ^{60}Co, ^{106}Ru, ^{90}Sr, and rare earths. Comparison of the inventory to quantities of these radionuclides released over White Oak Dam seems to indicate that only the retention of Cs (about 20%) is important. The desorption of I_2 from Clinch River flood-plain sediment with either water or air gives a rapid initial desorption of the loosely bound I_2. This is followed by a slower desorption of more tightly bound I_2. The amount of bound I_2 is a function of the temperature of the system. (J.R.D.)

234 SAFETY WITH RADIOISOTOPES. Sula J. Marsalis (Mississippi State Coll. for Women, Columbus, Miss.); and E. E. Beauchamp. Isotopes Radiation Technol., 2: 172-81(Winter 1964-65).

Safety in the handling of radioisotopes is discussed, and the hazards are related to those of other items of commerce. Safety precautions used in radioisotope processing and shipping are described. Safety in utilization of the isotopes and in waste disposal is also considered. (M.C.G.)

235 RADIOACTIVE WASTE RESEARCH AND DEVELOPMENT ACTIVITIES—A REPORT OF PROGRESS. Belter, Walter G.; Regan, William H. (Atomic Energy Commission, Washington, D. C.). J. Water Pollution Control Federation, 37: 316-35(Mar. 1965).

Basic objectives of the radioactive waste management research and development program are discussed. The nature of radioactive wastes is reviewed. Objectives of the waste management operations are outlined. The use of a scavenging-precipitation ion-exchange process for waste processing is discussed. Foam separation methods of waste processing are described. Hydraulic fracturing of shale for waste disposed is considered. High-level waste solidification by the various methods of calcination and the phosphate glass process is described. Economic aspects of high-level waste treatment and disposal are discussed. Disposal of gaseous effluents is also considered. Environmental studies in support of the waste-disposal program are described. (M.C.G.)

236 SELECTIVE SORPTION REACTIONS OF CESIUM WITH SOIL MINERALS. Tamura, T. Nucl. Safety, 5: No. 3, 262-5(Spring 1964).

A review of work done on retention of Cs from radioactive waste solutions by soils, Cs sorption by clay minerals, and Cs–K relations with soils.

237 OPERATING EXPERIENCE AT SAXTON. Howard, D. E. Nucl. Safety, 5: No. 3, 269-77(Spring 1964).

The Saxton plant is described and information is presented on plant operation, operator training and licensing, reactor scrams, radiation levels and shielding, and waste disposal.

238 SAFETY PROBLEMS ASSOCIATED WITH THE DISPOSAL OF RADIOACTIVE WASTE. Walton A. Rodger. Nucl. Safety, 5: 287-302(1964).

A review of waste problems in the nuclear industry is presented. Discussions are included concerning areas of potential hazard, methods of dealing with hazards, and effects of accidents on the waste disposal experience. (J.R.D

239 NUCLEAR SAFETY. Volume 6, Number 3. Cottrell, William B., ed. (Oak Ridge National Lab., Tenn.). Spring 1965. $0.70(GPO) (domestic), $0.90 (GPO) (foreign). (One of the Technical Progress Reviews prepared quarterly for the Division of Technical Information, USAEC). 90p. Dep.

An article on safety-system design technology is presented along with a book review on reactor physics and control. Other information is included on radiation protection in space, transportation of irradiated fuel elements, status of SPERT, fast reactor safety problems, testing of rock for underground reactor containment, removal of radioiodine from gases, waste disposal from U mills and mines, fallout from nuclear weapons, Indian Point power station operating experience, nuclear accident at Wood River Junction R.I., regulatory action on reactor projects, and changes in reactor licensing regulations. (J.R.D.)

240 UNITED STATES AND SOVIET UNION WASTE-DISPOSAL STANDARDS. Parker, F. L. Nucl. Safety, 6: 433-6; 439-40(Summer 1965).

The standards for waste disposal in the two countries are compared. Disposal of liquid, gaseous, and solid wastes is discussed along with findings and conclusions.

241 SELECTIVE SORPTION REACTIONS OF STRONTIUM WITH SOIL MINERALS. Tamura, Tsuneo. Nucl. Safety, 7: 99-103(Fall 1965).

The reactions of radiostrontium with soils and clay minerals and with chemical solutions of soils and clay minerals containing microconcentrations of the radionuclide and macroconcentrations of dissolved stable ions are reviewed. In the reactions of strontium in soil-related systems, it is important not only to consider the clay minerals with their ion-exchange properties but also to remember that these systems may contain limestone and the hydrous iron and aluminum oxides. The presence of the latter components appears to have little direct influence on cesium sorption, but these components can play an important role in strontium sorption.

242 WASTE DISPOSAL. Reactor Fuel Process., 6: No. 4, 39-52(1963).

Developments in liquid-waste treatment by ion exchange and foam separation are discussed. Methods of converting high-level-activity wastes to solids by spray calcination and pot calcination are described. Engineering, economic, and safety evaluations of interim storage of calcined wastes in water-filled canals are presented. The feasibility of waste disposed by hydraulic fracturing into shale is discussed. Tests of Wyoming bentonite as a cover material for preventing the leaching of buried solid wastes are described. The CSREX solvent-extraction process for separating fission products from Purex process waste is described along with improvements in other methods. (34 references) (M.C.G.)

243 WASTE DISPOSAL. Reactor Fuel Process., 7: No. 1, 54-70(Winter 1963-64).

Development of methods of waste processing, including treatment of liquid wastes by ion exchange and foam separation and conversion of high-level-activity wastes to solids, is reviewed. Recovery of fission products from wastes is discussed. Land burial operating experience and Savannah River Plant is described. (32 references). (M.C.G.)

244 WASTE DISPOSAL. Reactor Fuel Process., 7: No. 2, 131-45(1964).

Research progress or liquid-waste treatment by ion exchange and foam separation, conversion of high-level-activity wastes to solids, final disposal methods, recovery of fission products, and corrosion in waste processing is reviewed. (23 references). (M.C.G.)

245 WASTE DISPOSAL. Reactor Fuel Process., 7: No. 3, 200-13(1964).

Development of equipment and methods of waste processing, including treatment of liquid wastes by evaporation and conversion of high-level-activity wastes to solids, is reviewed. The conversion of wastes to glasses is discussed along with pot, spray, fluid-bed, and rotary-ball calcination. (30 references). (N.W.R.)

246 WASTE DISPOSAL. Reactor Fuel Process., 8: No. 1, 37-49(Winter 1964-65).

A review is presented on waste disposal methods, including management of tritium in fuel-processing wastes, ion exchange of liquid wastes, conversion of high- and intermediate-level-activity wastes to solids, and recovery of strontium from Purex-waste tank sludge. (P.C.H.)

247 WASTE DISPOSAL. Reactor Fuel Process., 8: 111-23(Spring 1965).

The treatment of liquid wastes by foam separation and freeze drying is discussed briefly. Development of methods for the conversion of high-level-activity wastes to solids by calcination and glass fixation is reviewed. Final disposal methods (hydraulic fracturing in shale and calcined-solids storage in salt mines) are also discussed. Work on the recovery of fission products from wastes is surveyed. Other literature on waste treatment and disposal is reviewed. (D.L.C.)

248 WASTE DISPOSAL. Reactor Fuel Process., 8: 172-82(Summer 1965).

Information is reviewed concerning treatment of low- and intermediate-level wastes by incorporation in asphalt. Processes for conversion of high-level-activity wastes to solids are described. Information is also included on recovery of fission products.

249 REACTOR FUEL PROCESSING. Volume 8, Number 4. Lawroski, Stephen (ed.). One of the five Quarterly Technical Progress Reviews published for the Division of Technical Information, USAEC. Argonne National Lab., Ill., 1965. 60p. GPO $0.70 Domestic, $0.90 Foreign.

RADIOLOGICAL HAZARDS AND HEALTH CONSIDERATIONS

General

See also the following references:

34, 355, 508

250 A REVIEW OF THE ENVIRONMENTAL HAZARD OF RADIOACTIVE WASTES. Joe O. Ledbetter (Univ. of Texas, Austin). 15p. (CONF-672-1)

From American Society of Civil Engineers Environmental Engineering Conference, Salt Lake City, May 1964.

Types of radioactive wastes and hazards, both environmental and human, from such wastes are defined. Sources of radioactive waste hazards are identified, and methods for radioactive waste disposal are evaluated. (H.M.G.)

251 (DPSPU-63-30-42) RADIOACTIVE WASTE: RADIATION-PROTECTION STANDARDS AND CATEGORIZATION. W. C. Reinig (Du Pont de Nemours (E. I.) & Co. Savannah River Plant, Aiken, S. C.). Dec. 1963. Contract AT(07-2)-1. 11p. (CONF-367-1)

From Annual Meeting of the National Technical Task Committee on Industrial Wastes, St. Louis, Mo., Dec. 1963.

Hazards from the discharge of radioactive wastes to the environment are reviewed. It is pointed out that more is known today about the effects of ionizing radiation than about many other noxious materials that man has introduced into his environment. The formulation of radiation protection standards is reviewed. It is postulated that these standards will be relaxed or tightened as more knowledge becomes available about the behavior of radionuclides in the environment and the biological effects of low levels of long-continued irradiation. (C.H.)

252 (NAVDOCKS-MO-218) RADIOACTIVITY IN WATER SUPPLY AND WASTE WATER SYSTEMS: PEACETIME DETECTION AND CONTROL. (Bureau of Yards and Docks). Oct. 1962. 80p.

Discussion is presented on: the causes and characteristics of radioactivity; biological effects of radiation; allowable concentrations of radioisotopes; methods of determining nuclide concentrations and removing radioisotopes; the detection, analysis, and monitoring of radioactivity in water supplies and wastes; personnel protection through safety precautions in plant operations; and the authority and responsibility of the bureaus of the Navy Department and the field activities concerned. Practical procedures for determining the gross alpha and gross beta radioactivity of water, soil, and biological samples are appended. (P.C.H.)

253 ENVIRONMENTAL HAZARD OF RADIOACTIVE WASTES. Ledbetter, Joe O. (Univ. of Texas, Austin). J. Sanit. Eng. Div. Am. Soc. Civil Engrs., 91: 59-66(Feb. 1965).

Hazards involved in the following three methods of radioactive waste disposal are discussed: dilution to acceptable environmental levels; storage for decay; and reclamation to useful products. The wastes must variously be handled, transported, processed, and stored and, in addition, the accidental spills and releases must be cleaned up and isolated. All of these operations involve the possibility of hazardous radiation exposures. (TCO)

254 RADIOACTIVE WASTES. Lancet, 2: 1285 (Dec. 14, 1963).

A brief report of a parliamentary debate concerning the potential hazards and the level of contamination caused by release by hospitals of radioactive materials into sewage, especially radioiodine and ^{224}Ra. (BBB)

255 THE PRINCIPAL NATURAL AND ARTIFICIAL RADIOACTIVE SUBSTANCES ENCOUNTERED IN THE SURROUNDINGS. G. Vacca. Maroc Med., 43: 159-66 (1964).

Information on radioactive contamination of surroundings from natural and artificial sources is summarized. Principal sources of natural radioactivity are radioactive substances in the soil and cosmic radiations producing radionuclides by impact on the atmosphere. The main source of artificial radioactivity is atomic industry, which produces radioactive contamination of the environment during extraction of radioactive minerals, isotope separation, construction and use of atomic piles, use of radioisotopes, and disposal of radioactive wastes. It was concluded that so far, few accidents accompanied with serious contamination of the surroundings have occurred during the 20 years of existence of atomic industry. (Excerpta Med., Sect. XXIII, 2: No. 1, (Jan. 1965))

256 ENVIRONMENTAL CHANGE AND RESULTING IMPACTS ON HEALTH. REPORT OF A WHO EXPERT COMMITTEE. World Health Organ., Tech. Rept. Ser., No. 292, 1-23(1964).

Health problems connected with radiation, air pollution, liquid and solid wastes, food hygiene, and other manifestations of environmental change in modern life are reviewed. Special attention is given to the increase in sources of radiation, air pollution, and the great rise in the volume of wastes requiring new and more effective methods of disposal. It was concluded that the future should be viewed with modified optimism rather than with alarm: the danger constituted by radiation, for example, is very limited in terms either of the population at risk or of the area involved. It is recommended that high priority be given in health control programs to food hygiene, the provision of safe drinking water, and the proper treatment of wastes, in the light of the character and financial resources of each individual country. (P.C.H.)

Disposal Policies and Regulations

See also the following references:

34, 35, 138, 143, 168, 240, 501, 842, 856

257 THE MANAGEMENT OF RADIOACTIVE WASTES PRODUCED BY RADIOISOTOPE USERS. (International Atomic Energy Agency, Vienna (Austria)). 1965. Safety Series No. 12. 59p. $1.50; S 31,50; 9/-stg; F.Fr.6,-; DM 4,80 (IAEA). (STI/PUB/87)

A code of practice is presented for the handling of the small quantities of waste produced by the use of radioisotopes in laboratories. The chapters deal respectively with the control of radioisotope uses and waste disposal, waste management by individual users of radioisotopes, waste management in a large central facility, and waste storage and containment. The types of waste associated with some radioisotope uses and examples of national regulations are appended.

258 THE RADIOACTIVE SUBSTANCES ACT 1960. N. D. Baines (Ministry of Housing and Local Government, London). Ann. Occupational Hyg., 7: 47-51(Jan. 1964).

The Radioactive Substances Act 1960 that came into effect on December 1, 1963 requires persons who keep and use radioactive material to be registered with the Ministry of Housing and Local Government, and to obtain authorization for the accumulation and/or disposal of radioactive waste. Exemption from registration or authorization is granted where the use or nature of the material does not call for the full application of the controls in the Act. Exemption is usually subject to various conditions. Methods of radioactive waste disposal are described briefly. (auth)

259 IMPACT OF LEGAL CONTROL OF DISCHARGE OF RADIOACTIVE MATERIALS ON THE RESPONSIBLE DEPARTMENTS. G. Wortley (Ministry of Agriculture, Fisheries and Food, London). Food Agri. Organ. U. N., FAO At. Energy Ser., No. 5, 313-29(1964).

Legislative control of the disposal of radioactive wastes is discussed in general terms. Legislation in the United Kingdom covering the discharge of radioactive materials is reviewed in detail. (C.H.)

260 IMPACT OF LEGAL CONTROL OF DISCHARGE OF RADIOACTIVE MATERIALS ON THE RESPONSIBLE DEPARTMENTS. I. C. Roberts and S. H. Small (International Atomic Energy Agency, Vienna). Food Agri. Organ. U. N., FAO At. Energy Ser., No. 5, 345-66(1964).

Problems involved in the international control of the discharge of radioactive materials resulting from peaceful uses of nuclear energy are reviewed. Current waste disposal practices are described. It is concluded that so far national controls have been adequate for the safeguarding of international interests but that future developments will almost certainly require some form of international agreement for the control of the discharge of radioactive wastes. (C.H.)

261 THE PRESENT STATE OF RADIATION CONTROL AT THE NATIONAL INSTITUTE OF RADIOLOGICAL SCIENCES. Hiroshi Sumida (National Inst. of Radiological Sciences, Chiba, Japan). Genshiryoku Kogyo, 9: No. 10, 49-58(Oct. 1963). (In Japanese)

The present state of radiation control at the National Institute of Radiological Sciences is described in detail, the author's personal views being added occasionally. The article consists of (1) how the radiation control and protection should be, (II) the works of radiation control and protection, (III) radiation control, (IV) radioactive waste disposal, (V) radiation accidents, and (VI) problems for the future. Radiation control is, in general, divided into direct protection of personnel from radiation and indirect protection by monitoring the radiation or radioactive facilities, but adequate radiation control is achieved only when the two are carried out side by side. In (II) above, the Radiation Health Control Unit, the Facility Safety Unit, and the Decontamination and Waste Disposal Unit are described, and in (III), the present state of radiation control of the various Institute facilities (RI experiments, α-ray experiments, radiation generators, wards, etc.), blood examination, survey meters, etc. In (IV) are described the treatment and disposal facilities, the processes of waste treatment, liquid waste, and the radioactive feces and urine. Six years after the establishment, the facilities of the Institute are varied, and radiation uses range from basic studies to diagnosis and therapy. The radiation control and protection carried out under such conditions should provide examples of radiation control and protection for other similar institutions, and the practices and methods must be further improved. When considering the organization of radiation control of an institution, the factors listed below will also have to be considered: (1) the types of radiation work, (2) the scale of the institution, (3) the organization and structure of the institution, and (4) the type of radiation control service. (Nucl. Sci. Abstracts of Japan, 3: No. 2(Apr. 1964))

262 PROGRESS AND PROBLEMS IN RADIATION STANDARDS AND RADIOACTIVE WASTE DISPOSAL. Ramey, James T. Nucl. Energy, 333-41(Nov.-Dec. 1965).

The development and application of radiation standards are reviewed and the specific application of standards in the field of radioactive waste disposal are discussed. Concepts of radiation standards, the role of ICRP and NCRP, the role of the Federal Radiation Council, the role of the Public Health Service, the role of the AEC, the responsibilities of the states, and problems of standards setting are considered. Radiation protection in the AEC radioactive waste disposal program is described. The concepts of concentrate and contain and dilute and disperse for waste disposal are considered. Estimates of wastes from the civilian power program are presented.

263 THE CONTROL OF RADIOACTIVITY IN WATERS AND SOIL. O. Jaag (Eidgenoessische Technische Hochschule, Zurich). Oncologia, 16: (3) 230-44(1963). (In German)

The establishment of research atomic reactors in Switzerland and conduction of atomic bomb tests in other countries has necessitated extensive organization of facilities to ensure the control of radioactivity in surface waters, rivers, lakes, reservoirs, and ground waters. For instance, the concentration of radioactivity permitted in the Aare river at the Wuerlingen reactor is $1/20$ of the value for drinking water (1000 pC/l). Very close attention is given to concentrations of the deleterious radioisotopes, such as $Sr^{90}-Y^{90}$, Cs^{137}, and I^{131}. Air and water contamination studies showed concentration peaks in summer 1957, fall 1958, and fall 1961 resulting from atomic bomb tests. Water contaminations remained below dangerous levels. Water plants and animals can markedly accumulate the activity, though, and plankton from the Aare river had activities of 10,000 to 15,000 pC/kg dry weight when the water concentration was 4 to 5 pC/l, and 40,000 to 200,000 pC/kg when the water concentration was 6 to 8 pC/l. In grasses the activity level (1961) was 14 to 20 pC/g dry weight and 200 pC/g ash. The Sr^{90} concentration was over 4 pC/g ash. The Sr^{90} concentration in the top 5 cm of soil (air dried) in Wuerlingen was ~170 pC/kg and 50 to 80 pC/kg in the next 10 cm below. In Zuerich the Sr^{90} concentration in the upper layer was ~110 pC/kg. In order to purify contaminated waters, ion exchangers, precipitants, adsorbents, and evaporation procedures are used. Mixed ion-exchange beds are extremely effective for removal of many radioisotopes, but many adsorbents and precipitants are much less useful. Peat from the Jura region removes over 99% of Sr^{90} and 90 to 95% of mixed radioisotopes from cistern water. This water contains as much as 20 to 50 pC/l, but only 3 pC/l of this is Sr^{90}, so that it is still drinkable. A comparatively simple glass filtration apparatus can be used with peat for purification of cistern water. (BBB)

264 SAFE DISPOSAL OF RADIOACTIVE WASTE IN WATER AND ATMOSPHERE. Davids, J. A. G. (Water-Bodem-Lucht, 54: No. 4, 74-84(1964).

ICRP recommendations on disposal of radioactive waste in water and the atmosphere are discussed. Some practical problems are considered: the disposal of ^{41}Ar in the atmosphere, the disposal in the Thames by the center at Harwell, the disposal in the Columbia river at Hanford, the disposal of liquid waste in the sea, and the disposal of low-level waste water at Petten. (J.V.E.)

265 RECOMMENDATIONS OF THE INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION. ICRP PUBLICATION 5. REPORT OF COMMITTEE V ON THE HANDLING AND DISPOSAL OF RADIOACTIVE MATERIALS IN HOSPITALS AND MEDICAL RESEARCH ESTABLISHMENTS. New York, Pergamon Press, 1965. 55p.

Handling and disposing of radioactive materials in hospitals and small laboratories are discussed. Practices are described by which it should in general be possible to maintain radiation doses below the maximum permissible levels recommended by the Commission. Topics covered include: responsibility for radiation safety measures, medical examinations, personnel monitoring, environmental monitoring, procedures in work with sealed sources, procedures for work with unsealed radioactive materials, decontamination of personnel, decontamination of buildings and equipment, disposal of radioactive waste, storage of radioactive materials, transportation of radioactive materials, discharge of patients during treatment with radioactive materials, and disposal of radioactive corpses. Recommendations of the commission are given. (M.C.G.)

Environs Monitoring

See also the following references:

24, 34, 101, 122, 184, 211, 212, 217, 235, 317, 435, 456, 471, 482, 502, 817, 823, 828, 837, 848, 865, 915, 918, 922, 995

266 (A/AC.82/G/L.986) CONTROLE RADIOLOGIQUE DE L'ENVIRONNEMENT DU C.E.N. A MOL. ORGANISATION ET TECHNIQUES DE MESURES. (Radiological Monitoring of the Surroundings of C.E.N. at Mol. Organization and Techniques of Measurement). Boulenger, R.; Colard, J.; de Maere, X.; Koch, G.; Osipenco, A.; Tytgat, R. (Centre d'Etude de l'Energie Nucleaire, Brussels (Belgium)). Jan. 1964. 88p. (R-2298). Dep. mn.

The radiological monitoring of the C. E. N. environment is done by monitoring chains. The first monitoring chain measures the concentration of mixtures of radioisotopes or of particular isotopes in the effluents of the plant. The second monitoring chain is located at a slight distance from the nuclear installations and makes a continuous measurement of the radioactivity of dust in suspension in the air and the ambient radioactivity. Samplings were also made of milk and of surface water or rain. The third monitoring chain is composed of sampling points of dust in suspension in the air and of precipitations at several tens of km from C. E. N. The techniques and methods used in each of these monitoring chains are described. (J.S.R.)

267 (ANL-6945) ENVIRONMENTAL RADIOACTIVITY AT ARGONNE NATIONAL LABORATORY. REPORT FOR 1962 AND 1963. J. Sedlet and F. S. Iwami (Argonne National Lab., Ill.). Jan. 1965. Contract W-31-109-eng-38. 62p. Dep.; $3.00(cy), 2(mn) CFSTI.

Results are presented from the environmental monitoring program at Argonne National Laboratory during 1962 and 1963. The purposes of the program are to measure the natural radioactive content of the ANL site and its surroundings, and to determine the origin, identity, and magnitude of any radioactivity not naturally present. Of primary interest is the detection of radioactive materials released to the environment as a result of Argonne operations. The outstanding feature in the radioactivity of the environment during 1962 and 1963 was the relatively high level of fallout activity resulting from the U.S.S.R. tests in 1961 and 1962 and the U. S. tests in 1962. Fallout averaged somewhat higher in 1963 than in 1962, although there was no atmospheric testing during 1963. By the end of 1963 the expected decrease in fallout was noted. The fallout activity, as measured by the beta activity in 24-hour air-filter samples, averaged 6.7 pC/m^3 in 1962 and 7.2 pC/m^3 in 1963. From 1953, when routine measurements of this type were begun, through 1961, the annual averages have ranged from 0.5 pC/m^3 in 1954 to 4.1 pC/m^3 in 1961. The amount of fallout activity deposited by precipitation was also relatively high, approximately 2500 mC/sq mi during 1962 and 1963. From 1953 to 1961, the annual totals varied from 640 mC/sq mi in 1960 to about 950 mC/sq mi in 1961. The highest total recorded was 11,000 mC/sq mi in 1952. Most of this activity was carried down by a rainfall in June 1952. Most of this activity was carried down by a rainfall in June 1952, which occurred a few days after an atmospheric nuclear detonation in Nevada. In 1962, the amount of fallout activity in air reached a peak in April and May and again in November and December. The increase in the spring was the combined result of a spring maximum in stratospheric fallout of fission and activation products produced in the fall of 1961, and of tropospheric fallout of fission products produced in tests conducted in the spring of 1962. The bulk of the fission products detected in the spring of 1962 originated in the U. S. Pacific tests. The increase in fallout

activity in the fall of 1962 was due primarily to fission products produced by USSR tests conducted at that time. In 1963, a spring maximum was again evident in April, May, and June. During the second half of the year, the fallout activity in air decreased steadily, from an average of 12.4 pC/m^3 in June to 1.0 pC/m^3 in December.

268 (BNL-915) 1963 ENVIRONMENTAL RADIATION LEVELS AT BROOKHAVEN NATIONAL LABORATORY. Hull, A. P. (Brookhaven National Lab., Upton, N. Y.). Nov. 1964. Contract AT(30-2)-Gen-16. 19p. Dep. mn; CFSTI $1.00 cy, $0.50 mn.

Measurements were made during 1963 in the environment of Brookhaven National Laboratory and vicinity of natural background radiation levels (including fallout contributions) and of radiation from the ^{41}Ar content of the BGRR cooling-air effluent, from an ecology field gamma source, and from the discharge of low-level liquid wastes. The natural background, as measured 6 in. above roof level, reached 6.22 mR/wk in August. This is attributable to fallout from the large-scale weapons testing conducted during the fall of 1962 and is the highest level on record at this location. The corresponding background at an elevation of 3 ft was ≈65% of that at 6 in. During 1963, radiation levels attributable to Laboratory operations were maintained well below the established MPE, MPC, and RPG levels for whole-body exposures, air particulate concentrations, and liquid effluent concentrations. Radiation levels attributable to fallout were also below these established limits. (TCO)

269 (BNWL-90) EVALUATION OF RADIOLOGICAL CONDITIONS IN THE VICINITY OF HANFORD FOR 1964. Wilson, R. H. (ed.) (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). July 1965. Contract AT(45-1)-1830. 206p. Dep. mn; CFSTI $6.00 cy, $1.25 mn.

The evaluation of results obtained from the Hanford environmental surveillance program for 1964 indicates that most of the environmental radiation exposure for the majority of persons living in the neighborhood of the Hanford project was due to natural sources and world-wide fallout rather than to Hanford operations. Of the low-level wastes that are released to the environment from the Hanford plants, neutron-induced radionuclides present in reactor cooling water discharged to the Columbia River continued to be the source of greatest potential exposure to people in the environs. The primary pathways of exposure from this source are drinking water derived from the river, consumption of fish and waterfowl which inhabit the river, and foodstuffs grown on land irrigated with water pumped from the Columbia downstream from Hanford. ^{131}Iodine in the Hanford environs remained at very low concentrations in 1964. The Chinese nuclear test on October 16 caused a brief increase in ^{131}I, but concentrations soon returned to the low levels experienced during most of 1964. The postulated maximum exposure from ^{131}I to the thyroid of a small child amounted to only about 5% of the Radiation Protection Guide recommended for individuals by the Federal Radiation Council.

270 (BNWL-122, pp 198-201) REACTOR EFFLUENT MONITORING - 1964. Olson, P. A.; Nakatani, R. E. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.).

Since monitoring of Hanford's reactor effluent started in 1945, the biological effects of the primary factors (heat, chemical toxicity, and radiation) have been defined in terms of growth and mortality for young chinook salmon. Comparisons between groups of chinook fingerlings reared in regular and cooled reactor effluent showed the importance of the accumulative effect of small temperature increments. Higher mortalities were observed in the present studies with local stock chinooks than in past studies with Puget Sound chinooks, due, perhaps, to physiological differences in racial stocks. Data are included on the content of ^{32}P and ^{65}Zn in fish reared in regular effluent for 10 weeks.

271 (BNWL-165) EVALUATION OF RADIOLOGICAL CONDITIONS IN THE VICINITY OF HANFORD, JANUARY-JUNE 1965. Wilson, R. H. (ed.) (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). Nov. 1965. Contract AT(45-1)-1830. 29p. Dep. mn; CFSTI $2.00 cy, $0.50 mn.

Surveillance of the Hanford environs during the first half of 1965 showed that the amounts of radioactive materials present during this period were well within the nationally accepted limits at all times and that releases of radioactive materials from the Hanford plant were adequately controlled. ^{32}P released with Hanford effluent in the Columbia River continued to be the radioisotope that contributed the most exposure to individuals who ate locally caught fish in quantity.

272 (BNWL-SA-465) ENVIRONMENTAL MONITORING. Honstead, J. F.; Soldat, J. K. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). Mar. 25, 1965. Contract AT(45-1)-1830. 73p. Dep. mn. CFSTI $3.00 cy, $0.75 mn.

The environmental radiation monitoring program at the Battelle Memorial Institute Pacific Northwest Laboratory, Richland, Washington is described in detail. Criteria for environmental monitoring programs are reviewed and methods used for pre-operational surveys, emergency surveys, operational surveys, radiological monitoring of the Columbia River and the atmosphere are described. Exposure pathways that must be combined to assess the total exposure of employees and the public are discussed.

273 (CONF-337-7) MONITORING THE OFF-SITE GASEOUS RELEASES AT R.C.N. RESEARCH ESTABLISHMENT. J. R. D. Stoute (Reactor Centrum Nederland, Petten). [nd]. 7p.

From International Conference on Radioactive Pollution of Gaseous Media, Saclay, France, Nov. 1963.

Procedures used at the Reactor Centrum Nederland Research Establishment, Petten, the Netherlands, for monitoring gaseous releases from the reactor, chemistry building, and dismantling hotlab are outlined. The gaseous effluents are measured at points where little dilution of the contamination has occurred. Provisions are made for enlisting the help of village constables for environmental monitoring in case of an accidental release. Periodic analyses of filters from the emergency monitors are used as a check on environmental contamination in the environs. (C.H.)

274 CURRENT PRACTICE IN THE SURVEY OF THE WORKING ENVIRONMENT AT THE WANTAGE RESEARCH LABORATORY. G. W. Cox (United Kingdom Atomic Energy Authority, Wantage, Berks, Eng.). 9p. (CONF-365-11)

From Symposium on Environmental Monitoring, Berkeley, Gloucestershire, England, Oct. 1963.

The use of large sealed and small sealed and unsealed sources at the Wantage Research Laboratory is discussed. The health physics program for monitoring at the laboratory is given in detail. The amount of environmental monitoring required at Wantage is kept to a minimum by careful planning at the initial stages of a program of work, due consideration being given to such matters as good design, provision of adequate shielding, use of minimum quantities of radioactive material consistent with requirements and the use, wherever possible, of radioisotopes with short half lives so that any residual activity decays quickly. (P.C.H.)

275 ENVIRONMENTAL MONITORING IN THE OPERATION OF A NUCLEAR POWER STATION. B. Emmerson (Central Electricity Generating Board, London). 30p. (CONF-365-13)

From Symposium on Environmental Monitoring, Berkeley, Gloucestershire, England, Oct. 1963.

The environs monitoring program by the health physics staff at nuclear power plants in England is described. Various tests are described and the specific duties of monitors during burst cartridge detection tests are listed. A typical civil nuclear station health physics staff tree is illustrated. (P.C.H.)

276 ENVIRONMENTAL SURVEYS AROUND RESEARCH REACTORS. K. Boddy. 12p. (CONF-365-19)

From Symposium on Environmental Monitoring, Berkeley, Gloucestershire, England, Oct. 1963.

Experience gained from environmental surveys during the commissioning and operation of the 5 Mw MERLIN Reactor and the proposed program for the UTR-100 Scottish Universities Reactor is reported as a guide to the extent and pattern of the environmental surveys required at other sites. (P.C.H.)

277 AN ALTERNATIVE TO DISTRICT SURVEY SAMPLING. W. N. Saxby (Atomic Weapons Research Establishment, Aldermaston, Berks, Eng.). 8p. (CONF-365-25)

From Symposium on Environmental Monitoring, Berkeley, Gloucestershire, England, Oct. 1963.

The possibility of adverse radiological effects on individuals in the neighborhood of a modern atomic plant in its routine work can be reduced below a significant level by the design of, and the operational procedure in, the plant. Of the methods of checking that the plant is continuing to operate correctly, monitoring at the discharge outlets is the most convenient and economical and the most sensitive procedure, and is the preferred alternative to District Survey sampling, provided that the plant has been suitably designed. This method is sufficiently sensitive for changes leading to increased discharges to be found and for corrections to be made quickly before significant levels are reached in the neighborhood. (auth)

278 PROPORTIONAL SAMPLING FOR RADIONUCLIDES IN RIVER SYSTEM BELOW OAK RIDGE. M. A. Churchill (TVA, Chattanooga, Tenn.). 17p. (CONF-402-1)

From American Geophysical Union 43rd Annual Meeting, Washington, D. C., Apr. 1962.

The methods used and early results obtained in sampling the Clinch and Tennessee Rivers at several locations between Oak Ridge and a point (Chattanooga) some 120 river miles downstream are reported. The techniques used permitted not only determinations of concentrations of specific radionuclides, but also the proportions of the quantities released that, in the flowing water, reach various downstream locations. (P.C.H.)

279 (DPSPU-64-30-2) EFFECT OF THE SAVANNAH RIVER PLANT ON ENVIRONMENTAL RADIOACTIVITY. Semiannual Report, January–June 1964. C. Ashley (Du Pont de Nemours (E. I.) & Co. Savannah River Plant, Aiken, S. C.). Contract [AT(07-2)-1]. 15p.

Results of the environmental monitoring program for the atmosphere, for vegetation and food, and for water are presented for the period January 1 through June 30, 1964. The quantity of radioactive waste released by the Savannah River Plant to its environs was, for the most part, too small to be distinguished from natural background radiation, or was obscured by world-wide fallout from nuclear weapons testing during past years. Beta activity in air, which shows no relationship with Plant operations, was about one-eighth of that for the same period of 1963. Radionuclides in milk were at the same levels as those reported for most sections of the United States by the U. S. Public Health Service. The concentration of radioactive materials in river algae indicated some slight contribution from the Plant. Radioactive materials in fish flesh continued to be far below levels considered significant from a health standpoint. In no instance was the concentration of radionuclides in river water in excess of 3.3% of the Maximum Permissible Concentration. (auth)

280 (HW-72229(Rev.)) DOSE-RATE MEASUREMENTS ON BEACHES AND ISLANDS ON THE COLUMBIA RIVER BETWEEN RINGOLD AND RICHLAND. McConnon, D. (General Electric Co., Richland, Wash. Hanford Atomic Products Operation). Aug. 10, 1962. Contract AT(45-1)-1350. 22p. Dep.(mn); $1.00(cy), 1(mn) CFSTI.

Radioactive materials released with the effluent of the Hanford production reactors can provide measurable radiation exposure to persons who use the Columbia River downstream. Dose rates measured on beaches in the Ringold to 300 Area section of the river were in the range of 0.05 to 1.1 mR/hr with maximum values near the water's edge. Dose rates in the water were in the range of 0.05 to 0.2 mR/hr. March and October values, obtained at a river flow of 50,000 cfs and 60,000 cfs, respectively, were about twice those observed in July at a river flow of 175,000 cfs. In general, the levels of contamination found in the Ringold to 300 Area section were not appreciably different from those which exist in the vicinity of Richland where the river is now used extensively for recreational purposes. (auth)

281 (HW-77609(Sect.3)) RADIOLOGICAL CHEMISTRY. (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). 106p.

282 (HW-78395) EVALUATION OF RADIOLOGICAL CONDITIONS IN THE VICINITY OF HANFORD, APRIL–JUNE 1963. R. H. Wilson, ed. (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Aug. 16, 1963. Contract AT(45-1)-1350. 39p.

Comprehensive surveillance of the occurrence of radioactive materials in air, water, and foodstuffs was continued in the vicinity of the Hanford plant. Gross beta activity in air during the second quarter of 1963 remained at levels persisting since September 1962 with the exception of a temporary small increase in May from increased deposition of world-wide fallout expected each spring. The average concentrations of I^{131} in milk from farms in the neighborhood of the Hanford plant was about 8 picocuries per liter, and the maximum in an individual sample was 84 picocuries per liter. This represents an increase from the first quarter which was probably due to increased grazing on open pasture. Natural background radiation, with some contribution from world-wide fallout continued to be the primary source of environmental exposure for most people in the Hanford environs. Cooling water from the reactors, which is discharged to the Columbia River, con-

tinued to be the low level waste with the greatest potential for contributing to the radiation exposure of persons in the plant environs. For the twelve month period ending June 1963, the estimated dose to the GI tract for persons drinking Pasco sanitary water was about 2% of the applicable limit recommended by the National Committee on Radiation Protection and Measurements (NCRP). Residents of Kennewick using Columbia River water received about one-third as much exposure. In May 1963, a fuel element failute released some fission products into the reactor cooling water and subsequently into the river that was distinguishable at Pasco as a single event. The potential dose to the thyroid of a small child consuming one liter of water during the period of maximum concentration was approximately 10 mrems. The dose to other organs was negligible. Radiophosphorus concentrations measured in whitefish caught between Ringold and Richland continued to be within the expected range based on previous years results. Locally caught fish continued to be the major source of radioisotopes for the few persons who eat them in quantity. Summing of exposures from various sources in a manner that tends to maximize the total dose to an individual leads to estimates of about 10% of the appropriate NCRP limit for the GI tract, about 15% of the FRC guide for the total body, 30% of the FRC guide for a child's thyroid, and about 30% of the NCRP maximum permissible rate of intake for bone seeking radionuclides. (auth)

283 (HW-79652) EVALUATION OF RADIOLOGICAL CONDITIONS IN THE VICINITY OF HANFORD, JULY–SEPTEMBER 1963. R. H. Wilson, ed. (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Nov. 15, 1963. Contract AT(45-1)-1350. 37p.

Comprehensive surveillance of the occurrence of radioactive materials in air, water, and foodstuffs was continued in the vicinity of the Hanford plant. Gross beta activity in air during the third quarter of 1963 decreased to levels observed prior to USSR weapon tests that started in August 1962. The concentration of I^{131} measured in milk from farms in the neighborhood of the Hanford plant during the third quarter was usually less than 10 pC/l; the maximum found in an individual sample was 136 pC/l which occurred in September following an unplanned release from one of the chemical separations plants. Natural background radiation, with some contribution from world-wide fallout continued to be the primary source of environmental exposure for most people living in the Hanford environs. The largest contribution of radioactive material from the Richland Operation continued to be the highly dilute low level waste in the reactor cooling water discharged to the Columbia River. For persons drinking Pasco sanitary water during the 12-month period ending September 1963, the estimated dose to the GI tract was about 2½% of the applicable limit recommended by the National Committee on Radiation Protection and Measurements (NCRP). Residents of Kennewick using Columbia River water received about one-third as much exposure. Sanitary water for Richland is now supplied by a pumping plant located on the Columbia River that started operation in June 1963. Because Richland's water supply was previously supplied by wells charged from the Yakima River, the estimated dose to the GI tract for the 12 months ending September 1963 was less than 1% of the applicable limit recommended by the NCRP. Radiophosphorus content of fish caught downstream from the reactors was within the expected range and probably constituted the major source of radioisotopes for the few people who eat them in quantity. Exposure from the various sources, summed in a manner that tends to maximize the total dose to an individual, were estimated to be about 10% of the Federal Radiation Council (FRC) guide for the total body, 30% of the FRC guide for a child's thyroid, and about 30% of the NCRP maximum permissible rate of intake for bone-seeking radionuclides.

284 (HW-81746(p.3.81-3)) SEDIMENT STUDIES: PARTICLE SIZE SEPARATION OF BOTTOM SEDIMENTS FROM THE McNARY RESERVOIR OF THE COLUMBIA RIVER. J. L. Nelson and L. L. Humphreys (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.).

A wet sieving particle sizing technique was used to size bottom sediments from the McNary Dam reservoir. The radionuclide contents of the size fractions are reported.

285 (KY–484) ENVIRONMENTAL MONITORING SUMMARY FOR THE PADUCAH PLANT FOR 1964. Baker, R. C.; Russell, A. W. (Paducah Gaseous Diffusion Plant, Ky.). July 1, 1965. Contract W-7405-eng-26. 12p. Dep(mn); $1.00(cy), 1(mn) CFSTI

Outdoor monitoring of air, water and vegetation in the vicinity of the Paducah Plant is summarized for the year of 1964. It was found that effluents from the Paducah Gaseous Diffusion Plant operations were channeled and controlled as necessary for the environmental concentrations of chemicals and radioactivity to beat acceptably low levels during the time period surveyed. (H.G.D.)

286 (MLM-1210) ENVIRONMENTAL MONITORING SEMI-ANNUAL REPORT: JANUARY THROUGH JUNE, 1964. P. C. Adams (Mound Lab., Miamisburg, Ohio). July 24, 1964. Contract AT(33-1)-Gen-53. 16p.

Data are presented on the concentration of radioactive materials detected in the environment surrounding Mound Laboratory, Miamisburg, Ohio, during the first half of 1964. The concentration of polonium and tritium in the Great Miami River was within the maximum permissible level recommended by the National Committee on Radiation Protection and Measurements. The concentration of polonium and plutonium in the environmental air was also within the maximum permissible level recommended by the National Committee on Radiation Protection and Measurements. There was no detectable concentration of tritium in the environmental air during this period. (auth)

287 (NLCO-933) RECENT BIO-ASSAY ACTIVITIES AT NATIONAL LEAD COMPANY OF OHIO. Michael W. Boback and Richard C. Heatherton (National Lead Co. of Ohio, Cincinnati). Sept. 28, 1964. Contract AT(30-1)-1156. 17p. (CONF-727-4). Dep.(mn); $1.00(cy), 1(mn) OTS.

From 10th Annual Bioassay and Analytical Chemistry Meeting, Cincinnati, Oct. 1964.

Some recent non-routine problems which have occupied the interest of the Bio-Assay Department, Health & Safety Division, National Lead Company of Ohio are discussed. These include the use of infrared and induction heating for the fusion step in the fluorometric determination of uranium; uranium in lung and lymph node tissues from unexposed persons; passivity of uranium slivers in hand injuries; radium isotopes found in residue from uranium remelt operations; investigations of the increase in plant radiation levels; uranium in fecal samples; and radionuclides in a plant effluent which results from precipitation run-off. (auth)

288 (ORNL-3721(Suppl. 1)) CONCENTRATIONS, TOTAL STREAM LOADS, AND MASS TRANSPORT OF RADIONUCLIDES IN THE CLINCH AND TENNESSEE RIVERS. Supplement No. 1 to Status Report No. 5 on Clinch River Study. Churchill, M. A.; Cragwall, J. S. Jr.; Andrew, R. W.; Jones, S. Leary (Oak Ridge National Lab., Tenn.). Aug. 1965. Contract W-7405-eng-26. 76p. Dep. mn; CFSTI $3.00 cy, $0.75 mn.

Progress is reported on collection and interpretation of information on radionuclides suspended or dissolved in the waters of the Clinch River downstream from Oak Ridge, Tennessee. Daily samples were taken at six stations between November 1960 and November 1962 and at one station from January 1962 to November 1962. Determinations were made of concentrations and total loads of ^{90}Sr, ^{137}Cs, ^{60}Co, and ^{106}Ru. Weekly average concentrations in pC/liter are reported for the seven stations. At only one station did the flow-weighted mean concentrations of ^{90}Sr and ^{106}Ru exceed MPC values for drinking water. Because of extreme difficulty in analyzing the gamma spectrum to identify the activity due strictly to ^{137}Cs when there is a high concentration of ^{106}Ru present, no great confidence could be placed in any of the ^{137}Cs data. Maximum concentrations of ^{60}Co found at all stations were far below MPC values. Data suggest that from 90 to 98 percent of the ^{90}Sr activity was associated with the dissolved solids, whereas the greater part of the ^{137}Cs activity was found in suspended solids. Seventy to 98 percent of the total ^{60}Co present in the water phase was actually in solution. In White Oak Creek and in the Clinch River, approximately 20 to 30 percent of the ^{60}Co present was associated with the suspended solids, but in the Tennessee River the percentage dropped to only 2 or 3 percent. This seems to indicate loss of sediment from the water phase in a downriver direction. From 79 to 94 percent of the ^{106}Ru activity was associated with the dissolved solids. Recommendations concerning the monitoring program and improvements in studies of this type are listed.

289 (PNROO-DEV-129) ENVIRONMENTAL RADIOACTIVITY AT THE SHIPPINGPORT ATOMIC POWER STATION FOR THE SECOND HALF OF 1963 AND FOR CALENDAR YEAR 1963. (Pittsburgh Naval Reactors Operations Office, AEC). 9p.

During the second half of 1963, measurements of radioactivity were made in the Ohio River, atmosphere, and fallout in the vicinity of the Shippingport Atomic Power Station. The results of these measurements support the conclusion that the radioactive waste released by the power station into the air and into the Ohio River did not result in a significant change in the naturally occurring radioactivity in the environment. (auth)

290 RADIOLOGICAL HEALTH DATA [MONTHLY REPORT], VOLUME IV, NUMBER 3. (Public Health Service, Washington, D. C.). Mar. 1963. 50p. $0.50(GPO) (RHD-Vol.IV, No.3)

Data are presented on fission product radioactivity in samples of airborne particulates, precipitation, and surface water collected at various locations throughout the U. S., Canada, and Mexico during 1962. Data are included on the daily consumption of milk, apples, lettuce, beef, and white bread by sex and region throughout the U. S.; the Sr^{90} content of the diet in selected areas; and the Sr^{90} activity in U. S. soil and wheat from 1958 through 1962. Results are included from a survey of Sr^{90} content in the diet in the United Kingdom from 1958 through 1961. Results re included from the analysis of milk samples collected throughout the U. S. and Canada during 1961 and 1962 for fission product activity with emphasis on the content of Sr^{89}, Sr^{90}, and I^{131}. Environmental levels of radioactivity during 1961 and 1962 are reported for the Feed Materials Production Center, Fernald, Ohio, and the Mound Laboratory, Miamisburg, Ohio. Data are summarized on nuclear detonations reported during Feb. 1963. (C.H.)

291 (TID-20200) PRELIMINARY REPORT ON THE SEDIMENTS AND RADIOACTIVITY IN THE VICINITY OF THE COLUMBIA RIVER EFFLUENT. Brown Bear Cruise 291, July 29–August 12, 1961 and Brown Bear Cruise 311, July 23–August 14, 1962. Technical Report No. 84. M. Grant Gross, Dean A. McManus, and Joe S. Creager (Washington. Univ., Seattle). Jan. 1963. Contracts AT(45-1)-1725 and Nonr-477(10). 35p. (AD-404020; M63-2)

The sediments in the area affected by the runoff from the Columbia River were studied to determine the distribution of sediment types and the presence and amount of radioactivity in the sediments. The study of the dispersal processes and distributional patterns of the sediment and its radioactivity is in the initial phase of a three-phase program. Various sedimentological analyses of 178 samples provided preliminary data on sediment distribution. The inner part of the continental shelf is covered with sand which grades into silt along the outer part of the shelf. The sediment of the abyssal plains is principally a silty clay, although local coarsening of the sediment is produced by the accumulation of microorganisms. The presence of quartz sand in samples from the abyssal plain appears to be restricted to those samples collected from deep-sea channels. Analyses of the gamma spectra of sediments collected near the Columbia River mouth in 1961 indicate that radionuclides, in addition to the naturally occurring K^{40}, are present in the sediments. The sandy sediments from Grays Harbor and the beaches adjacent to the Columbia River had very low activities although Cr^{51}, Zn^{65}, Co^{57}, and Co^{60} were detected. (auth)

292 (TID-21279) ENVIRONMENTAL RADIOACTIVITY AT ARGONNE NATIONAL LABORATORY, JANUARY–JUNE 1964. (Argonne National Lab., Ill.). Contract [W-31-109-eng-38]. 25p. Dep.(mn); $1.00(cy), 1(mn) OTS.

Data are presented on the level of radioactivity in the environment and on the amounts of radioactive materials, if any, added to the environment as a result of operations at Argonne National Laboratory. The radioactivity of the environment was determined on a continuing basis by measuring the radioactivity in naturally-occurring materials collected at selected locations both on and off the Argonne site. Air filter samples were collected continuously at seven locations on the Argonne site and at five locations off the site. The total alpha and beta activities are summarized.

293 (TID-22172) ENVIRONMENTAL CONCENTRATIONS OF RADIOACTIVE MATERIALS NEAR THE PADUCAH PLANT. Report for the First Half of 1965. (Paducah Gaseous Diffusion Plant, Ky.). Contract W-7405-eng-26. 9p. Dep. mn; CFSTI $1.00 cy, $0.50 mn.

Data summarizing the environmental concentrations of radioactive materials in air and water and the gamma radiation levels in the vicinity of the Paducah Gaseous Diffusion Plant are tabulated. Air monitoring was done by continuous filters, and water samples were taken weekly from the Big Bayou. Alpha counts from air sampling averaged 1.9% of the Radioactivity Concentration Guide (RCG) maximum, and mean beta counts were 0.04% of the RCG value. The average uranium content of water samples was 0.04% of the applicable RCG value for water beyond a controlled

area. Uranium content in water samples from the Ohio River below the plant averaged less than 0.01% of the RCG value. The concentration of beta emitters in the Big Bayou averaged 1% of the RCG value for the decay products of ^{238}U, and the beta activity of all samples from the Ohio River averages less than 3% of the RCG value.

294 CALIBRATION OF γ DETECTORS FOR ATMOSPHERE MONITORING IN $\mu C/cm^3$ BY MEANS OF A POINT RADIATION SOURCE. W. M. Pusch (Philips GmbH, Vienna). Atomkernenergie, 8: 410-14(Nov. 1963). (In German)

In the surroundings of a reactor plant, the specific concentration of the activity in the atmosphere has to be surveyed. The approximation of this natural or artificial activity is possible when every counting tube (ionization chamber) is sufficiently sensible. But there is the problem of calibrating these detectors, which means the determination of the connection between the contamination of the air (measured in $\mu C/cm^3$) and the expected counting rate (ionization current), which cannot be effected directly. A possible method for calibration of such tubes is described. The calculated flux, which is caused by a contamination of the atmosphere in the point of interest, is equalized with the flux being caused by a standard point source, having the same energy as the γ emitter in the atmosphere. This equation delivers the certain amount of specific activity concentration, for which the flux and the counting rate would be the same as for the point source. The counting rate of the point source, however, can be easily measured. So the counting rate to be expected for a certain activity is determined and furthermore, because of the proportionality of specific concentration and counting rate, for any other contamination, too. Absorption and scattering in the atmosphere were considered in this calculation, and also, in a simplified matter, the dependence of the detector sensitivity from the direction of the incident beam. (auth)

295 RADIOACTIVITY IN WASHINGTON SURFACE WATER, JULY 1962–JUNE 1963. Robert R. Mooney and Peter W. Hildebrandt. Radiol. Health Data, 5: No. 8, 386-9(Aug. 1964).

Data are presented on the radioactivity in river water collected at 71 stations located on 65 rivers throughout the State of Washington between July 1962 and June 1963. Analytical methods used are described. Samples from the Columbia River are γ scanned for ^{51}Cr, ^{106}Ru, ^{95}Zr, and ^{65}Zn and samples from other rivers are analyzed for total β activity of dissolved and suspended fractions. (C.H.)

296 DESIGN AND DEVELOPMENT OF MARINE MONITORING PROGRAMS. F. Morgan (Ministry of Agriculture and Fisheries, Lowestoft, Eng.). Food Agri. Organ. U. N., FAO At. Energy Ser., No. 5, 239-44(1964).

Radioactive wastes can generally be disposed of at sea by conventional methods provided there is no return of the radioactivity to man. The paths by which the radioactivity is returned to man depend on the composition of the waste, the character of the water mass, the species of marine organisms present, and the habits of the human population concerned. These factors vary from site to site and a preliminary site survey must be carried out before any license for marine waste disposal is issued. Procedures are outlined for such site surveys and monitoring during the experimental period. (C.H.)

297 RADIOLOGICAL SURVEY OF THE LOWER COLUMBIA RIVER IN OREGON, JANUARY 1962–JULY 1963. George L. Toombs and John G. Bailey. Radiol. Health Data, 5: No. 8, 395-400(Aug. 1964).

Results are reported from a radiological survey of the lower Columbia River in Oregon between Jan. 1962 and July 1963. The contribution of cooling waters from Hanford reactors and fallout to the radioactivity of the Columbia River are discussed. Data are included on the radioactivity of river water, sediments, and biological organisms. (C.H.)

298 RADIOACTIVITY OF THE COLUMBIA RIVER EFFLUENT. Gross, M. Grant; (Univ. of Washington, Seattle); Barnes, Clifford A.; Riel, Gordon K. Science, 149: 1088-90(Sept. 3, 1965).

Chromium-51 and zinc-65 were detected as far as 115 and 15 kilometers, respectively, from the mouth of the Columbia River; zinc-65 also was found at a few isolated stations. Zirconium-95 and niobium-95 from atmospheric fallout occurred in the river effluent but they were most abundant in surface waters further offshore. Distribution of the radionuclides is controlled largely by surface currents and by upwelling of sea water near the coast.

299 NEW CHALLENGES IN ENVIRONMENTAL HEALTH CONTROL. Gilbertson, W. E. (Public Health Service, Washington, D. C.). Trans. Coll. Physicians Philadelphia, 31: 294-8(Apr. 1964).

Problems in control of atmospheric radionuclides and other air pollutants are considered.

Ecological Studies and Radioisotope Uptake

See also the following references:

22, 23, 175, 183, 210, 269, 297, 350, 511, 615, 829, 834, 838, 840, 844, 850, 852, 854, 921, 927

300 (A/CONF.28/P/849) HAZARD EVALUATION OF RADIOACTIVE CONTAMINATION OF SEA AND DECONTAMINATION OF RADIOACTIVITY IN FRESH WATER. Yoshio Hiyama (Tokyo Univ. Fisheries Inst.), Masamichi Saiki, Hiroshi Kamada, Taku Koyanagi, Yasuo Kurosawa, and Y. Hiyama. May 1964. 15p.

The experimental analysis of the quantitative relation in the levels of radioactive contamination between sea water and organisms on the elements thought to be important in the waste disposal problem, such as Cs, Sr, I, Co, Zn, Cd, Fe, Ru, and Ce, is discussed. As a result of numerous rearing experiments it was possible to know the level of the concentration of a certain radionuclide in a certain tissue of an organism by knowing the concentration of it in the sea water and how long it was in that environment. The concentration factors, a ratio of the concentration of an element in a tissue of an organism to that of sea water, were obtained during the rearing experiments. The equilibrium values of the concentration factor obtained by this series of experiments were compared with those which were obtained by chemical analysis of the stable and radioactive elements by previous workers. A way to control and to monitor the radioactive contamination of the sea is described. Studies based on experiments on the relation of radioactivity among water sources (river water, etc.), service water (drinking, etc.) and sewage, environmental behavior of radionuclides,

and the various aspects of radiological decontamination are also discussed. Behavior and concentrations of ^{137}Cs and ^{90}Sr in fresh water in Japan, the relation of radioactivity between air and sewage, and radioactivity decontamination materials for drinking water are discussed. (C.E.S.)

301 (A/CONF.28/P/870) SANITARY PROBLEMS RAISED BY THE ELIMINATION OF RADIOACTIVE WASTE AND BY NUCLEAR ACCIDENTS. Henri Jammet, David Mechali, and Marc Dousset (France. Commissariat a l'Energie Atomique, Paris). May 1964. 14p.

To be able to evaluate the risks run by the population and to fix tolerance limits for waste discharge, it is necessary to possess an exact knowledge of the course taken by the radioelements discharged, from their source until they reach man. The incorporation of radioelements in food cycles is the first risk to take into consideration. The factors involved in all stages of this transmission must be foreseen, whether they concern physical or biological media, because of their continuity and their interdependence. Finally, socio-economic and dietetic data must be collected in order that the risks and tolerance levels estimated are based on concrete and experimental rather than theoretical knowledge. The risk of nuclear accidents in the atomic industry, although very improbable, must be taken into consideration because of the seriousness of their consequences. The health problems arise in the field of professional hygiene on the one hand and in that of public hygiene on the other. In the first field, the risk is two-fold and involves irradiation and contamination. The public sphere is reduced essentially to the risk of contamination by radioactive substances accidentally released in the surrounding medium. The health studies to be conducted in this field therefore include research not only on irradiation or contamination therapeutics but also on the transfer of radioelements from the accident site to man, mainly through food cycles, in their physical and in their biological components. Studies of this kind form the basis of decisions in the health field which would have to be taken in the case of an accident. (auth)

302 (BNWL-122, pp 187-9) RADIONUCLIDE ACCUMULATION AND TRANSPORT BY PLANKTON IN THE COLUMBIA RIVER. Cushing, C. E. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.).

A study was made of the accumulation and transport of ^{32}P and ^{65}Zn by net plankton in the Columbia River below the Hanford reactors. Net plankton, as used in this study, is defined as the free floating organic matter retained by a No. 20 mesh plankton net (68 meshes/cm) and includes both living and dead material. Cursory analysis of monthly averages of net plankton samples revealed that ^{32}P and ^{65}Zn concentration (nCi/g dry weight) was lowest during the summer months, probably related to dilution by runoff. Transport of these radionuclides by the plankton was highest in spring and late summer, coincident with seasonal increases of the plankton. The ability of the plankton to concentrate radioisotopes from the water varied seasonally with lowest values during the summer months.

303 (BNWL-122, pp 190-3) RADIONUCLIDE ACCUMULATION AND PRODUCTIVITY OF PERIPHYTON IN THE COLUMBIA RIVER. Cushing, C. E. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.).

The periphyton and the phytoplankton are important communities in the Columbia River in relation to radionuclide accumulation. Results are reported from a study of biological and environmental factors that are most influential in governing the amount of ^{32}P and ^{65}Zn accumulated from the river by the periphyton, and a comparison of the relationships among the commonly used production indices and certain environmental measurements. Data on the method of sorption of these nuclides were also evaluated. Best correlations among four biomass measurements of periphyton were between dry weight, ash weight, and chlorophyll a. Net production rate varied from 0.005 to 0.070 mg dry weight/cm^2/day and was closely related to chlorophyll a content and solar energy. The accumulation of ^{32}P and ^{65}Zn was best related to dry and ash weight, and chlorophyll a. Low correlations were found between radionuclide accumulation and the radionuclide content of the river. The data suggested that adsorption was the dominant mode of uptake.

304 (BNWL-SA-53) EVALUATION OF EXPOSURE FROM MULTIPLE ENVIRONMENTAL SOURCES IN THE ENVIRONS OF THE HANFORD PLANTS. Foster, R. F.; Wilson, R. H. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). [June 8, 1965]. Contract AT(45-1)-1830. 21p. (CONF-650616-4). Dep. (mn); $1.00(cy), 1(mn) CFSTI.

From Health Physics Society, Annual Meeting, Los Angeles.

The people that live near the Hanford Atomic Energy site can acquire radioactive materials by several different environmental pathways that begin with the release of reactor effluent to the Columbia River, the release of stack gases to the atmosphere, and world-wide fallout. In 1964 the radiation dose received by the vast majority of the local residents, from all of the sources combined did not exceed ten per cent of the appropriate limits. Except for the dose to the GI tract from drinking water, the exposure resulting from world-wide fallout was at least as great as that which could be attributed to the Hanford Plants. For the relatively few people who eat large numbers of fish caught from the Columbia River immediately below the Hanford reservation, ^{32}P is an important source of exposure. In 1964 the hypothetical individual who had the maximum credible intake of ^{32}P and other nuclides of Hanford origin, could have taken in about 25 per cent of the permissible annual intake of bone-seeking radioisotopes. (auth)

305 SYMPOSIUM ON RADIATION AND TERRESTRIAL ECOSYSTEMS. Hungate, F. P. Nucl. Safety, 7: 92-9; 102(Fall 1965). (BNWL-SA-219).

A symposium on Radiation and Terrestrial Ecosystems was held May 3–5, 1965, in Richland, Wash. The symposium consisted basically of three parts: sources of radionuclides and how these radionuclides enter a food chain through plants; ecological paths and rates of nuclide transfer; and radiation as an environmental factor. Tracers of all kinds were utilized in the studies, those of natural origin, those widely dispersed as fallout, those intentionally introduced into the system, and even nonradioactive simple mineral constituents. It was pointed out that radiation produces both stimulatory and inhibitory effects, depending on levels used and features of the system observed.

306 (BNWL-SA-293) UPTAKE AND RETENTION OF ^{65}Zn FROM CERTAIN FOODS. Honstead, J. F.; Hildebrandt, P. W. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). Sept. 13, 1965. Contract AT(45-1)-1830. 9p. (CONF-651014-1). Dep. mn; CFSTI $1.00 cy, $0.50 mn.

From Western Industrial Health Conference, San Francisco.

The uptake and retention of ^{65}Zn from the consumption of oysters collected on the Washington coast near the mouth of the Columbia River, was measured by whole-body counting of a man. The subject was counted in a whole-body counter to establish an initial background count. During the next 20 mo. he ate 81 meals of oysters, an average of about one meal per week. The ^{65}Zn content of each entire oyster

meal was determined. A total of 13 whole-body counts were made during the period. The total ^{65}Zn ingested by the subject was 709 nanocuries included in 17,430 g of oysters. The maximum ^{65}Zn body burden attained by the subject was 40 nanocuries. It was concluded that the ^{65}Zn content in Pacific oysters does not contribute significantly to the radiation exposure of members of the local population.

307 (BNWL-SA-623) ACCUMULATION AND TRANSPORT OF RADIONUCLIDES BY COLUMBIA RIVER BIOTA. Cushing, C. E.; Watson, D. G. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). Mar. 31, 1966. Contract AT(45-1)-1830. 24p. (CONF-660507-5). Dep. mn. CFSTI $1.00 cy, $0.50 mn.

From Symposium for the Disposal of Radioactive Wastes into Sea, Oceans, and Surface Waters, Vienna.

Data from investigation of the uptake and transport of radionuclides by periphyton, plankton, fish, and selected invertebrates in the Columbia River are presented. A preliminary study of the periphyton was made to determine which biological and environmental factors are most influential in governing the accumulation of ^{32}P and ^{65}Zn from the river and to investigate the relationships between some commonly used indices of production and certain environmental factors. Phosphorus-32 concentrations in Stigeoclonium fluctuated markedly with no apparent trend. Solar radiation had a high correlation with net production rate (NPR) and chlorophyll a. It is suggested that the high correlation between NPR and chlorophyll is consistent with the fact that closest relationships between pigment content and productivity are usually found only during periods of rapid growth. Accumulation and transport of ^{32}P and ^{65}Zn by net plankton was determined in the Columbia River below the Hanford reactors. Reactor effluent contribution to the river remained relatively constant. In general, despite lesser concentrations of radionuclides per unit weight in spring and late summer, more ^{32}P and ^{65}Zn are transported by the plankton at those times due to the increased biomass. In late fall of 1964 and 1965 a wide variety of invertebrates were collected and analyzed for 18 different radioisotopes by multidimensional gamma-ray spectrometry. Samples were collected for the purpose of evaluating the effect of shutting down three production reactors during the first half of 1965. Except for October levels of ^{32}P, lower concentrations of ^{32}P, ^{51}Cr, and ^{65}Zn (representing 90% of the total gamma activity) were present in lower quantities in 1965. Zinc-65 measurements in whitefish were used to define the extent and time of resident fish movement upstream from the Hanford reactor effluent outfalls and in a tributary stream.

308 (CEA-Bib-53) CONTAMINATION DIRECTE DES VEGETAUX ET IRRIGATION PAR ASPERSION: ETUDE BIBLIOGRAPHIQUE. (Direct Contamination of Plants and Irrigation by Spraying. Literature Survey up to April 1st, 1964). Bittel, Robert; Clement, Jean-Louis (Commissariat à l'Énergie Atomique, Saclay (France). Centre d'Études Nucléaires). Apr. 1965. 83p. Dep.(mn).

Data are summarized from a literature survey concerning fundamental theoretical aspects of radio-element absorption by the aerial parts of plants (leaves, flowers and fruits, base of the plants); the transfers in the plant; and the various factors governing absorption and transfers (physical, physico-chemical and biological factors). An attempt was made to evaluate radio-contamination risks which may result from irrigation by spraying with water contaminated by radioactive wastes. (157 references). (C.H.)

309 MARINE ENVIRONMENTAL MONITORING IN THE VICINITY OF WINDSCALE. H. Longley and W. L. Templeton (United Kingdom Atomic Energy Authority, Windscale, Cumb, Eng.). 38p. (CONF-365-18)

From Symposium on Environmental Monitoring, Berkeley, Gloucestershire, England, Oct. 1963.

The extensive environmental monitoring during the last ten years has shown that there has been no systematic increase in the radioactivity levels of concern to man due to the discharges of radioactive liquid waste to the Irish Sea from Windscale. The radiation doses of people who may eat seaweed or fish collected in the area, or who fish in the sea, or bathe from the beaches, can never have reached the maximum permissible values. The ruthenium-106 content of edible seaweed is the most restrictive factor which limits the discharges, but accumulation by the edible seaweed also restricts the permissible discharges of other radionuclides. From the monitoring data for edible seaweed, it is possible to deduce values for the discharges which may produce the derived maximum permissible concentrations on the seaweed, although the accuracy of these estimates will be poor for discharge rates of less than about 500 curies per month. The extended survey traced the effluent both northwards and southwards from Sellafield towards the Mull of Galloway in the north and towards Anglesey in the south. The concentration of β activity in bottom marine organisms decreases very rapidly in the first 20 miles northwards and southwards from Windscale. Since the maximum permissible concentrations in seaweed, fish, and sea bottom material are not exceeded at Sellafield, no radiological hazard to man is likely to occur in any other part of the north eastern Irish Sea. (auth)

310 THE PRE-OPERATIONAL DISTRICT SURVEY AT HUNTERSTON. G. Jackson (Hunterston Nuclear Generating Station, Eng.). 18p. (CONF-365-22)

From Symposium on Environmental Monitoring, Berkeley, Gloucestershire, England, Oct. 1963.

Results are given of the pre-operational district survey at Hunterston which was conducted to determine the activity levels in foodstuffs and other materials prior to operation. The results include Sr^{90}, Sr^{89}, Cs^{137}, and I^{131} in milk, herbage, root mat, soils and reservior water, and gamma surveys. These measurements were made over a period of time of a year before operation, but it is felt that single measurements taken immediately prior to reactor commissioning would have sufficed. (D.L.C.)

311 THE G.E.C. DISTRICT SURVEY OF THE HUNTERSTON AREA. D. J. Mathias. 11p. (CONF-365-23)

From Symposium on Environmental Monitoring, Berkeley, Gloucestershire, England, Oct. 1963.

A district survey of the Hunterston area was conducted from Feb. 1959 to Nov. 1961, and it included gamma background and radioactivity of biological samples. The summarized results include the beta activities of rushes, grasses, seaweed, milk, and fish. Rushes were found to concentrate fission product fallout in rainfall to a high degree. It is felt that a routine district survey is not necessary to the safe operation of a nuclear power station.

312 (EUR-2407.f) ASPECTS SCIENTIFIQUES ET SANITAIRES DE LA CONTAMINATION RADIOACTIVE DE LA MER. (Scientific and Health Aspects of Marine Radioactive Contamination). Recht, P.; Van Hoeck, F. (European Atomic Energy Community). 1965. 59p. (CONF-640530-1). Dep. mn.

From International Congress on the Sea, Saint-Cast, France.

Scientific and health aspects of the marine contamination resulting from the disposal of radioactive waste at sea are reported, and factors to be considered from the health and biology standpoint are surveyed. The physical and chemical components of the marine environment are reviewed. The irradiation doses received from natural background radioactivity are specified. Particular attention is paid to the biological factors that must be taken into account in evaluating the degree of radioactive contamination. The artificial radioactivity found in the marine environment stems mainly from radioactive fallout but also, within limited areas, from discharged radioactive waste. The problem of radioactive waste is discussed, with a summary of the data concerning its origin, composition and classification. At present such waste is disposed of in three ways: as liquid effluents discharged into coastal waters; in submerged drums; and as liquid and solid effluents from nuclear-powered ships. Difficulties encountered in experimental biological work in a marine environment are discussed. A sample research programme is presented. (auth)

313 (HW-79279) PRELIMINARY MEASUREMENTS OF THE ADSORPTION OF REACTOR EFFLUENT WATER RADIONUCLIDES BY COLUMBIA RIVER PARTICULATE MATERIAL. L. L. Humphreys and R. W. Perkins (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Oct. 16, 1963. Contract AT(45-1)-1350. 10p.

Laboratory experiments showed that dilution of reactor effluent water ten fold with river water resulted in a rapid adsorption of several of the soluble radionuclides on the river water particulates. Between 25 and 40% of the soluble radionuclides adsorbed during a 24-hr period were adsorbed during the first minute of contact. Data are presented on the adsorption of ^{56}Mn, ^{24}Na, ^{239}Np, ^{140}La, ^{64}Cu, ^{51}Cr, ^{46}Sc, and ^{65}Zn by the inorganic and organic detritus and microorganisms in Columbia River water. (C.H.)

314 (HW-80500(p.205-8)) SWIMMING PERFORMANCE OF CHINOOK SALMON REARED IN REACTOR EFFLUENT. [PART] II. R. E. Nakatani (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.).

Swimming performance of chinook salmon at three different water velocities after being reared for 17 weeks in 3 and 6% reactor effluent with a corrosion inhibitor additive of either sodium dichromate or INHIB-7 showed no difference due to effluent concentration or type of additive. The size of fish (51 to 55 mm and 56 to 60 mm) tested emphasized the importance of size by the obviously better performance of the longer fish. The question is raised of whether within a particular size stratum the runts are poorer swimmers than fish representing the average size of a lot. (auth)

315 (IDO-12035) CONTROLLED ENVIRONMENTAL RADIOIODINE TESTS AT THE NATIONAL REACTOR TESTING STATION. C. A. Hawley, Jr., C. W. Sill, G. L. Voelz (Idaho Operations Office. Health and Safety Div., AEC), and N. F. Islitzer (Weather Bureau, Idaho Falls, Idaho). June 1964. 73p. Dep.; $3.00(cy), 2(mn) OTS.

The CERT project consists of a series of planned releases of radioiodine over different vegetation and during various meteorological conditions, with the prime objective being to measure the amounts of radioiodine through the air-vegetation-cow-milk-human chain. Results are reported from a preliminary experiment at the National Reactor Testing Station (NRTS) in southeast Idaho that consisted basically of releasing radioiodine (iodine-131) gas over a natural Crested Wheatgrass pasture and using the contaminated grass for milk cow grazing. The resultant radioactive milk was fed to seven human volunteers. It was desired from this experiment to determine if the experimental design would establish, under known natural release conditions, the basic relationships between the amount of radioiodine in the air to that on the soil and vegetation, the amount of radioiodine on the vegetation to that in the milk, and the quantity in the milk to that in the human thyroid after drinking the milk. A total of 970 millicuries of iodine-131 gas was released at 1500 MST on May 27, 1963, near ground level over a 30-minute period under moderately unstable meteorological conditions and an average wind speed of 6.6 meters per second. About 13 percent of the total released iodine was deposited on the sampling grid, with 1.5 percent being actually on grass. The Crested Wheatgrass covered about 15 percent of the total plot, the remaining surface being soil. Deposition velocities ranged from 0.4 to 0.8 cm/sec with an average of 0.6 cm/sec. The activity on carbon fallout plates was found to be proportional to the grass measurements. Controlled grazing, which consisted of daily changes of 2×10^3 m^2 crosswind grazing strips, progressed from 300 meters downwind, toward the source, permitting quantitative measurement of grass consumption and activity. The effective half-life on grass was found to be about 3.5 days. The effective cow consumption measurement, which represents the accumulation of activity within the cow from the current and previous days' grazing, based on a one day half-life retention factor, and the measured activity in the milk showed good correlation. The ratio of activities of milk and grass (pCi/l : pCi/g) was 240 ± 35. The average human thyroid uptake of ingested iodine-131 in milk was 19 percent. A model to predict thyroid activity levels was developed and shown to be accurate. Thyroid doses to the volunteers averaged 0.39 rad. The preliminary experiment showed that the basic experimental procedures were adequate. Further tests in the series will employ the same general procedures in investigating the behavior of radioiodine under various meteorological and physical conditions.

316 (JPRS-21816) DISTRIBUTION OF RADIOISOTOPES IN BASIC COMPONENTS OF FRESH-WATER BODIES. Elena Aleksandrovna Timofeeva-Resovskaya. Translated from Tr. Inst. Biol., Akad. Nauk SSSR, Ural'sk. Filial, No. 30, 1-79(1963). 104p. (OTS-63-41112)

Results are reported from a series of studies of the distribution of radioactivity from 18 different radioisotopes in surface waters, sediments, and aquatic organisms. Emphasis was placed on the concentration of the various radioisotopes by freshwater organisms. The radioisotopes investigated included P^{32}, S^{35}, Cr^{51}, Fe^{59}, Co^{60}, Zn^{65}, Ge^{71}, Rb^{86}, Sr^{90}, Y^{91}, Zr^{95}, Nb^{95}, Ru^{106}, Cd^{115}, I^{131}, Co^{137}, Ce^{144}, and Hg^{203}. (187 references.) (C.H.)

317 (NP-14161) SURVEY OF INTERSTATE POLLUTION OF THE ANIMAS RIVER (COLORADO-NEW MEXICO). E. C. Tsivoglou, S. D. Shearer, R. M. Shaw, Jr., J. D. Jones, J. B. Anderson, C. E. Sponagle, and D. A. Clark (Robert A. Taft Sanitary Engineering Center, Cincinnati). May 1959. 123p.

The Animas River from Durango, Colorado, to Farmington, New Mexico, a distance of some 60 river miles, serves as a source of water supply for several communities, is used in the raw state as a water supply for some 2,000 persons, is used for recreational purposes, including swimming and fishing, provides water for the irrigation of some 26,000 acres of cropland, provides water for livestock, is the source of industrial process water for a uranium ore refinery, and carries away the wastes from this mill as well

as wastes from the city of Durango and other small sources. Data are presented from studies of pollution problems of the River. Measurements were made of the extent of radioactive, chemical, organic, and bacteriological pollution, and the effects of pollution on existing legitimate uses of the interstate waters. The internal radiation exposure due to Ra and ^{90}Sr was evaluated in the Durango-Farmington population area, and an estimate was made of the portion of this exposure that resulted from Ra and other waste discharges from the Vanadium Corp. of America uranium refinery at Durango, Colo. (C.H.)

318 (NP-15159) STUDIES ON UPTAKE OF RADIOISOTOPES BY EDIBLE MARINE PRODUCTS. Progress Reports, 1960-1962. Hiyama, Yoshio (Tokyo Univ. (Japan)). IAEA Contract No. 20R/RB. 51p. Dep. mn.

The concentration factors, defined as ratio of concentration of radionuclide in tissue or organ of marine organisms to that in water where these are living, of several radionuclides (^{131}I, ^{60}Co, ^{106}Ru, ^{144}Ce, ^{65}Zn, ^{115m}Cd, ^{134}Cs, and ^{89}Sr) were determined for representative edible marine organisms including fish, shell fish, crustacea, sea-urchins, and algae. Rate of up-take, or time to reach equilibrium, were found to vary according to the kind of radionuclide, kind of marine organisms, age of the organism, and temperature of the water. Concentration factors in equilbrium were found to vary according to the kind of tissue or organ of organism, the kind of organisms, and the kind of radionuclide, but were not affected by water temperature and the concentration of radionuclide in water. Existence of carrier in sea water reduced the concentration factors in some degree for ^{131}I, ^{60}Co, ^{65}Zn, and ^{115m}Cd, but not for ^{134}Cs and ^{89}Sr. In case of Fe uptake by seaweeds, the existence of light affected the concentration factor. The forms of radionuclides in sea-water affected the up-take, especially the particulate or colloid form. Difference of concentration factor obtained by rearing experiment in equilibrium from that by ingestion or injection experiment is not remarkable for the nuclides tested. The destination of radionuclides (critical organs) in the body is similar in any method. It is concluded that the concentration factors of edible marine organisms for radionuclides are the keys to the evaluation of hazards caused by radioactive contamination of sea-water.

319 (NP-15916) ANNUAL REPORT ON RESEARCH CONTRACT No. 201/R1/RB, JULY 1964–JUNE 1965. (Institut "Rudjer Boskovic," Zagreb (Yugoslavia)). 104p. Dep. mn.

Progress is reported in investigations on the radioecology of the Bay of Venice. Field observations on several permanent hydrographic stations were made. Besides the common meteorological, physical, and chemical parameters for the whole vertical column of water and above it, investigations concerning the quality and quantity of the phytoplankton were made. These results were completed with the in situ and laboratory determination of the photosynthetic rate for phytoplankton samples. The physico-chemical behavior of some microconstituents in sea water was investigated using electromigration techniques and studying precipitation phenomena. The bioaccumulation of various radioelements in some characteristic animals and natural populations of phytoplankton was studied in laboratory experiments. In these experiments some radionuclides which can play a certain role as potential hazard in controlled and uncontrolled waste discharge from nuclear plants or weapons were used, including $^{56,57,58}Co$, ^{54}Mn, ^{65}Zn, ^{106}Ru. Shrimps (Leander sp.) and mussels (Mytilus galloprovincialis Lam.) were chosen as experimental animals because they can serve as an ideal indicator for the contamination of coastal waters. Besides the uptake and loss rate under more or less natural conditions, the mechanism of the exchange was studied using EDTA as modifier. Some interesting results were obtained when the influence of the increased iron, cobalt, and zinc concentration on the uptake of radiozinc and radiocobalt was followed.

320 (ORNL-3530) GROWTH AND MOVEMENT OF SMALLMOUTH BUFFALO, ICTIOBUS BUBALUS (RAFINESQUE), IN WATTS BAR RESERVOIR, TENNESSEE. R. E. Martin, S. I. Auerbach, and D. J. Nelson (Oak Ridge National Lab., Tenn.). Jan. 20, 1964. Contract W-7405-eng-26. 104p.

The smallmouth buffalo fish, Ictiobus bubalus (Rafinesque), population of Watts Bar Reservoir, of the Tennessee River down stream from Oak Ridge National Laboratory, was investigated in order to describe its age distribution, growth rates, dispersion, and importance as an accumulator of radionuclides. Measurements and scale samples were taken from commercially-caught fish and fish caught in the ORNL tagging operations. Scale impressions were analyzed for age and growth phenomena. Dispersion of smallmouth buffalo was investigated by conventional tagging methods and by autoradiographic analyses of scales. Stable and radiochemical composition of scales was examined by spectrographic analysis, flame spectrophotometry, and radiometric surveys. Calcium was the most abundant element in fish scales with at lease twenty-three other elements present in varying quantities. Fish scales and bone were found to contain radionuclides of ruthenium, cesium, zirconium, zinc, and cobalt. Radiometric surveys of scales revealed the Watts Bar Reservoir smallmouth buffalo population was a relatively minor accumulator of radionuclides with only 0.08 per cent showing the presence of artificially produced radionuclides. Approximately 6 per cent of the Clinch River fish and 77 per cent of the White Oak Creek fish had accumulations. Limited data on dispersion were determined from conventional tags. Much more dispersion and life history data were determined from autoradiographic analyses of scales. These dispersion data were applied only to individuals because the number was too small for generalizations for the population as a whole. All normal scales containing radionuclide accumulations were found to produce identical autoradiographic patterns of concentric circles which were associated with growth of the fish in contaminated areas. This phenomenon was combined with conventional capture-recapture methods of population estimates in a proposed technique of population studies. A laboratory experiment showed that scales could be tagged with cesium-134, but this radionuclide was found to accumulate in much larger concentrations in the soft tissues than in the bony tissues. (C.H.)

321 (ORNL-3531) CHROMOSOMAL ABERRATIONS IN A NATURAL POPULATION OF CHIRONOMUS TENTANS EXPOSED TO CHRONIC LOW-LEVEL ENVIRONMENTAL RADIATION. B. G. Blaylock, S. I. Auerbach, and D. J. Nelson (Oak Ridge National Lab., Tenn.). Jan. 29, 1964. Contract W-7405-eng-26. 88p.

Thesis submitted to Univ. of Tennessee, Knoxville.

The salivary gland chromosomes of Chironomus tentans larvae collected from White Oak Creek, an area contaminated by radioactive waste from the Oak Ridge National Laboratory, and from six uncontaminated areas were examined for chromosomal aberrations. White Oak Creek populations were exposed to absorbed doses as high as 230 rads per year or about 1000 times background. Chromosomal maps were constructed to make a general comparison of the banding pattern of the salivary chromosomes of the C. tentans in the East Tennessee area with those of

Canada and Europe. These maps were used as a reference in scoring aberrations. Fifteen different chromosomal aberrations were found in 365 larvae taken from the irradiated population as compared with five different aberrations observed in 356 larvae from six control populations, but the mean number of aberrations per larva did not differ in any of the populations. The quantitative amount of heterozygosity was essentially the same in the irradiated and the control population, but there were three times the variety of chromosomal aberrations found in the irradiated area. From this evidence it was concluded that chronic low-level irradiation from radioactive waste was increasing the variability of chromosomal aberrations without significantly increasing the frequency. It was also concluded that chromosomal polymorphism can be maintained in a natural population without superiority of the heterozygous individuals. (C.H.)

322 (ORNL-3849, pp 43-100) RADIATION ECOLOGY. Auerbach, S. I. (Oak Ridge National Lab., Tenn.).

Results of investigations concerning radiation effects on insects, plants, and animals in reactor and waste areas are summarized. Investigation results are also summarized on processes and components of terrestrial ecosystems. Clinch River and related aquatic studies are reported along with theoretical studies of food chains and systems ecology.

323 (ORNL-P-2021) EVALUATION OF RADIATION DOSE TO MAN FROM RADIONUCLIDES RELEASED TO THE CLINCH RIVER. Cowser, K. E.; Snyder, W. S.; McCammon, C. P.; Straub, C. P.; Kochtitzky, O. W.; Hervin, R. L.; Struxness, E. G.; Morton, R. J. (Oak Ridge National Lab., Tenn.). [1965]. Contract W-7405-eng-26. 35p. (SM-72/41; CONF-660507-4). Dep. mn. CFSTI $2.00 cy, $0.50 mn.

From Symposium for the Disposal of Radioactive Wastes into Sea, Oceans, and Surface Waters, Vienna.

Low-level contaminated waste water, generated by the operations at Oak Ridge National Laboratory, has been released to the Clinch River near Oak Ridge for the past 20 years under controls consistent with the recommendations of ICRP and NCRP. Results are reported from an evaluation of radiation dose equivalents to populations downstream, made on the basis of long-term monitoring data and the identification of critical exposure pathways and critical population groups. A study of water utilization downstream indicated that the important avenues of exposure resulting from discharge of radioactive fluids to the Clinch River include the consumption of contaminated water and fish, the consumption of agricultural produce that may be irrigated with river water, exposure to contaminated water and bottom sediments during recreational and industrial use of the water, and exposure to build-up of radionuclides in sludge and deposits in water systems utilizing river water. The major sources of exposure were due to the ingestion of contaminated water and fish. Mathematical models were developed for internal dose calculations which include the differences in intake and size of the critical organ as a function of the individual's age. (26 references.) (C.H.)

324 (TID-19562) THE POPULATION STATUS OF THE LARGER VERTEBRATES ON THE ATOMIC ENERGY COMMISSION SAVANNAH RIVER PLANT SITE. Final Report. James H. Jenkins and Ernest E. Provost (Georgia. Univ., Athens). 1963. Contract AT(38-1)-224. 49p.

Results are reported from a survey on the population status of large vertebrates at the Savannah River Plant site. The effects of forest management practices on the wildlife population are discussed briefly. (C.H.)

325 (TID-19618(App.A) PLANT TAXA ON A THREE ACRE PORTION OF WHITE OAK LAKE BED, OAK RIDGE, TENNESSEE, 1956-1960. (Tennessee. Univ., Knoxville). 7p.

A list of plants found on a 3-acre portion of the drained White Oak Lake bed between 1956 and 1960 is presented. (C.H.)

326 (TID-21713) ONSITE ECOLOGICAL RESEARCH OF THE DIVISION OF BIOLOGY AND MEDICINE AT THE SAVANNAH RIVER ECOLOGY LABORATORY. Golley, Frank B.; Loyal, Ann H. (eds.) (Savannah River Ecology Lab., Aiken, S. C. and Georgia Univ., Athens. Inst. of Radiation Ecology). Jan. 1965. 103p. Dep.(mn); $4.00(cy), 2(mn) CFSTI.

Studies of the ecosystems on the 250,000-acre reservation of the Savannah River Plant were initiated in 1952 to provide basic environmental information that would be helpful in evaluating changes that might result from the operation of the Plant. The dominant types of ecosystems were identified and an inventory was made of plant and animal species found in them, the numbers of organisms, and their weight or biomass. Successive studies of succession, productivity, and bioenergetics have been conducted. Research studies in ecology carried out since 1962 are described. Emphasis was placed on studies of the structure and functions of populations of an old-field ecosystem; the effects of γ radiation on ecological systems; and the use of radionuclides to measure ecological function in small mammals, cotton rats, and bobcat populations; and the use of phosphorus-32 to isolate food chains in intact ecosystems.

327 (UWFL-86) RADIONUCLIDES OF COLUMBIA RIVER ORIGIN IN MARINE ORGANISMS, SEDIMENTS, AND WATER COLLECTED FROM THE COASTAL AND OFFSHORE WATERS OF WASHINGTON AND OREGON, 1961-1963. Allyn H. Seymour and Gary B. Lewis (Washington Univ., Seattle. Lab. of Radiation Biology). Dec. 10, 1964. Contract AT(45-1)-1385. 78p. Dep.; $3.00 (CFSTI).

A program was conducted to determine the distribution of Hanford-produced radionuclides in the marine environment. Zinc-65 measurements were made in coastal organisms, offshore plankton, demersal fish, and water collected in surveys in 1961 through 1963. The ratio of ^{51}Cr to ^{65}Zn in water samples was determined. Annual, seasonal, and areal differences were studied. Other radionuclides also determined included ^{32}P, fallout radionuclides, and ^{40}K.

328 UPTAKE AND ELIMINATION OF COBALT-60 BY CRAYFISH. C. W. Wiser and D. J. Nelson (Oak Ridge National Lab., Tenn.). Am. Midland Naturalist, 72: 181-202(July 1964).

Comparative studies of ^{60}Co uptake by living and dead crayfish (Cambarus longulus longerostris Ort.) indicated that about one-half of the accumulation was by adsorption and one-half by absorption. However, most of the absorbed ^{60}Co was deposited in the exoskeleton, and 95% of the body burden was lost at molting. Newly molted crayfish accumulated about one-half as much ^{60}Co as did individuals with hardened exoskeletons. Large animals accumulated more ^{60}Co than small ones, but the uptake per gram was greater in the small crayfish, indicating the importance of surface adsorption phenomena. Rates of accumulation, measured by biological half-times, were 70 hours for small animals and 66 hours for large animals living in a continuously renewed medium. Uptake half-times under conditions where the medium was not renewed were 24 hours for small crayfish and 16.5 hours for large ones, showing that the higher percentages of total uptake occurred more quickly when the supply of ^{60}Co was limited and with larger animals. The half-time for accumulation by dead crayfish was about three times that for live

crayfish. Elimination rates were much slower than accumulation, with losses being as slow as 0.9% to 1.6% per day until ecdysis occurred. The half-time for elimination by small crayfish was 37 days and that for large crayfish 70 days, both having been dosed by exposure to radioactive solutions. Elimination by crayfish dosed by injection was more rapid, with half-times 13¾ and 19 days for low-activity and high-activity injections, respectively. Cobalt-60 occurred at low and equal levels in blood, muscle, and gonads, while higher concentrations were found in the gut, hepatopancreas, and integument. The integument contained the greatest amount per unit weight, with the hepatopancreas ranking second.

329 GROWTH RESPONSE OF TWO SEDGES INHABITING A RADIOACTIVE WASTE DISPOSAL AREA. Plummer, G. L. (Univ. of Georgia, Athens); Crossley, *D. A. Jr.*; Gardiner, D. A. Ecology, 46: 346-8(1965).

On White Oak Lake bed, an area contaminated with radioactive wastes, sedges (Carex frankii and C. vulpinoidea) were found to have longer inflorescences than sedges growing on nearby uncontaminated sites. This lake bed is a unique ecological system of about 48 acres, which once served as a final holding basin in the Oak Ridge National Laboratory's low-level radioactive waste disposal system. In 1955 the lake was drained, and during the spring season that followed plants and animals invaded the radioactive sediments. Soils contained ^{90}Sr, ^{137}Cs, ^{60}Co, ^{106}Ru, and other radioisotopes. The observed increase in the size of the inflorescences of the two sedges inhabiting this radioactive area is correlated with the complex interaction between various edaphic factors that contribute to plant growth and an effect of ionizing radiation. These perennial sedges tolerate a continuous dose rate generally in excess of 2 R/day at ground level. The air dose rate of γ radiation measured at 1 m above ground surface provided the best correlation with inflorescence length for the two sedge species. Although various soil factors were found to affect inflorescence length, those influences did not explain the increase in length in the radioactive areas. The variables doubtless include many more factors than radiation, however, and it is likely that other soil characteristics (nitrates and sulfates in particular) associated with inorganic constituents of industrial wastes, influence the lengths of inflorescences. The absorbed radiation dose (in rad) delivered to the plants is probably greater than the air dose measurements (in mR) would estimate. Even at minimal estimates, the dose rates on this lake bed are sufficient to induce both genetic and physiological disturbances in plants. No morphological distortions were seen in the plants; except perhaps for being larger and greener they seemed identical with controls. (BBB)

330 TRANSFER OF RADIONUCLIDES FROM THE ENVIRONMENT THROUGH AQUATIC FOOD PRODUCTS TO MAN. W. L. Templeton (Atomic Energy Research Establishment, Seascale, Cumb., Eng.). Food Agri. Organ. U. N., FAO At. Energy Ser., No. 5, 43-67(1964).

Problems involved in the radioactive contamination of the aquatic environment are discussed in terms of the transfer of radionuclides to man. It is pointed out that pollutants normally encountered in industrial waste and sewage are rapidly neutralized by chemical and biological processes in the aquatic environment while radioactive contaminants cannot be destroyed and reliance must be placed on dilution and dispersion and on radioactive decay. In order to have a base line upon which to measure the contribution and effects of artificially produced radionuclides in the aquatic environment the levels and kinds of naturally occurring radionuclides in the medium and in organisms must be determined. Data are presented on the contribution of fallout fission products and radioisotopes in reactor wastes to the radioactivity of seafoods in various areas. Results are included from laboratory and field studies on the metabolism of certain fission products by marine organisms. Emphasis is placed on the content of ^{137}Cs, ^{90}Sr, ^{89}Sr, ^{65}Zn, and ^{106}Ru in sea foods. Data on the consumption of sea foods by several population groups are used to estimate the contribution to radionuclides from this source to the radiation exposure of man. (C.H.)

331 ON THE RADIOACTIVE CONTAMINATION OF SOILS AND CROPS. A. C. Schuffelen (State Agricultural Univ., Wageningen, Netherlands). Food Agri. Organ. U. N., FAO At. Energy Ser., No. 5, 68-91(1964).

Fission products that reach the surface of the earth as fallout or as waste from reactors contain more than 30 radioactive nuclides, but only 3 of these are potential hazards. These are ^{90}Sr, a bone seeker with a long half life; ^{137}Cs that circulates in the body with a long half life; and ^{131}I that accumulates in the thyroid gland but has a relatively short half life. Iodine-131 is of particular interest in fresh fallout and in the case of reactor accidents while ^{137}Cs and ^{90}Sr are especially important in older fallout. The movement of these three radioisotopes from the soil, through soil, through plants, and to man is discussed. Emphasis is placed on the role of soil in the accumulation of fission products by fixation on soil particles. (C.H.)

332 EXPERIMENTAL UPTAKE OF STRONTIUM-85 BY FRESH-WATER ORGANISMS. W. A. Brungs (Robert A. Taft Sanitary Engineering Center, Cincinnati). Health Phys., 11: 41-6(Jan. 1965).

Factorial studies were conducted to determine the effects of variable concentrations of calcium, magnesium, sodium, and potassium on the uptake of dissolved strontium-85 by small bluegills, Lepomis macrochirus. The distribution of strontium-85 introduced to a pond containing macrofauna indigenous to the Midwest was also investigated. Of the four cations tested, only calcium was found to have a significant effect within the experimental range on the uptake of strontium-85 by small bluegills. Data from the pond uptake study indicate that after the 80-day experimental period only 25% of the strontium-85 remained in the dissolved phase. The total calculated activity in the experimental fauna never exceeded 0.5% of the amount added. The remaining radioactivity was associated with the substrate. The data also indicate that strontium-85 accumulation is related to age of the test organisms.

333 EVALUATION OF MARINE CONTAMINATION WITH RADIOISOTOPES OF LONG HALF-LIVES. Yamagata, N. (Inst. of Public Health, Tokyo). Health Phys., 11: 1005-8(Oct. 1965).

A concept using the average concentration factor of stable Co, Cr, Cs, Fe, Mn, and Zn from sea water to the sea food as a whole, or the equivalent daily intake of sea water, was introduced in the evaluation of marine contamination with radioisotopes of long half-lives. The factors were determined on the basis of accumulated data on the trace-element composition of typical diets in Japan.

334 IONIC REGULATION IN THE SQUAT LOBSTER GALATHEA SQUAMIFERA, WITH SPECIAL REFERENCE TO THE RELATIONSHIP BETWEEN POTASSIUM METABOLISM AND THE ACCUMULATION OF RADIOACTIVE CESIUM. Bryan, G. W. (Plymouth Lab., Eng.). J. Marine Biol. Assoc. U. K., 45: 97-113(1965).

This work formed part of a survey made to find whether any marine invertebrates were able to accumulate Cs to a particularly high degree. Such a species might produce a hazard if the fission product ^{137}Cs (half life = 30 yr) were present in the sea water either as a result of the disposal of radioactive waste or as a result of weapon tests. Generally, the small amount of Cs in sea water (0.37 $\mu g/l$) can be concentrated in most of the tissues of marine invertebrates by factors which are higher than those for K. Thus ^{42}K,

^{134}Cs, and ^{137}Cs were used to study the absorption of Cs from sea water in relation to K metabolism, and a general study was made of the ability of Galathea to carry out osmotic and ionic regulation in comparison with better known decapod crustaceans, previously used for radioactive Cs experiments. Galathea appears to be a truly marine decapod in that it is probably unable to osmoregulate and the body surface is very permeable to water and to Cs and K ions. Despite this high permeability, Galathea is able to regulate K and Cs at blood plasma levels which are 1.3 and 1.1 times those of sea water. The Na and Cl concentrations in the plasma are not regulated and the Ca concentration is variable, but Mg and SO_4 are maintained at concentrations below those of sea water. Urinary excretion does not assist in the conservation of K or radioactive Cs in the plasma but assists in removing Mg and SO_4. Regulation of K in plasma and muscle tissue also occurs in low K sea water and causes the whole-animal equilibrium concentration factors which can be attained by ^{42}K absorption to be increased (e.g. from 4.5 in sea water with 12 mM/kg of K to 7.5 in sea water with 6 mM/kg of K). Radioactive Cs is absorbed far more slowly than ^{42}K, but higher concentration factors are usually reached. Both ions penetrate the body surface rapidly, but the tissues constitute the limiting factors in uptake by the whole animal. Losses of both radioactive Cs and K take place chiefly across the body surface and losses due to urinary excretion are negligible, although the equivalent of about 15% of the body weight is excreted per day. Comparative results in other marine invertebrates are discussed.

335 THE ACCUMULATION OF RADIONUCLIDES BY THE THORNBACK RAY, RAIA CLAVATA L., IN THE IRISH SEA. John Mauchline and Angela M. Taylor (United Kingdom Atomic Energy Authority, Windscale, Cumb., Eng.). Limnol. Oceanog., 9: 303-9(July 1964).

A study of the levels of radionuclides accumulated by the thornback ray, Raia clavata, a nonmigratory species, was made in the Irish Sea, where these fish have lived most of their lives in an environment contaminated with the fission products ^{106}Ru, ^{144}Ce, ^{137}Cs, and ^{90}Sr. The relative importance of the different radionuclides as contaminants of fish tissues and the importance of determining whether a steady-state system exists in a marine area receiving radioactive wastes are discussed. (auth)

336 RADIOEKOLOGIYA MORSKIKH ORGANIZMOV. NAKOPLENIE, BIOLOGICHESKOE DEISTVIE RADIOAKTIVNYKH VESHCHESTV. (Radioecology of Marine Organisms. Accumulation and Biological Effects of Radioactive Materials). G. G. Polikarpov. Moscow, Atomizdat, 1964. 295p.

Data are given on the accumulation of radioactive isotopes (from eight chemical groups) in marine ecology. The uptake of fission-product isotopes by algae and aquatic organisms was evaluated, and the radioresistance of hydrobiota was estimated. An extensive list of references is included. (R.V.J.)

337 CHROMOSOMAL ABERRATIONS IN A NATURAL POPULATION OF CHIRONOMUS TENTANS EXPOSED TO CHRONIC LOW-LEVEL ENVIRONMENTAL RADIATION. B. Gordon Blaylock. Thesis, Knoxville, Tenn., Univ. of Tennessee, 1963. 89p.

The salivary gland chromosomes of Chironomus tentans larvae collected from White Oak Creek, an area contaminated by radioactive waste from the Oak Ridge National Laboratory, and from six uncontaminated areas were examined for chromosomal aberrations. White Oak Creek populations were exposed to 230 rads per year or about 1000 times background. Chromosomal maps were constructed to make a general comparison of the banding pattern of the salivary chromosomes of the C. tentans in the East Tennessee area with those of Canada and Europe. These maps were used as a reference in scoring aberrations. Fifteen different chromosomal aberrations were found in 365 larvae taken from the irradiated population as compared with five different aberrations observed in 356 larvae from six control populations, but the mean number of aberrations per larva did not differ in any of the populations. The quantitative amount of heterozygosity was essentially the same in the irradiated and the control population, but there were three times the variety of chromosomal aberrations found in the irradiated area. From this evidence it was concluded that chronic low-level irradiation from radioactive waste was increasing the variability of chromosomal aberrations without significantly increasing the frequency. Five different inversions were found in both the irradiated and control populations; three inversions — 1Ra, 2Lab, and 3Ra — were found at a relatively high frequency and were used in testing the seasonal and geographic distribution. In the White Oak Creek population the frequencies of the heterozygous inversions did not change significantly for a three year period from 1960 to 1962. Also, no changes that could be attributed to seasonal change were detected in the frequencies of these inversions from June through December of 1961 and September through December of 1962. There was no significant difference in the frequencies of these same heterozygous inversions between three local geographic populations of C. tentans. It was concluded that the three inversions which occurred at the highest frequencies are endemic to the population of East Tennessee and have been established in the population for a long time. The frequencies of the standard arrangement, the heterozygous inversion, and the homozygous inversion of 1Ra were significantly different from the expected frequencies as predicted by the Hardy-Weinberg Law. A deficiency of inversion heterozygotes was found which is contrary to the theory of balanced chromosomal polymorphism, but is consistent with a model proposed by Levene, where under certain assumptions genetic equilibrium can be obtained without the heterozygote being superior. The results show that chromosomal polymorphism can be maintained in a natural population without superiority of the heterozygote individuals. (Dissertation Abstr.)

Handling and Transportation

See also the following references:

31, 40, 53, 88, 103, 155, 158, 253, 886, 937, 959

338 (AFWL-TDR-64-81) AIR TRANSPORTABILITY OF REACTOR AND RADIOISOTOPE MATERIALS. James Fox (Air Force Weapons Lab., Kirtland AFB, N. Mex.). Aug. 1964. 258p. (AD-606056). $6.00(CFSTI).

A study was made to investigate the prime considerations involved in air transportation of nuclear reactor components, radioactive wastes, and radioisotope power-generating units. The logistics, economics, and safety of supplying a remote military nuclear power plant by air, and of transporting SNAP-type reactor and radioisotope auxiliary power units by USAF aircraft are explored in detail. It

was determined that such air transport operations are feasible and reasonably safe if proper planning is accomplished in the design of equipment and in the definition of mission profiles and ground handling procedures. Container criteria and procedural standards for the transport of the different categories of materiel are recommended. (auth)

339 (CONF-650103-6) INVESTIGATION OF LOW-LEVEL RADIOACTIVE WASTE CONTAINERS IN AN ACCIDENT ENVIRONMENT. Joseph E. Minor and Leonard U. Rastrelli (Southwest Research Inst., San Antonio, Tex.). Jan. 14, 1965. Contract AT(30-1)-2140. 28p. Dep.(mn); $2.00(cy), 1(mn) CFSTI.

From International Symposium on Packaging and Transportation of Radioactive Material, Albuquerque, N. Mex.

A summary is presented of the work accomplished during an AEC-sponsored effort to investigate the structural integrity of low-level radioactive waste containers in an accident environment. The effort associated with this program was concentrated in two major areas: (1) a survey of current practice for low-level radioactive waste transport and disposal and (2) the structural analysis of typical low-level radioactive waste containers in an accident environment. Each of these areas is discussed and summarized, and conclusions concerning the dynamic response of typical containers are outlined. (auth)

340 (JUL-249-RW) PACKAGING DISASSEMBLED FUEL ELEMENTS AND RADIOACTIVE WASTE IN THE HOT CELLS AT THE KERNFORSCHUNGSANLAGE JUELICH. Retz, A.; Schmiddem, P.-J.; Stephan, W.; Rau, J. (Kernforschungsanlage, Juelich (West Germany). Institut fuer Reaktorwerkstoffe). Apr. 1965. 22p. Dep. mn.

Depending on type of material, activity level, and fire and health hazard, several methods of waste and fuel packaging are used in the Hot Cells at Kernforschungsanlage Jülich. Low level wastes such as plastics, rags, and small tools are remotely welded into plastic bags of 0.2–0.4 mm thickness. Wastes of higher activity level are packed into disposable cans. Tests for the tightness of the containers are described. For highly active fuel wastes which must be most securely packaged and stored, three methods were tested: remote argon-arc welding of stainless-steel and aluminum cans, electromagnetic closing of aluminum cans, and cold-pressure welding of aluminum cans. By comparison of leak-tightness, resistance to mechanical damage, and costs, the method of cold-pressure welding was chosen. (auth)

341 TRAILED CONTAINER FOR RADIOACTIVE LIQUIDS. Bourdrez, J.; Leconnetable, J.; Rouanet, G. (CEN, Saclay, France). Bull. Inform. Sci. Tech.,(Paris), No. 93, 27-40(May 1965). (In French).

A trailed shielded tank for loading, transporting, and discharging radioactive liquid wastes is described. The stainless steel container packed in a cast-lead shell can contain 4.500 liters of corrosive waste, the permissible radioactivity level is about ten curies β/γ per cubic meter. Equipment was studied to give an absolute safety for users; pumping up and out does not cause any risk of irradiation or liquid and atmospheric contamination. Tank characteristics are in accordance with the code for road traffic and with the specific rules applied to dangerous and radioactive materials conveyance; so it can be employed for transport outside the nuclear plant.

342 ORGANIZATION OF A CENTRALIZED REMOVAL OF RADIOACTIVE WASTES FROM THE SITES OF THEIR ACCUMULATION. Sobolev, I. A.; Khomchick, L. M. Gigiena i Sanit., No. 5, 78-9(May 1965). (In Russian).

Safe removal of radioactive wastes from cities or industrial installations where they are generated and accumulated as a result of large-scale operation with isotopes can be best achieved by establishing one or several central sites that can be conveniently reached by an existing road network. Only solid and liquid wastes, the activity of which exceeds by a factor of 100 the activity of the maximum permissible concentrations for isotopes with half lives up to 60 days and by a factor of 10 that of isotopes with longer half lives, are collected in these central disposal areas. Liquid wastes with an activity of 1×10^{-4} C/l or more are accepted without a daily limit; lower-activity wastes are accepted only from installations that accumulate less than 200 l/day; if the daily effluents are larger, these installations must operate their own disposal plants. Liquid waste must be pretreated to have a pH $\cong$ 7. Solid wastes must be packaged in order to exclude the possibility of dust formation during loading and unloading operation and transportation; the surface activity of the packages should not exceed 70 mr/hr and the dose rate in the driver's cab should be kept below 0.8 μr/sec. A bill of lading containing all the necessary technical data must accompany each shipment. Dosimetric control is maintained at each phase of the operation. Special trucks are provided for ultimate removal from these centralized locations; the driver serves as health physics agent and carries out the measurements.

343 NORME PER IL TRASPORTO DI MATERIALE RADIOATTIVO. (Standards for the Transport of Radioactive Materials). Rome, Comitato Nazionale Energia Nucleare, 1963. 146p.

The book is divided into three sections. The first part (Regulations for the Transport of Radioactive Materials) is the Italian version of STI/PUB/40 NSA 15: 23800). The third part (Standards for Protection Against Radiations During the Transport of Irradiated Fuel Elements) is the AEC Regulation 10 CFR, Part 72. The second part concerns international regulations for the transportation of radioactive materials by rail. The classification of radioactive materials is given and the conditions for the transportation with regard to the packing, method of transport, bill of lading, transportation materials, prohibitions, and warning lettering are reported.

Maximum Permissible Concentrations

See also the following references:

34, 251, 265, 355, 488, 854, 857

344 DERIVED "MAXIMUM PERMISSIBLE LEVELS" USED IN NEIGHBOURHOOD AND WORKING ENVIRONMENT SURVEYS. H. J. Dunster and R. J. Garner (United Kingdom Atomic Energy Authority, Harwell, Berks, Eng.). 10p. (CONF-365-1)

From Symposium on Environmental Monitoring, Berkeley, Gloucestershire, England, Oct. 1963.

The interpretation of measurements and calculation of derived working limits in environmental surveys is discussed. Interpretation of the results of environmental surveys in terms of ICRP recommendations or statutory requirements is stressed. To illustrate the wide range of applications in which DWLs have proved useful, and to give some indication of the limits of their applicability, a number of values in use in the UKAEA is summarized. The concept of derived working limits can be used with great effect in the interpretation of environmental survey results provided that their inherent inaccuracies are remembered and provided that they are not used in conditions for which they were not intended. (P.C.H.)

345 (HW-SA-3550) CALCULATION OF MPC_E FOR U^{235} FISSION PRODUCT RADIOAEROSOLS. F. E. Owen (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). [May 26, 1964]. Contract AT(45-1)-1350. 14p. (CONF-642-7). Dep.(mn); $1.00(cy), 1(mn) OTS.

From Health Physics Society, Cincinnati, June 1964.

The effective maximum permissible concentration (MPC_E) for an airborne mixture of radionuclides may be calculated from the weighted averages of the radionulides present, based on their individual $MPC_A^{(1)}$ and the composition in the mixture. In developing an MPC_E, there are several factors which may be of considerable significance in the final result. The composition of a mixture is constantly changing with time as the various radionuclides decay away. Radionuclides decay by different modes, alpha, beta, and some by internal conversion with only a gamma emission. Each radionuclide has a different deposition affinity in the various organs of the body, thus requiring a separate set of MPC's for each organ. Solubility of the mixture or portions of the mixture also changes conditions. In some cases, it may be necessary to overcome the complication created by the interplay of these factors by arbitrarily assuming the most restrictive condition. The calculation of an applicable MPC_E for airborne particulate contamination (radioaerosols) of a changing mixture is demonstrated. Uranium-235 fission products were chosen to illustrate the method because they are prevalent at many reactors and supporting chemical separations facilities. The radioaerosol MPC_E for fission product mixtures from 70-days irradiation was calculated over a decay period of 1 to 1000 days. Four organs were considered, lung, GI tract, bone, and total body. During the first few months of decay, the MPC_E for the lungs is limiting, starting at 8×10^{-8} $\mu c/cc$ after one day, and reducing to 3×10^{-8} $\mu c/cc$ after three months. After six months, the MPC_E for bone becomes limiting, starting at 2×10^{-8} $\mu c/cc$, and reducing to 1×10^{-8} $\mu c/cc$ after a year and to 2.7×10^{-9} $\mu c/cc$ after 1000 days.

346 PERMISSIBLE CONCENTRATIONS OF RADIOACTIVE FISSION PRODUCTS IN AIR, DRINKING WATER, AND FOOD FOR THE GENERAL POPULATION. Th. Franke (Max-Planck-Institut für Biophysik, Frankfurt am Main). Biophysik, 1: 123-34(1963). (In German)

The permissible concentrations of radioactive fission products in air and water were determined, using as a basis the concentrations of radioactive nuclides in air and water judged permissible by the International Commission on Radiological Protection. The calculations were carried out for the following critical organs: the whole body, bones, gastrointestinal tract, and lungs. The neutron-induced radioisotopes and the isotopes of bomb materials present in fallout were considered. Under consideration of the contamination of the biosphere arising as a result of nuclear test explosions and the resulting radiation effects on man, permissible concentrations of radioactive fallout products in air, drinking water, and the principal foods were given. (tr-auth)

347 MAXIMUM PERMISSIBLE LEVELS OF CONTAMINATIONS. R. Oliver (Churchill Hospital, Oxford). Health Phys., 9: 951-2(Sept. 1963).

Maximum permissible levels of radioactive contamination in Great Britain are discussed from the point of view of the practical health physicist. (C.H.)

348 WHAT IS A PERMISSIBLE DOSE? E. E. Pochin (University Coll., London). Health Phys., 9: 1091-4(Dec. 1963).

Discussion is presented on permissible doses under various conditions. It is stressed that there is a very wide range of the levels of hazard in different existing industries. However, levels of permissible population exposure that are the lowest ones necessitated by the processes involving the exposures, and yet are justified, or more than justified, by the needs for these processes can be hoped for. (P.C.H.)

349 RECOMMENDATIONS OF PUBLICATION 6 OF THE ICRP. Karl Z. Morgan. Nucl. Safety, 6: No. 1, 20-31 (1964).

Recommendations on radiological protection in the sixth ICRP report are summarized. Factors considered in the report are long-term effects, medical exposure of the population, medical exposure of women of reproductive age, exposure groups, short-term and emergency exposures, reduction of external exposure of employees with body burdens of radionuclides, area and volume to be considered when estimating dose, effects of dose rate, and eye exposure. Reduction of maximum permissible concentration (MPC) values for uranium, increase of MPC values for ^{90}Sr, and changes in MPC values for other isotopes are reported.

350 AN IDEA ON THE MAXIMUM PERMISSIBLE CONCENTRATION OF RADIOACTIVE MATERIALS IN SEA WATER. Hiyama, Yoshio (Tokyo Univ.). Rec. Oceanogr. Works Jap., 5: No. 2, 98-104(Mar. 1960).

The author tried to find a relationship between the level of seawater contamination by several radionuclides and the level of radiation dose-rate inside the human body caused by these radionuclides of marine origin, using the recommendation of the ICRP and data on the chemical analysis of sea water and of the human body. Radionuclides which are important in view of their presence in waste and their effect on public health are considered. The evaluation of the importance of these radionuclides cannot be made merely on the basis of the amount of discharge and the speed of decay. Additional consideration should be given to the fact that some elements are concentrated more than others from seawater by marine organisms and from marine products by the human body. Furthermore, in the estimation of physical decay from the time of discharge to the time of human intake, the period of preservation and storage of the marine products, which possibly differs according to national food habits, must be considered in addition to the period of time required for the physical and biological transmission of the materials from the place of discharge to the food-producing zone of the sea. The speed of uptake of radionuclides from the environmental seawater is higher in microorganisms, which also collect nuclides adhering to the body surface, than in macroorganisms. The latter, which generally have more value as human food than the former, are contaminated either by intake of macroorganisms or by direct material exchange through the gill membranes to and from the seawater. The time required to get equilibrium between the seawater and the body of a fish varies according to the nature of the element and the physiological activity of the fish. Numerous experiments have shown that this time is rarely on the order of several hours but commonly several days or weeks. Thus short-lived radionuclides, with half lives of less than a day, cannot enter into consideration for this problem. The radionuclides considered include ^{90}Sr, ^{137}Cs, ^{60}Co, ^{65}Zn, ^{144}Ce, ^{131}I, ^{103}Ru, ^{99}Mo, ^{140}Ba, and ^{140}La, and radioisotopes of Cd and Fe.

351 CALCULATION OF PERMISSIBLE CONCENTRATION LIMITS OF RADIOACTIVE GASES IN THE AIR OF WORK AREAS AND IN THE ATMOSPHERE FOR VARIOUS EXPOSURE CONDITIONS. G. M. Obaturov. Symp.

Tech. Zagadnien Ochrony Promieniowaniem, Warsaw, 355-76(1962). (In Russian)

The calculations of the maximum permissible concentration were made for most radioactive gases in the laboratory air and in atmospheric air. Factors considered in the calculations included internal irradiation from the radioactive gases and from their decay products; volume of the work area, which affects the portion of γ radiation in the MPC magnitudes; radiological sensitivity of the individual human organs; and absorption of the β particles in various organs. Tables of the MPC values were compiled for the individual radioactive gases. All isotopes were grouped according to their half lives and radiation energy. The analytical methods for the different groups were given. The possibility of constructing an isodose gas dosimeter was discussed. The formulas for the calculation of irradiation of critical organs were given and the maximum half life for the isotopes forming the radioactive aerosols was determined. (tr-auth)

GASEOUS WASTES

GENERAL

See also the following references:

12, 17, 18, 21, 37, 153

352 (TID-7677) EIGHTH AEC AIR CLEANING CONFERENCE, OAK RIDGE NATIONAL LABORATORY, OCTOBER 22-25, 1963. B. H. Kornegay, D. K. Jamison, and J. M. Morgan, Jr., eds. (Johns Hopkins Univ., Baltimore). Contract AT(30-1)-1477. 693p.

Thirty-one papers and 4 panel discussions presented at the Eighth AEC Air Cleaning Conference are given along with 16 supplementary papers. Separate abstracts were prepared for 33 of the papers. The remaining 13 were previously abstracted for NSA. (M.C.G.)

353 DESIGN CONSIDERATIONS FOR OFF-GAS SYSTEM MAJOR COMPONENTS. Fuller, A. B. Nucl. Safety, 6: 422-5; 431(Summer 1965).

Designs of gaseous waste handling systems are discussed. Most of the design approaches discussed have evolved from repeated trials of these and less effective methods of obtaining the required reliability of operation.

354 CURRENT PRACTICES IN THE DISPOSAL OF WASTE RADIOACTIVE GASES FROM NUCLEAR REACTORS. Stolzenbach, C. F. Nucl. Safety, 6: 436-40 (Summer 1965).

A review of information concerning disposal of radioactive gases is presented. Discussions are included of limitations on disposal of waste radioactive gases, source and identity of waste radioactive gases, waste disposal methods, and presently used waste disposal practices.

355 V. RADIOBIOLOGICAL AEROSOL PROBLEMS. PROPERTIES AND RISKS OF ATOMIC INDUSTRY RADIOACTIVE AEROSOLS. I. RADIOIODINE (REVIEW REFERENCES). W. Herbst (Universitat, Freiburg i. B.). Zentr. Biol. Aerosol Forsch., 11: No. 3, 232-73(Sept. 1963). (In German)

The properties, distributions, and biological dangers of radioiodine are reviewed along with detection methods and safety. Topics covered include: physical aspects of the genesis of radioiodine, radioiodine in the biosphere and environment, radiobiological toxicity of radioiodine, maximum permissible doses and concentrations as well as radiation protection standards for radioiodine at various risk levels, actual situations of the contamination of the biosphere with radioiodine, and first aid and control. Two hundred and seventeen references are included. (M.C.G.)

356 (AP-Note-439) AIRBORNE ACTIVITY SAMPLING PROGRAM, SUBTASK 8.2. J. F. Cox (Alco Products, Inc., Schenectady, N. Y. Nuclear Power Engineering Dept.). June 22, 1962. Contract AT(30-1)-2639. 17p. Dep.(mn); $1.00(cy), 1(mn) CFSTI.

The necessity for characterizing, in detail, airborne activities being discharged from Army nuclear power plants is discussed. A preliminary system is presented. A technique for quantitatively measuring gaseous airborne activity levels is discussed in detail.

AEROSOLS AND PARTICULATES

Analysis and Monitoring

See also the following references:

34, 379, 382, 538

357 STUDY OF RADIOACTIVE WASTE FROM THE EL.2 AND EL.3 REACTORS OF THE SACLAY CENTER OF NUCLEAR STUDIES. L. Fitoussi and P. Lebouleux. 21p. (CONF-337-11). (In French)

From International Conference on Radioactive Pollution of Gaseous Media, Saclay, France, Nov. 1963.

After a brief introduction on the necessity for the control of radioactivity from the point of release of gaseous effluents from nuclear installations, the air circuits of the EL-2 and EL-3 Reactors and the installations for the measurement of the radioactivity released by these reactors are described. These measurements include γ radiation, gases, radioactive aerosols, and, particularly, ^{131}I. The results obtained during several months of operation are reported, and the different radioisotopes released are defined. The results make possible the evaluation of the practical efficiency of filters for the different radioisotopes. The results are discussed, and simplified solutions for the control of gases released are proposed. (tr-auth)

358 (HW-SA-3434) PLUTONIUM AEROSOL PARTICLE SIZE DISTRIBUTIONS IN ROOM AIR. B. V. Anderson (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Apr. 15, 1964. Contract AT(45-1)-1350. 23p. (CONF-561-1)

From the Symposium on Inhaled Radioactive Particles and Gases, Richland, Wash., May 1964.

The particle size of plutonium aerosols in employee work areas is an important consideration in the analysis of long-term continuous exposure received by radiation workers. The particle-size distributions found in work areas of a plutonium processing area were determined by routine air sampling. Autoradiographic examination of the

air samples was utilized to determine the equivalent spherical particle sizes in a range from about 0.05 to 10 microns. Aerosols consisting of essentially nonradioactive inert materials and plutonium oxide or plutonium carbide were examined. The plutonium particle distributions observed are compared to similar data in the literature on controlled oxidation or burning of plutonium with controlled atmospheres and conditions. (auth)

359 (NP-15102) IDENTIFIKATSIYA I IZMERENIE AKTIVNOSTI DOLGOZHIVUSHCHIKH OSKOLOCHNYKH BETA-AKTIVIYKH IZOTOPOV, VKHODYASHCHIKH V SOSTAV DISPERSNOI FAZY AEROZOLEI. (Identification and Measurement of the Activity of Long-Lived Beta-Active Isotope Fission Fragments in the Dispersion Phase of Aerosols). Report No. 118. Bolotin, V. F.; Nemilov, N. F. (Gosudarstvennyi Komitet po Ispol'zovaniyu Atomnoi Energii SSSR). [1964]. 13p. Dep.(mn).

Radiometric analysis was made of beta active fragments (^{137}Cs, ^{90}Sr, ^{90}Y, ^{106}Ru, ^{106}Rh, ^{144}Ce, ^{144}Pr, ^{95}Zr, and ^{95}Nb) simultaneously present in the air of radiochemical plants. The identification and measurements were carried out with an air-collector with filter belt and β- and γ-spectrometer transducers erected on one stand and an electron-measuring equipment consisting of seven identical differential discriminators regulated by the channel width and seven identical numerical counters on another stand. The sensitivity of the installation is about 1×10^{-13} c/l in cases in which the content of all isotopes is identical.

360 (NRL-5970) FILTER PACK TECHNIQUE FOR CLASSIFYING RADIOACTIVE AEROSOLS BY PARTICLE SIZE. PART 1. PRELIMINARY REPORT AND EVALUATION. L. B. Lockhart, Jr. and R. L. Patterson, Jr. (Naval Research Lab., Washington, D. C.). Aug. 16, 1963. 10p.

Study of the efficiency of filters for the retention of natural airborne radioactive aerosols showed that the particles with which the radioactivity of the short-lived radon daughters is associated have greater ability to penetrate the various filter media that has the fission product debris in the atmosphere; therefore the former is associated with aerosols of smaller size. A preliminary evaluation of the techniques of employing packs of filters of different retentivity characteristics to determine the particle size and/or particle size distribution of radioactive aerosols indicated the feasibility of the method. (auth)

361 (ORNL-TM-1384) DESCRIPTION OF THE CELL VENTILATION MONITOR FOR THE FISSION PRODUCTS DEVELOPMENT LABORATORY BUILDING 3517. Knowles, D. J.; Danforth, H. P.; Chase, L. H. (Oak Ridge National Lab., Tenn.). Dec. 15, 1965. Contract W-7405-eng-26. 7p. Dep. mn. CFSTI $1.00 cy, $0.50 mn.

The exhaust ventilation air from the Fission Products Development Laboratory (FPDL) Building 3517 is filtered before it flows to the 3039 stack. The air is monitored for radioactive particulates after being filtered at Building 3517 and again at the 3039 stack. The intended function and details of the components in the FPDL Duct Monitor at Building 3517 are described. A checkout procedure is attached.

362 CONTINUOUS OUTDOOR AIR SAMPLER. Mitchell, Kenneth K. (Union Carbide Corp., Paducah, Ky.). Am. Ind. Hyg. Assoc. J., 26: 285-8(May-June 1965).

A 115-volt electric-powered air sampler that continuously samples outdoor air with high efficiency for particulate gaseous contaminants was assembled and tested by field use. It operates with little attention for more than a week and was successfully tested in ambient temperatures ranging from −10 to 100°F. The sampler consists of a vacuum pump which pulls air first through a membrane filter for removing particulate materials and then through a liquid-filled scrubber for removing gaseous contaminants. The complete sampler, including the aluminum cabinets which house the sampling equipment, costs several hundred dollars.

363 ANALYTIC FILTERS MADE OF ORGANIC MICROFIBERS AND THEIR USE IN DETERMINING THE CONCENTRATION OF AEROSOLS AND DUST. K. Spurny, M. Polydorova, and Z. Starcuk (Inst. of Physical Chemistry, Czechoslovak Academy of Sciences, Prague). Pracovni Lekar., 14: 369-75(Oct. 1962). (In Czech.)

The filtration properties of Czechoslovak analytic filters are reported. Altogether 15 different kinds were examined, some of which were equal in quality to similar filters of foreign make. The staining properties of filters made of organic fibers, their clogging by industrial aerosols, and their applicability for determining the concentration of radioactive aerosols are considered. Filters made of chlorinated, cold-pressed polyvinylchloride proved best. These filters have a very low resistance to flow, are sufficiently strong, and are able to retain large quantities of samples (tenths of mg). Their efficiency is also satisfactory (>95% with particles of 0.1 μ at 10 cm/sec gas velocity). These filters proved very efficient for retaining radioactive aerosols, because they allow large quantities of air to be sucked through and radioactive particles penetrate the filter only insignificantly. However, they cannot be made as transparent as diaphragm filters and are therefore less suitable for determining the numerical quantity of aerosol concentrations, but they can be easily dissolved in organic solvents. (BBB)

364 DOSIMETRIC CONTROL. M. I. Gneushev, V. I. Bad'in, and E. A. Fishevskaya. Radiats. Gigiena, Sb., 1: 126-34(1962).

Principles of general and individual dosimetric control, the chief instruments for recording of x-ray, β-, and γ-radiations, n dosimetry, and measurement of radioactive aerosols and gases, are described. (Ref. Zh., Khim. 1963)

365 FIBROUS FILTERS AS PARTICLE ANALYZERS. M. D. Silverman and W. E. Browning, Jr. (Oak Ridge National Lab., Tenn.). Trans. Am. Nucl. Soc., 6: 401(Nov. 1963).

366 AEROSOLI. (Aerosols). K. Spurnyi, Ch. Iex, B. Sedlachek, and O. Shtorkh. Moscow, Atomizdat, 1964. 359p.

Physico-chemical and biological properties of aerosols are studied, and methods for measuring and filtering air pollution from industrial dust, smoke, steam, and radioactive stack disposal are discussed. Monitoring devices and filters are described. 344 references are included.

367 USE OF ACTIVATED CHARCOAL FOR DETERMINATION OF RADIOIODINE CONTENT IN AIR. G. M. Obaturov and V. D. Kuranov. Symp. Tech. Zagadnien Ochrony Promieniowaniem, Warsaw, 261-79(1962). (In Russian)

The application of activated charcoal to the determination of the concentration of radioiodine in air was investigated. The absorptive and desorptive capacity of the charcoal, the filter FPP-15, and tubes filled with various types

of materials was determined as a function of the velocity of the air flow. A method for the measurement of iodine concentrations in the presence of radioactive gases was proposed. Under the most unfavorable conditions, 0.1 of the MPC of ^{131}I in the air could be determined with a background of 1 MPC in mixtures of gaseous fission products. (tr-auth)

Collection, General

See also the following references:

34, 369

368 (CONF-337-5) GASEOUS EFFLUENT STUDIES AND AIR CLEANING ACTIVITIES IN THE UNITED STATES ATOMIC ENERGY INDUSTRY. David C. Costello, Jr. (Division of Reactor Development, AEC). [1963]. 51p.

From International Conference on Radioactive Pollution of Gaseous Media, Saclay, France, Nov. 1963.

A review is presented of the AEC program of the removal from air of radioactive particulates which are produced by the various AEC nuclear energy activities. The production and control of gaseous wastes in the nuclear fuel cycle, fuel reprocessing, laboratories, mobile power reactors, etc. are discussed. The AEC research and development program on meteorology and air cleaning in gaseous waste control operations is described. (D.L.C.)

369 (IDO-14607) REMOVAL OF PARTICULATES FROM WASTE CALCINING FACILITY OFF-GAS. R. D. Modrow and L. T. Lakey (Phillips Petroleum Co. Atomic Energy Div., Idaho Falls, Idaho). June 1964. Contract AT(10-1)-205. 26p.

In testing prior to radioactive operation of the ICPP Waste Calcining Facility, the overall particulate decontamination factor for the off-gas cleanup system, consisting of a cyclone, a quench tower, a venturi-scrubber and its associated de-entrainment cyclone, silica gel beds, and off-gas filters, was determined to be at least 3.6×10^5. The cleaned off-gas contained not over 5×10^{-6} grains/acf of particulate solids which were essentially spherical and less than 0.1 μ in diameter. Descriptions of the cleanup equipment and the special techniques, i.e., gas sampling, radiotracer, and sample activation, used to obtain data on the rate of flow of solids at various points in the system are included. These data were used to determine the decontamination factor for each of the various pieces of equipment. Subsequent sampling during operation with actual high-level radioactive waste confirmed the test results on all but the off-gas filters which had a higher than expected decontamination factor. The overall decontamination factor for the off-gas cleaning system during the initial two months of radioactive operation of the WCF was higher than that measured in the nonradioactive tests by a factor of ten or more. (auth)

370 (TID-7677(p.280-96)) REMOVAL OF PARTICULATES FROM WASTE CALCINING FACILITY OFF-GAS. R. D. Modrow, L. T. Lakey, and J. A. Buckham (Phillips Petroleum Co. Atomic Energy Div., Idaho Falls, Idaho).

The removal of particulate matter from off-gas from the Waste Calcining Facility (WCF), a unit that will convert radioactive liquid wastes to granular solids in a heated, fluidized bed is discussed. Descriptions of the off-gas cleaning equipment employed are given together with data for each unit. Also discussed are sampling of the off-gas for particulate matter and other special techniques used in evaluating the system performance. The overall decontamination factor for the off-gas cleanup system, consisting of a cyclone, a spray tower, a venturi scrubber and separator, silica gel beds, and AEC filters, was about 3.6×10^5. This resulted in a reduction of the solids loading from approximately 1.4 to about 5×10^{-6} grains/acf; particulates in the final off-gas were essentially spherical and were less than 0.1 micron in diameter. This high degree of decontamination indicated that a noncondensing-type cleanup system can be used to reduce radioactive particulate contamination in a gas stream to a level low enough to permit the gas to be released safely to the atmosphere. The use of a radioactive tracer in the system indicated the occurrence of a delayed-type of penetration of the AEC filters by the submicron particles. About ten days were required to reach an equilibrium value of penetration through the filters. (auth)

Collection by Centrifugation

See the following reference:

34

Collection by Filtration

See also the following references:

19, 32, 34, 88, 95, 208, 360, 361, 363, 365, 368, 369, 370, 389, 485

371 VARIATION OF THE EFFICIENCY OF FILTERS AS A FUNCTION OF THEIR WARPING. F. Billard, G. Madelaine, and J. Pradel. 21p. (CONF-337-48)

From International Conference on Radioactive Pollution of Gaseous Media, Saclay, France, Nov. 1963.

The warping of filters by atmospheric dusts is very slow. Therefore a method was studied that made possible a rapid warping of the different filters used by the C.E.A. The knowledge of this variation curve as a function of time and of the dust mass yielded data on the behavior of filters. The evolution of the filters was also followed as a function of their warping by different aerosols. The efficiency of different filter papers for various types of aerosols was determined with the use of a spectrophotometer for particles with diameter greater than 0.3 μ. The filters were warped with these aerosols and the variations of the efficiency were determined with a standard aerosol. The results showed that for a solid or a non-oily aerosol the efficiency increased with warping. The phenomenon was reversed for an oily aerosol and the permeability increased with the warping. (J.S.R.)

372 DRY PURIFICATION OF INCINERATOR OFF-GASES. P. Dejonghe, J. Pyck, and N. Van de Voorde. 17p. (CONF-337-57)

From International Conference on Radioactive Pollution of Gaseous Media, Saclay, France, Nov. 1963.

During operation of an incinerator for radioactive wastes, black smoke is formed, primarily while burning plastics or rubber. The black smoke, which is not cleared in a cyclone, coagulates on the prefilters and causes almost instantaneous plugging. In order to solve this problem an attempt was made to preclean the gases by filtration through different types of tissues or by electrofiltration. For gases, containing 5 g of humidity per m^3, one succeeded in retaining 93% of the carbon black in a double stage electrofilter. The life time of the tissue-filters was thus prolonged by a factor of approximately 10. The flow-rate through the electrofilter was between 0.4 and 1.0 m/sec. (auth)

373 (CONF-650407, pp 1204-15) DEVELOPMENT OF MEDIUM FLOWRATE, HIGH PURITY PARTICULATE AND IODINE FILTERS. Rodgers, S. J.; Udavcak, R. J.; Mausteller, J. W. (MSA Research Corp., Callery, Pa.).

A program was conducted to develop a high-purity ($\leq$150 ppm total impurities) particulate filter for high-volume air samplers, and an efficient elemental iodine filter. Both filters were required to be completely dissolvable. Of the different fibers evaluated, Fortisan rayon and microfibers (synthetic acetate) were found to be the most promising. Both these fibers were fabricated into filter mats which gave flow rates of 30–35 cfm and had efficiencies >90% for 0.07-μ dia particles. For the iodine filter, organic fibers were carbonized and activated, and then formed into filter mats. The efficiency was 99.5% at flow rates of 30–35 cfm (face velocity 325 ft/min), and the filter can be dissolved in HNO_3–$HClO_4$ or 5% NaOH. A composite filter consisting of separated mats of Fortisan and carbon fibers was also made.

374 (IDO-12032) HAZARDOUS MATERIALS FILTER MANUAL FOR THE NATIONAL REACTOR TESTING STATION. B. J. Held, G. Wehmann, D. R. Mousseau, R. D. Fletcher, and R. G. Anderson (Idaho Operations Office, AEC). Aug. 26, 1963. 34p.

This manual was prepared as a guide to the persons having a responsibility in a filter program. Included are filter specifications, selection, testing, inspection, handling, maintenance, filter box design, and field leak testing at NRTS. (D.L.C.)

375 (SZS-2(1965)) DIE BESTIMMUNG DER EINDRINGTIEFE VON AEROSOLEN IN FILTERN. (The Determination of Penetration of Aerosols in Filters). Loessner, Volkhard (Staatliche Zentrale fuer Strahlenschutz, Berlin (East Germany)). Apr. 1965. 35p. Dep. mn.

A procedure for the determination of the depth of penetration of a radioactive marked monodisperse aerosol in membrane and fiber filters is given. On a largely monodisperse SiO_2 aerosol of 0.55 μ, secondary products of thoron were deposited. This aerosol was deposited in membrane and fiber filters. The residual range of the alpha rays from the ThC′ which was emitted from the filter surface was determined. Under certain conditions, the measured curves for the number of alpha particles, when subjected to graphical differentiation, yield a picture of the depth of penetration of the aerosol. On the basis of the results obtained with membrane and fiber filters, three different forms of the deposition can be derived. Two of them can be explained by the filtration theory (experimental law for filtration). For the third, a possible explanation is given. It is shown that the differential equation for filtration has no general validity. The aerosol, its size distribution, its radioactive tagging, its deposition in the filter, the measuring process, and the measuring chamber are described.

376 (UJV-1103/64) FILTRATSIYA AEROZOLEI V PARE PRI POMOSHCHI VOLOKNISTYKH FIL'TROV. (Filtration of Aerosols in a Vapor by Means of Fibrous Filters). F. Kepak (Československá Akademie Věd. Ústav Jaderného Výzkumu, Řež). [1960?]. 11p.

Tests were made of the filtration efficiency of fibrous filters for aerosols during decontamination of radioactive liquid waste by evaporation. The filtration efficiency was determined for 20-, 15-, 6-, and 3-μ silicon fibers and 3-μ basalt fibers. (R.V.J.)

377 (TID-7677(p.419-24)) FILTER EFFICIENCY AND CLOGGING. J. Pradel, F. Billard, G. Madelaine, and J. Brion (France. Commissariat à l'Energie Atomique. Centre d'Etudes Nucléaires, Fontenay-aux-Roses).

It is generally admitted that filter efficiency increases with clogging. Experiments, conducted with three kinds of aerosols showed that efficiency may decrease with clogging, especially when filtering oily aerosols. (auth)

378 TRAPPING OF SHORT-LIVED RADON DECAY PRODUCTS WITH FP-TYPE FIBER FILTERS. B. I. Ogorodnikov, V. N. Kirichenko, P. I. Basmanov, and I. V. Petryanov. At. Energ. (USSR), 15: 230-7(Sept. 1963). (In Russian)

Filtering materials fabricated from polymeric fibers are widely used for the removal of finely dispersed radioactive aerosols; their filtering efficiency was therefore examined, comparing the various available materials on the basis of their ability to retain the short-lived daughters of Rn^{222}. This nuclide is an intermediate product of the U–Ra decay chain and diffuses continuously into the atmosphere from uraniferous deposits. In view of the importance of the filters for the protection of uranium miners and ore treatment plant operators, the tests were carried out both under laboratory conditions and at an experimental mining site. It was found that the filtering efficiency depends primarily on the basic properties of the material, such as aerodynamic resistance, fiber diameter, and the presence of electrostatic charges and also on the filtration rate and the degree of dispersion of the aerosol. Linear flow velocities from 1 to 200 m/sec were used for the tests. At high linear velocities, filters made of 0.5 and 1.34-micron thick fibers were found to be the most effective; at a standard resistance of 1 mm of water gauge at least 99% of the Rn daughters were retained by the filters. Use of these materials for large-scale industrial filters, for personnel protection and for analytical purposes is also reviewed.

379 STUDIES ON AIR POLLUTION OF ISOTOPE DEPARTMENTS. Laszlo Bozoky (Országos Onkológiai Intezet, Budapest). Magy. Radiol., 14: 228-32(July 1962). (In Hungarian)

The need for a simple, effective, low-cost filter to measure atmospheric radioactive pollution in radiotherapy departments employing isotopes is considered. It was noted that not only may the patients emit substantial γ radiation, but various body discharges, soiled laundry, treatment

equipment, storage areas for laundry, and urine specimens present a radiation hazard for hospital personnel. A light, portable sampling and monitoring device was designed that would preclude the use of large built-in filters and their time-consuming operation. The filter was formed from 20 layers of gauze sewn together making a pad of 25-mm dia. This pad filtered out only 50% of the aerosol particles, but its air resistance was 10 times less than conventional types, allowing it to be attached to an ordinary vacuum cleaner. A gas flow meter, installed between the pad and the vacuum cleaner, showed that 1 m^3 air passed through the pad in ~5 min. The filter samples were assayed for radioactivity with a 25-mm dia G-M tube. When the results were checked against a 99.8% effective Soviet-made filter, the 20-layer gauze filter compared favorably. When measurements were made in rooms of patients treated with I^{131}, the samples showed that the activity decreased much more rapidly than the normal 8-day half life of I^{131}. Since there was no other isotope pollution in the area tested, the only explanation for this atmospheric activity is the rapid sublimation of iodine. The first 24 hours following I^{131} administration showed a generally higher air pollution level than had been assumed. Average measurements showed 10^{-9} $\mu C/m^3$ before the rooms were ventilated. The activity measurements were about the same during treatment with I^{131}, P^{32}, Au^{198}, and Na^{24}. Airing the rooms thoroughly three times a day is considered absolutely necessary, as this decreased the atmospheric pollution considerably. (BBB)

380 FILTRATION OF RADIOACTIVE PARTICULATES. S. E. Smith (Atomic Weapons Research Establishment, Aldermaston, Berks, Eng.). Progr. Nucl. Energy, Ser. IV, 5: 449-75(1963).

The removal of solid particulate radioactive matter from air and other gases in areas where health hazards might form and from reactor cooling systems is discussed. The nature of the aerosols of concern is discussed. The theory of filter action and types of filter media as well as equipment for removing aerosols by electrostatic means, inertial separation, and particle impaction and enlargement are discussed. Some applicable filtration systems are discussed. (R.E.U.)

381 EXPERIMENTAL ESTIMATION OF EFFECTIVENESS OF FILTERS AGAINST RADIOACTIVE AEROSOLS. A. Stankovicova and F. Minarik. Pracovni Lekar., 15: 343-4(Oct. 1963). (In Czech)

The efficiency of five filter materials for the removal of radioactive aerosols was evaluated. The β activity of the filters was measured after air labeled with Po^{218} and Pb^{214} was passed through them. Membrane filters were the most efficient. (C.H.)

382 INVESTIGATION OF AIR CONTAMINATION IN A NUCLEAR MEDICAL FACILITY. L. Bozoky (Central Inst. for Oncology, Budapest). Radiobiol. Radiotherap., 3: 315-18(1962). (In German)

A light portable device consisting of 20 layers of gauze sewn together to make a pad 25 mm diameter was used to measure air contamination in a medical radioisotope facility. The device filtered out only 50% of the aerosols, but its air resistance was low enough so that it could be used with a vacuum cleaner. Its efficiency was ~50% that of a commercial air filter. One m^3 of air was filtered each five min. Measurements of I^{131} air contamination in a medical radioisotope ward were made before and after ventilation. After ventilation the I^{131} air concentration decreased by one order of magnitude from an average of 1.1×10^{-9} $\mu C/cm^3$. This is close to 10% of the maximum permissible concentration (MPC) for a 40-hr work week. The radioactivity of the samples decreased more rapidly than the half life of I^{131} and indicated that some of the radioisotope evaporated. In view of the frequency that personnel dosimeters have recorded exposure to 50% of the MPC the I^{131} air concentrations may be significant. In other areas (laundry, treatment rooms, and laboratory spaces) the air specific activity of I^{131}, P^{32}, Au^{198}, and Na^{24} varied between 10^{-10} and 10^{-9} $\mu C/cm^3$, but increased sharply in areas where radioactive solutions were contained in open vessels, where surfaces were contaminated, or sterilization procedures were improperly carried out. Thorough ventilation of rooms three times daily is absolutely necessary to reduce radioactivity concentrations of air to safe levels. (BBB)

383 INVESTIGATION OF THE STANDING TIME INCREASES FOR RADIOACTIVE FINE-FILTERS BY THE ADDITION OF PRE-FILTERS. Przyborowski, Sieglinde (Staatliche Zentrale fuer Strahlenschutz, Berlin). Staub, 25: 291-4(Aug. 1965). (In German).

The possibility of increasing the life of filters for precipitation, of radioactive aerosols, by using inexpensive primary filters is shown. The effect of primary filters is not only manifested by a decrease in concentration, but also by changes in particle size distribution. The problem of filter life investigations and its solution by means of experiments are discussed in detail.

384 EFFECTIVENESS OF AEROSOL FILTRATION OF SHORT-LIVED PRODUCTS OF RADON ON FPP-15 MEMBRANE. I. I. Kornilenko, V. I. Zybin, and V. I. Glagolev. Symp. Tech. Zagadnien Ochrony Promieniowaniem, Warsaw, 293-326(1962). (In Russian)

The results of studies on separation capacity of the filter FPP-15 with respect to aerosols consisting of short-lived Rn daughter products are reported. The air velocities used were from 1 to 8 m/hr and the Rn concentration in the aerosol chamber from 7×10^{-13} to 6×10^{-10} c/l. The measurement results indicated that the FPP-15 type filter has a high removal efficiency for Rn daughter products (at the air velocities used it is not less than 99%). With the increase of the dust concentration in the air, the removal capacity decreased to 85 to 90%. The investigations with charcoal filters showed that, of the RaA aerosols arising in the air, a small fraction (about 0.5%) penetrates easily through the FPP filter. (tr-auth)

385 (TID-7677(p.56-72)) SAFETY ASPECTS OF THE DESIGN OF FILTERED VENTILATION SYSTEMS. S. E. Smith, F. J. Hall, and W. E. Holmes (United Kingdom Atomic Energy Authority. Weapons Group. Atomic Weapons Research Establishment, Aldermaston, Berks, England).

A number of safety aspects of the design of filtered ventilation systems for radioactive and toxic buildings were considered. These include theoretical and experimental assessments of the heat release from fires of various kinds in workrooms, fume cupboards, glove boxes, and extract systems, the use of spark and flame arrestors, the ignitability of dust deposits in extract systems and filters, and the cooling of hot gases in exhaust ducts. As a

result of this work, it was possible to formulate principles which should be followed in the layout and design of such systems, in the future; existing systems were found not to require any serious modification apart from installation of glass paper filters. (auth)

Collection by Electrostatic Precipitation

See also the following references:

38, 62

386 ATTEMPTS TO COMBINE ELECTROSTATIC PRECIPITATION AND INERTIAL COLLECTION IN A FILTER FOR RADIOACTIVE AEROSOL SAMPLING. J. Flattau. Central. Inst. Ochrony Pracy, 13: 109-26 (1963). (In Polish)

The development of a test filter to measure the radioactivity of aerosols, for the purpose of testing whether the sampling efficiency can be improved by combining electrostatic precipitation and inertial collection, is reported. The filter consists chiefly of a set of corona electrodes, a nozzle and a collection electrode adapted for mounting a cover glass. Filtering efficiency, depending on the type of filter and the conditions of measurement, was between 57 and 17%, whereas precipitation efficiency was from 3 to 6.6%. Filtering efficiency depended upon many other factors which were probably also electrostatic in character. It was established that radioactive decay products are deposited not only on fine particles but also on coarser particles, which is a result contrary to those obtained by Wilkening. (Occupational Safety Health Abstr., 2: No. 7, 1964)

387 ISOTOPE EXCHANGE INVESTIGATIONS ON SILICATE ION EXCHANGERS. II. USE OF THE ISOTOPE EXCHANGE METHOD TO INVESTIGATE FIXATION OF VARIOUS IONS IN CLAY MINERALS BY HEATING. H. W. Levi and E. Schiewer (Hahn-Meitner-Institut für Kernforschung, Berlin). Z. Anorg. Allgem. Chem., 337: 105-12 (Apr. 1965). (In German)

The exchangeability of Na, Cs, Tl, Sr, and Ba ions in clays, particularly in bentonite and vermiculite, after heat treatment was investigated. Two groups of ions could be distinguished. Small ions were more and more fixed above 100°C with increasing treatment temperature. Larger ions were not fixed below 500 to 700°C but showed strongly increasing fixation at higher temperatures. In bentonite Sr^{2+} belonged to the small-ion group, in vermiculite it behaved as a large ion. The results are discussed with regard to the fixation mechanism as well as to the applicability of the ion exchange technique.

Collection by Foaming Agents

See also the following reference:

208

388 (TID-7677(p.297-316)) FOAM SUPPRESSION OF RADIOACTIVE IODINE AND PARTICULATES. Robert E. Yoder and Leslie Silverman (Harvard Univ., Boston. School of Public Health).

Tests were made to evaluate the effects of high activity levels, steam, water, and general scale-up of previous laboratory studies on the efficiency of systems for foam supression of radioactive iodine and particulates. The experiments confirmed previous reports that other lauryl sulfate foams with sodium thiosulfate reactant can be used to decrease iodine concentration in a closed chamber. The I^{131} concentration reduction attributable to leakage, dilution, and wall adsorption was 70% of the initial concentration, 100 minutes after the iodine release. When foam was added to the system, the I^{131} concentration reduction was 90% with air recirculation and 92% with no air recirculation at 100 minutes. The rate of iodine concentration reduction was 1.15% of the initial concentration per minute without foam, and 4% per minute with foam and no recirculation. The presence of particulates altered the pattern of removal, which was far less rapid than if no aerosol was present. (M.C.G.)

Collection by Scrubbing

See also the following references:

34, 369, 370

GASES

Analysis and Monitoring

See also the following references:

34, 38, 357, 481, 525, 538

389 (NYO-841-2) SUMMARY REPORT ON PERFORMANCE OF AIR CLEANING SYSTEM FOR U. S. BUREAU OF MINES BOMAEC-30 INCINERATOR. Dennis, Richard; Silverman, Leslie (Harvard School of Public Health, Boston, Mass.). Dec. 1964. Contract AT(30-1)-841. 156p. Dep. mn; CFSTI $5.00 cy, $1.00 mn.

Evaluation of combustion and gas cleaning equipment tests are presented for studies on the BOMAEC-30 institutional type incinerator developed for low level, radioactive waste disposal. The BOMAEC-30 unit consists of a cylindrical, stainless steel-lined, chamber with all combustion air admitted tangentially above the waste charge. Results of several tests with sawdust and office wastes indicated that: only dry (<10% moisture) materials could be burned completely without auxiliary gas firing; the design burning rate objective of 30 lbs. per hour could not be attained in most tests; and the operating methods were not amenable to a safe, rapid means of waste incineration. A water spray cooling tower and glass fabric filtration system did not furnish adequate gas cleaning.

390 A LARGE VOLUME, CONTINUOUS GAS EFFLUENT MONITORING SYSTEM. Donald E. Feltz (Agricultural and Mechanical Coll., College Station. Tex.). 8p. (CONF-243-40)

From American Nuclear Society Conference on Problems of Operating Research and Power Reactors, Ottawa, Oct. 1963.

A gaseous effluent monitoring system was developed and installed for continuous monitoring of building exhaust gases. The system provides two channels for continuous gamma photopeak or integral monitoring of gases and is comprised of a 3'' × 3'' NaI (Tl) gamma scintillation detector, a shielded 30-liter volume gas sampler with machined helical insert for gas flow control, an air pumping system, a transistorized dual channel gamma spectrometer, and a multipoint recorder. Linear calibrations of 2.42×10^{-9} μc/ml-cpm and 1.21×10^{-9} μc/ml-cpm were obtained for the detection of argon-41 gas using photopeak counting and integral counting, respectively. A sensitivity of 6×10^{-8} μc/ml corresponding to 25 net counts per minute, which is twice background, was observed for photopeak counting of argon-41. (auth)

391 (DP-908) PROTOTYPE MONITOR FOR AIRBORNE IODINE AND FISSION PRODUCTS. Joseph W. Adams (Du Pont de Nemours (E. I.) & Co. Savannah River Lab., Aiken, S. C.). July 1964. Contract AT(07-2)-1. 8p. Dep.; $1.00(cy), 1(mn) OTS.

The design is presented for a new system for monitoring gases from the dissolution of irradiated uranium in radiochemical separation processes at the Savannah River Plant. The system includes filter paper and activated charcoal to trap particulate activity and iodine vapor; as activity builds up, it is detected by a concentrically mounted G-M tube. Early operating experience is also discussed.

392 (LA-3363) ^{131}I SAMPLING WITH ACTIVATED CHARCOAL AND CHARCOAL-IMPREGNATED FILTER PAPER. Ettinger, H. J.; Dummer, J. E. Jr. (Los Alamos Scientific Lab., Univ. of California, N. Mex.). Aug. 1965. Contract W-7405-eng-36. 18p. Dep. mn; CFSTI $1.00 cy, $0.50 mn.

Samples of stack effluent from an operating reactor, and from a hot cell complex in which spent reactor fuel elements are examined, showed activated charcoal efficiencies ranging from 70–99%, with a median efficiency of 90%. Three different charcoal-impregnated filter papers were unsatisfactory for monitoring radioiodine because their collection efficiencies varied widely. Both impregnated filter papers and activated charcoal cartridges were tested at various sampling velocities. The effects of methyl chloroform on the iodine collection efficiency of activated charcoal were evaluated. The collection efficiencies of charcoal cartridges soaked in methyl chloroform ranged from 72–90%, with a median of 81%. Concurrent sampling with charcoal cartridges not treated with methyl chloroform showed an efficiency range of 71 – 95%, with a median of 86%. Field tests indicate the need for carefully considering the iodine collection efficiency of activated charcoal for air cleaning systems and respirators.

393 (TID-7677(p.317-30)) LIMITS OF SENSITIVITY IN THE MONITORING OF RADIOACTIVE GASES, WITH PARTICULAR REFERENCE TO KRYPTON 85. J. Shapiro, R. E. Yoder, and L. Silverman (Harvard Univ., Boston. School of Public Health).

Calculations of efficiencies of idealized ionization chambers and particle detectors for monitoring radioactive gases are presented. Kr^{85} was used as the radioactive gas in the calculations and measurements. It is shown that for levels of the order of 10^{-7} μC/cc, a simple cylindrical G-M counter is adequate if a large volume can be viewed, and it is the simplest detector available. (M.C.G.)

394 A LARGE-VOLUME, CONTINUOUS GAS-EFFLUENT MONITORING SYSTEM. Donald E. Feltz (Texas A & M, College Station). Trans. Am. Nucl. Soc., 6: Suppl., 52(Oct. 1963).

395 RECENT ADVANCES IN MONITORING FOR AIRBORNE RADIOACTIVITY. Lister, B. A. J. pp 375-84 of Radiological Monitoring of the Environment. Godbold, B. C.; Jones, J. K. (eds.). Oxford, Pergamon Press, 1965.

Recent advances in the collection and analysis of radioactivity in the air are reviewed and the interpretation of data in terms of hazards to man are discussed. Emphasis is placed on the development of sampling equipment for use in operating areas in plants using Pu and U in the environs of reactors, and in stock sampling.

Separation, General

See also the following references:

34, 268

396 SEPARATION OF RADIOIODINE VAPORS FROM AIR. W. Ullmann (Institut für Staubforschung und Radioaktive Schwebstoffe, Berlin). Symp. Tech. Zagadnien Ochrony Promieniowaniem, Warsaw, 327-37(1962). (In German)

The construction of apparatus for the measurement of the separation efficiency and the flow resistance of iodine-131 filters is described in detail. The devices and the measurement methods used permit the investigation of iodine filters at very low iodine vapor concentrations and different operating conditions. The measurement results obtained for the separator agents used (5 different activated charcoals, silica gel, and fillers of silver and silver nitrate) are reported. Conclusions on the practical utilization of iodine filters are given on the basis of economic considerations. (tr-auth)

Separation by Adsorption

See also the following references:

19, 44, 48, 95, 163, 208, 215, 219, 367, 373, 392, 396,

425, 485, 498, 950

397 (AEC-tr-6459) STUDY OF THE TRAPPING OF IODINE BY ACTIVE CHARCOALS. F. Billard, G. Chevalier, and J. Pradel (France. Commissariat à l'Énergie Atomique. Centre d'Études Nucléaires, Saclay). Translated by W. S. Durant (Savannah River Lab., Aiken, S. C.), from report CEA-2237. 1962. 19p.

The efficiency of an active charcoal toward iodine-131 has been studied during a series of experiments. The influence of the iodine concentration of the air current, the rate of flow of air across the charcoal, and the charcoal thickness has been determined. The results obtained lead to the conclusion that the iodine, after passage across a so-called

"absolute" filter, is present in two different forms, and that the action of the active charcoals is different for the two forms. These results make it possible, however, to calculate industrial filters with a good degree of approximation. (auth)

398 (CEA-2237) ÉTUDE DE L'ARRÊT DE L'IODE PAR LES CHARBONS ACTIFS. (Study of the Trapping of Iodine by Active Charcoals). J. Pradel, G. Chevalier, and F. Billard (France. Commissariat à l'Énergie Atomique. Centre d'Études Nucléaires, Saclay). 1962. 40p.

The efficiency of active charcoal towards I^{131} was studied during a series of experiments. The influence of: the iodine concentration of the air current, the rate of flow of air across the charcoal, and the charcoal thickness, was determined. The results obtained lead to the conclusion that the iodine, after passage across a so-called "absolute" filter, is present in two different physical forms, and that the action of the active charcoals is different for the two forms. These results make it possible, however, to calculate industrial filters with a good degree of approximation. (auth)

399 (CONF-12-3) REMOVAL OF RADIOIODINE VAPOR FROM GASES AT ELEVATED TEMPERATURES. Robert E. Adams and William E. Browning, Jr. (Oak Ridge National Lab., Tenn.). [nd]. Contract [W-7405-eng-26]. 13p.

From the American Nuclear Society 8th Annual Meeting, Boston, June 1962.

After a survey of methods previously used for iodine removal from gas streams, it was concluded that a system employing activated charcoal would more completely satisfy the requirements of an emergency ventilation system then other methods. Iodine vapor removal from a gas stream of steam and air at 80 to 100°C was studied. The presence of water vapor was found to have a slightly deleterious effect on the iodine adsorption efficiency of charcoal. Increased temperature also lowered the iodine adsorption efficiency slightly. However, a high removal efficiency was observed for periods up to 2.75 hr and with gas velocities less than 75 ft/min. Tests of charcoal efficiency for removing iodine vapor from helium streams showed good results for temperatures as high as 430°C for up to 3 hr, but continued helium flow was found to slowly remove the iodine contained in the charcoal bed. Various types of charcoal exhibited variations in iodine removal and retention. (M.C.G.)

400 RESULTS OBTAINED IN RESEARCH ON CHARACTERISTICS OF TRAPPING OF IODINE-131 BY ACTIVATED CHARCOAL. F. Billard, G. Chevalier, J. Pradel (Commissariat à l'Énergie Atomique, Paris). 14p. (CONF-337-55).

From International Conference on Radioactive Pollution of Gaseous Media, Saclay, France, Nov. 1963.

The effects of the physical characteristics of activated charcoal on its capacity to trap ^{131}I were studied. The devices and the techniques used are described. The efficiency of the charcoal was independent of the iodine concentration between 0.1 mCi and 3 Ci. The logarithm of the efficiency of a filtering carbon layer was approximately proportional to its thickness, but the concentration in the last centimeter of the bed was slightly superior to the theoretical values. The coefficient of total purification was approximately inversely proportional to the air velocity. Some incomplete results on the effects of temperature, grain size, activated rate, carbon humidity, and foreign bodies are reported. (J.S.R.)

401 (CEA-R-2908) INFLUENCE DE L'HUMIDITE ET DE LA CONCENTRATION EN IODE SUR L'ADSORPTION DE L'IODE 131 PAR LES CHARBONS ACTIFS. (Influence of Humidity and Iodine Concentration on the Adsorption of Iodine-131 by Activated Charcoals). Billard, Francois; Chevalier, Gerard; Caron, Jacques; Van Der Meersch, Jean (Commissariat a l'Energie Atomique, Fontenay-aux-Roses (France). Centre d'Etudes Nucleaires). Jan. 1966. 23p. (CONF-650407-14). Dep. mn.

From International Symposium on Fission Product Release and Transport under Accident Conditions, Oak Ridge, Tenn.

Following an outline of the means of generating iodine-131 in the gas phase and a brief description of the apparatus used, experimental results obtained on the trapping of iodine-131 by activated charcoal were analyzed. The generation of ^{131}I in the gas phase is outlined, and the apparatus used is described. The efficiency depends concurrently on the relative humidity and iodine concentration of the air. While it is independent of this concentration in dry air, it does not exceed 50 in moist air, whatever the bed depth, when the concentration is less than 1 mC/m^3, then it increases with the concentrations to reach values close to those obtained in dry air for concentrations of around 1000 C/m^3. At the same time the efficiency of a dust filter with respect to iodine remains very low in dry air whatever the iodine concentration; in moist air this efficiency, which is high at low concentrations, decreases when the latter increases. Also, if a charcoal bed is divided, separated by an amount corresponding to a passage time of about a second, it was found that the efficiency of the whole is appreciably higher than that of the initial bed. The results appear to establish the existence of at least two forms of iodine, one molecular form and one particulate form probably formed by clusters, the forms' equilibrium being functions of the relative concentrations of iodine and water vapor in the air. The third form evidence, a gaseous form compound, remains hypothetical; the experimental curves can be explained either by the molecular and particular forms' reversibility or by the increase of cluster size with the relative humidity of the air.

402 (CONF-650407, pp 814-29) THE INFLUENCE OF PHYSICAL PARAMETERS ON THE ADSORPTION OF ^{131}I BY ACTIVATED CHARCOALS. Billard, F.; Caron, J.; Chevalier, G.; Van Der Meersch, J. (Commissariat a l'Energie Atomique, Fontenay-aux-Roses (France). Centre d'Etudes Nucleaires).

After an outline of the means of generating iodine-131 in the gas phase and a brief description of the apparatus used, the experimental results obtained on the trapping of iodine-131 by activated charcoal are analyzed. The efficiency depends concurrently on the relative humidity and the iodine concentration of the air. While it is independent of this concentration in dry air, it does not exceed 50 in moist air, whatever the bed depth. When the concentration is less than 1 mCi/m^3, it increases with the concentrations to reach values close to those obtained in dry air for concentrations of around 1000 Ci/m^3. At the same time, the efficiency of a dust filter with respect to iodine remains very low in dry air whatever the iodine concentration; in moist air this efficiency, which is high at low concentrations, decreases when the latter increases. In addition, if a charcoal bed is divided into several beds, separated by an amount corresponding to a passage time of about a second, it is found that the efficiency of the whole is appreciably higher than that of the initial bed. These results appear to establish the existence of at least two forms of iodine, one molecular form and one particulate form probably formed

by clusters, the equilibrium forms being functions of the relative concentrations of iodine and water vapor in the air. The third form, a gaseous compound, remains hypothetical; the experimental curves can be explained either by the molecular and particulate forms reversibility or by the increase of clusters size with the relative humidity of the air.

403 (CONF–650407, pp 853-68) THE REMOVAL OF METHYL IODIDE FROM REACTOR EFFLUENT GASES. Collins, D. A.; Taylor, R.; Taylor, L. R. (Reactor Group, United Kingdom Atomic Energy Authority, Risley (England)). (TRG-Report–982).

When fission product iodine is released from uranium dioxide fuel elements into carbon dioxide or into steam, some of it is found in the form of methyl iodide. Methyl iodide is relatively unreactive and will penetrate many of the trapping materials that can be used for elemental iodine, but it can be held on activated carbon if the conditions are right. The removal of methyl iodide from dry air or carbon dioxide in the range from room temperature to 100°C can be accomplished without difficulty by a coal-based carbon, provided a certain loading of methyl iodide on the carbon is not exceeded. A coconut-based carbon behaves similarly at 100°C, although breakthrough occurs at a lower loading. At room temperature, however, unacceptable breakthrough occurs at all loadings on coconut carbon after only a short delay due to physical adsorption. Retention on both coconut and coal-based carbons is seriously affected by high humidity, and a number of commercially available carbons have been tested without finding one which gives satisfactory performance in these conditions. Performance can be restored by preheating the humid gas stream so that the relative humidity is reduced below 30% at the carbon, but this is not a particularly attractive solution. A method was found for chemically impregnating coal-based carbon so that it continues to exhibit excellent retention of methyl iodide in air at 100% relative humidity. Work is in progress to define the operating limits of this type of charcoal.

404 (CONF–650407, pp 869-84) REMOVAL OF IODINE AND VOLATILE IODINE COMPOUNDS FROM AIR SYSTEMS BY ACTIVATED CHARCOAL. Adams, Robert E.; Browning, William E. Jr. (Oak Ridge National Lab., Tenn.).

The behavior of methyl iodide and iodine vapor against four types of activated charcoal under varying degrees of temperature and humidity was investigated. The charcoals were selected to represent four types of this material: bituminous coke, petroleum residue, coconut shell, and a metal-impregnated bituminous coke. Under dry conditions (relative humidity <3%) and at room temperature, all the charcoal types are effective against methyl iodide and iodine vapor. Under wet conditions (relative humidity approximately 75%) and at room temperature, all the charcoal types were effective against elemental iodine but relatively ineffective against methyl iodide. The efficiency for methyl iodide retention ranged from 50 to 97%, with coconut charcoal producing the higher efficiency. Under low humidity and high temperature (100 or 150°C), all charcoal types had high efficiencies for retention of both elemental iodine and methyl iodide. Under steam conditions, at temperatures ranging from 95 to 130°C, the efficiency of the charcoals for methyl iodide is quite low, ranging from 5 to 78%. The efficiency for elemental iodine is also reduced slightly under these conditions, averaging about 98%.

405 (DP–1028) PERFORMANCE OF ACTIVATED CARBON BEDS IN SRP REACTOR CONFINEMENT FACILITIES. Progress Report, September 1962–September 1965. Durant, William S. (Du Pont de Nemours (E. I.) and Co., Aiken, S. C. Savannah River Lab.). Jan. 1966. Contract AT(07-2)-1. 27p. Dep. mn. CFSTI $2.00 cy, $0.50 mn.

The efficiency of activated C beds for removal of elemental iodine remains greater than the design efficiency of 99.9% after ~2 years of continuous exposure to exhaust ventilation air from Savannah River Plant reactor buildings. Because of corrosion of both the Cd-plated and the painted mild steel frames that contain the C, replacement units are being fabricated of Type 304L stainless steel.

406 (HW-81746(p.4.72-3)) EFFICIENCY OF CHARCOAL FOR RADIOIODINE ABSORPTION. J. D. McCormack (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.).

The ^{131}I retention of charcoal deteriorates materially over a period of 60 days when used on a separations plant gaseous stream. Studies on the stream and laboratory studies using tracer ^{131}I showed: several organic constituents were present in the stream, hexone present didn't materially influence the elution of iodine, and methyl iodide at levels of 50 to 100 $\mu g/g$ of charcoal is rather easily eluted by passing air through the charcoal. The need for carefully evaluating the efficiency of charcoal under the conditions to be imposed was emphasized by these studies.

407 CHARCOAL TRAP WITH RAPID THERMAL RESPONSE. Don O. Coffin (Los Alamos Scientific Lab., N. Mex.). Rev. Sci. Instr., 35: 1534-5(Nov. 1964). (LADC-6486)

A charcoal trap for separation of gases by selective absorption is described. Unique features claimed for the design are operation to 300°C without auxiliary heaters, rapid thermal cycling over a 500°C range, and modular construction adaptable to a wide range of capacities. (auth)

408 (ORNL-3776(p.43-6)) REMOVAL OF IODINE AND VOLATILE IODINE COMPOUNDS FROM AIR SYSTEMS BY ACTIVATED CHARCOAL. W. E. Browning, Jr., R. E. Adams, and F. V. Hensley, Jr. (Oak Ridge National Lab., Tenn.).

The behavior of methyl iodide and iodine vapor toward four types of activated charcoal under varying degrees of temperature and humidity was investigated. The charcoals were selected to represent four types of this material: bituminous coke, petroleum residue, coconut shell, and a metal-impregnated bituminous coke. Under dry conditions (relative humidity <3%) and at room temperature, all the charcoal types are effective against methyl iodide and iodine vapor. Under wet conditions (relative humdity approximately 75%) and at room temperature, all the charcoal types were effective against elemental iodine but relatively ineffective against methyl iodide. The efficiency for methyl iodide retention ranged from 50 to 97%, with coconut charcoal producing the higher efficiency. Under low humidity and high temperature (100 to 150°C), all charcoal types had high efficiencies for retention of both elemental iodine and methyl iodide. Under steam conditions, at temperatures ranging from 95 to 130°C, the efficiency of the charcoals for methyl iodide is quite low, ranging from 5 to 78%. The efficiency for elementary iodine is also reduced slightly under these conditions, averaging probably about 98%. These studies are continuing.

409 (ORNL–3840) LARGE SCALE PREPARATION OF HIGH PURITY ^{131}I AND ^{133}Xe BY SORPTION TECHNIQUES. Case, F. N.; Acree, E. H. (Oak Ridge National Lab., Tenn.). Jan. 1966. Contract W-7405-eng-26. 15p. Dep. mn. CFSTI $1.00 cy, $0.50 mn.

Isolation of ^{131}I and ^{133}Xe from fission product mixtures was accomplished by sorbing the ^{131}I on platinum and the ^{133}Xe on charcoal or on a molecular sieve. Very high radiochemical purity is obtained by this method. The process is easily controlled to minimize the loss of ^{131}I to waste gas streams and compact equipment allows manipulator cell operation.

410 (SC-T-64-613) CONCERNING THE ADSORPTION OF RADON ON ACTIVATED CHARCOAL AND SILICA-GEL. O. Guebeli and K. Stammbach. Translated by Marcel I. Weinreich (Sandia Corp., Albuquerque, N. Mex.), from Helv. Chim. Acta, 34: 1257-43(1951). 9p.

The mixed adsorption of radium emanation (radon) with air as a carrier gas for the adsorbents silica gel and linden wood charcoal was investigated. At equal temperatures, the linden wood charcoal proves to be considerably more active than silica gel. The experimental data allow one to calculate the isosteric adsorption heats for both adsorption agents. (auth)

411 (SC-T-64-616) CONCERNING THE MIXED ADSORPTION OF RADON ON ACTIVATED CHARCOAL WITH VARIOUS CARRIER GASES. O. Guebeli and M. Stoeri. Translated by Marcel I. Weinreich (Sandia Corp., Albuquerque, N. Mex.), from Helv. Chim. Acta, 37: 2224-30(1954). 9p.

Radon adsorption on wood charcoal was investigated with respect to its dependence on various carrier gases. The isosteric adsorption heats were ascertained mathematically from the experimentally determined adsorption data, as far as radon is concerned, which was accompanied by foreign gases such as air, nitrogen and carbon dioxide, including hydrogen. (auth)

412 (SGAE-PH-31/1965) ANWENDUNG DER RADIOAKTIVEN TRACERTECHNIK IM HOCHVAKUUM ZUR KONSTRUKTION EINES MASSENSEPARATORKOLLEKTORS FUER XENON. **(Application of Radioactive Tracer Technique in High Vacuum for Construction of Mass Separator Collectors for Xenon). Dobrozemsky, R.; Hechtl, E.; Viehboeck, F. P. (Oesterreichische Studiengesellschaft fuer Atomenergie G.m.b.H., Seibersdorf. Physik-Institut).** [1965]. 11p. Dep. mn.

A mass separator collector was designed and constructed for the collection of xenon. ^{131}Xe was used in trace amounts for the determination of xenon adsorbed on the activated charcoal and the equilibrium time. The method used is described in some detail and a description is given of the collector installation.

413 (TID-7677(p.102-24)) EXPERIENCE IN TRAPPING IODINE-131 AND OTHER FISSION PRODUCTS RELEASED FROM IRRADIATED AGR-TYPE FUEL ELEMENTS. D. A. Collins, R. Taylor, and W. D. Yuille (Gt. Brit. National Inst. for Research in Nuclear Science. Rutherford High Energy Lab., Harwell, Berks, England).

I^{131} and other fission products were released from irradiated stainless steel-clad uranium dioxide under conditions relevant to potential accidents in the Advanced Gas-Cooled Reactor. Measurements were made of the penetration of I^{131}, Te^{132}, Cs^{137}, and Sr^{90}, released in the above manner, through a trapping system consisting of an absolute filter, followed by a 4-in. deep bed of charcoal and a final absolute filter, in series. No significant penetrations were observed except in the case of I^{131} in the form of alkyl iodides. Some small fraction of the iodine released from the irradiated fuel is in this form and investigations showed some conditions in which this form can be produced from iodine initially present in other forms. Experiments with macroscopic amounts of methyl iodide, representing the above form of iodine, showed that it is only delayed on activated charcoal at temperatures up to 100°C, but part of it can be permanently retained at 200°C. Oxidizing catalysts showed some promise of effectively retaining methyl iodide, but the presence of carbon monoxide interfered. (auth)

414 (TID-7677(p.163-76)) ANALYSIS OF A POSSIBLE INFLUENCE OF PARTICLE SIZE ON THE IODINE REMOVAL EFFICIENCY OF A CHARCOAL FILTER. L. A. Haack (Oak Ridge National Lab., Tenn.).

It is shown that to describe properly the efficiency of a filter in removing contaminants, especially I^{131}, from an air stream not only must the amount of impurities be specified but also that the size distribution of particles in the air stream must be specified. Calculations and comparisons are presented. (M.C.G.)

415 (TID-7677(p.248-58)) THE APPLICATION OF IODINE ABSORBER UNITS TO THE HWCTR CONTAINMENT SYSTEM. L. M. Arnett and G. C. Rusche (Du Pont de Nemours (E. I.) & Co. Savannah River Lab., Aiken, S. C.).

An analysis was made of the extent to which the installation of iodine absorber units in the containment building of the Heavy Water Components Test Reactor (HWCTR) can be expected to reduce radiation exposures of personnel in the vicinity of the facility during a release of fission products from the reactor core. Because the radioiodines contribute substantially to the whole-body dose and constitute almost entirely the thyroid dose, any mechanism that eliminates these isotopes from the materials escaping from the building reduces considerably the radiation hazard outside the building. This evaluation of the maximum credible accident revealed that the operation of efficient iodine absorbers reduces the radiation hazards to commonly acceptable levels even though the leak rates from the containment building are relatively high. At leak rates in the vicinity of 2% of the building contents per day, the whole-body dose and the thyroid dose received by personnel at the Plant boundary three miles from the HWCTR would be less than 25 r and 30 rem, respectively, in 30 days. The effect of the iodine absorbers was to reduce the expected whole-body dose by a factor of two and the expected thyroid dose by a factor of one hundred. (auth)

416 **(TID-7677(p.545-52)) PRELIMINARY DESIGN CONSIDERATIONS— EFFLUENT CLEANING SYSTEM FOR THE 630 A GAS-COOLED MARITIME REACTOR. J. E. Maxwell (General Electric Co. Nuclear Materials and Propulsion Operation, Idaho Falls, Idaho).**

A system for treatment of radioactively contaminated air from an air-cooled nuclear steam generator for maritime service before its release to the atmosphere is being developed. It is planned that any air passing to the atmosphere from the primary coolant loop will flow through charcoal adsorption beds. (M.C.G.)

417 (TID-7688(p.68-74)) EVALUATION OF CHARCOAL AND OTHER ADSORBENTS FOR REMOVING RADIOIODINE FROM HOT OFF-GAS SYSTEMS. W. Cox (Oak Ridge National Lab., Tenn.).

A four-stage charcoal column installed in the ORNL Isotopes Center hot off-gas system, was used to study the efficiency of I^{131} removal. The column showed an average total efficiency of 98.9% and an average individual stage efficiency of 75%. The particulate size of

the iodine was determined to be in the range of 60 to 100 A. The iodine adsorption characteristics of several types of charcoal and of Berl saddles coated with silver nitrate were studied. The effects of sample line, temperature, and very low I^{131} concentration on the adsorption were also studied. (D.L.C.)

418 (TRG-Report-888) A HIGH-ACTIVITY IODINE RELEASE FROM IRRADIATED FUEL AND ITS SUBSEQUENT CLEAN-UP. Hillary, J. J.; Taylor, J. C. (Reactor Group, United Kingdom Atomic Energy Authority, Sellafield (England)). Mar. 23, 1965. 18p. Dep.; BIS $0.60; HMSO 3s.

A test at kilocurie level of activity was performed, based on the most stringent conditions postulated in reactors of the advanced gas-cooled type and clean-up effected with a filter-charcoal bed-filter system. Iodine penetration of the treatment system was greater than previously experienced as a result of excessive loading of the charcoal bed by methyl iodide, produced from the iodine released by the fuel. Another type of charcoal was found to give satisfactory retention of methyl iodide.

419 INVESTIGATION OF RADIOACTIVE IODINE SORPTION BY ACTIVATED CARBON AND STUDIES OF SHAPES OF GASEOUS IODINE IN THE AIR. T. N. Smolkina and A. A. Chubakov. At. Energ. (USSR), 18: 298-9(Mar. 1965). (In Russian)

Sorption of gaseous iodine emitted with the air current from a fuel element heated at 900 to 1000°C by BAU carbon at an air current rate of 20 cm/sec was investigated. The fuel element was prepared with uranium enriched beryllium oxide without anti-fragment coating. The concentration of radioactive iodine was 10^{-9} to 10^{-11} C/l. The graphs of gaseous iodine concentration along the BAU carbon showed that 10 cm of BAU layer could reduce the concentration of iodine in the air about 500 times. The experimental data revealed the presence of three forms of iodine: elementary iodine, iodides, and iodates, and possibly other mixtures with positive valence iodine. With the fuel element temperature increase the percentage of iodides in the iodine vapor mixture slowly decreased; the percentage of the third form of iodine vapors first dropped down then reached a constant level at ~0.5 to 1%. The largest amount of iodine vapor mixture was observed at 100 to 200°C (75%) with 25% of the third form.

420 PHYSICAL CHEMISTRY OF IODINE AND REMOVAL OF IODINE FROM GAS STREAMS (REVIEW OF WORK IN AEROSOL GROUP, A.E.R.E., HARWELL). A. C. Chamberlain, A. E. J. Eggleton, W. J. Megaw, and J. B. Morris (Atomic Energy Research Establishment, Harwell, Berks, Eng.) J Nucl. Energy, Pt. A & B, 17: 519-50 (Dec. 1963)

The work of the Aerosol Group at Harwell on the removal of iodine from gas streams is reviewed. Work of other groups is mentioned only where it reflects directly on Harwell work. The main features of the work are: (a) Tests of the behaviour of iodine released into reactor containment vessels show that rapid sorption on the walls occurred, and that the iodine remaining airborne after periods of order an hour was in nonelemental form. (b) Laboratory and pilot plant studies show that the sorption of iodine on particulate matter and the formation of nonelemental gaseous compounds occurred with varying methods of release of iodine and in CO_2 as well as air. (c) Tests of iodine absorption in charcoal beds, and tests of the ignition temperature of some charcoals were made to enable the efficient removal systems to be specified. (d) Investigations of the physical chemistry of iodine at low gaseous concentrations enabled one of the nonelemental forms to be identified as methyl iodide. (auth)

421 DEVELOPMENT OF A PRODUCTION PROCESS FOR RADIO-KRYPTON RECOVERY BY FRACTIONAL ABSORPTION. R. F. Taylor and G. P. Wall (Atomic Energy Research Establishment, Harwell, Berks, Eng.). Progr. Nucl. Energy, Ser. IV, 5: 307-46(1963).

A description of the arisings of fission product krypton in gaseous effluents from the treatment of irradiated fuels is followed by a review of the principal methods of krypton recovery, based on adsorption on charcoal beds, low-temperature liquefaction and distillation, and absorption in various liquid solvents, together with grounds for choosing the McIlroy solvent absorption process for eliminating the bulk diluent gases. The separate adsorption process on charcoal beds that would be used for final enrichment of krypton is not discussed. The nature of gas feeds is discussed with emphasis on the types of impurity likely to be present and the pretreatment thereby entailed. The development of the selected processing scheme for the large-scale recovery of krypton from dilute feeds is presented. The problems and techniques of dealing with the recycling of carbon tetrachloride solvent of the absorber system are treated separately. The central features of the process design, the analysis and prediction of absorption behavior are discussed. A pilot plant was operated to extend the experimental data of McIlroy to large-scale plant working at high flow velocities, to longer columns of higher contacting efficiency, and to gas feeds of a variety of compositions. The plant and experimental techniques used are described and the experimental results are presented and analyzed in terms of stage computations. The assumption of McIlroy's computation method that all components have the same height is shown to be not a reasonable one. The consequent errors in predicted krypton concentrations are not necessarily excessive but the corresponding concentrations for other components are often seriously in error and unacceptable for design purposes. Simple empirical modifications are introduced so that computation proceeds in terms of krypton stages that can be associated with definite locations in a column. At these stages, or any intermediate level, the correct concentrations of the other components are obtained by interpolations based on the experimental data given herein. The design of absorber columns is outlined. Scale-up effects and pressure drops are discussed and calculations are given of column lengths required for typical separation duties. The main features of the process design based on the pilot plant work are followed by application of the analysis of the experimental data to bring out important points of production plant operation. The scope of the design and cost estimating study and observations made on some factors influencing recovery costs are described. (auth)

422 EFFECTS OF THE GAS MACROCOMPONENT ON THE DISTRIBUTION OF ^{85}Kr AND ^{133}Xe BETWEEN GASEOUS PHASE AND SOLID CARBON SORBENT. Trofimov, A. M.; Pankov, A. M. Radiokhimiya, 7: 293-8 (1965). (In Russian).

The temperature dependence of the absorption coefficients of ^{85}Kr and ^{133}Xe on different activated carbons from air, He, Ar, and CO_2 was investigated. The temperature limits, in which the linear rule of the distribution on gas microcomponents operates, were established. For example, with absorption on activated carbon from carbon dioxide

this rule operated at temperatures above 263°K, but from Ar and air it was valid above 230°K. The heats of absorption of microquantities of Kr and Xe on different activated carbons from He, Ar, air, and CO_2 were calculated. The absorption isobars of gas macrocomponents, He, Ar, air, and CO_2, on the activated carbon were determined.

423 REMOVAL OF FISSION-PRODUCT NOBLE GASES FROM THE HELIUM OF A HIGH-TEMPERATURE GAS-COOLED REACTOR USING CHARCOAL AT LOW TEMPERATURE. H. de Bruijn (Atomic Energy Establishment, Winfrith, Dorset, Eng.), A. Gruetter, E. Roemberg, J. C. Shorrock, L. Werner, and F. C. Wood. Trans. Inst. Chem. Engrs. (London), 42: T365-86(1964).

A study was made of the removal of fission-product noble gases from helium by low-temperature adsorption on charcoal in support of the development of the O.E.C.D. High-Temperature Reactor Dragon Project. Results of experiments on the breakthrough of krypton, xenon, and other single gaseous impurities in helium carrier-gas streams on both laboratory and semi-technical scales are presented. It was found that gas-pore diffusion governs the rate of breakthrough in the former, whereas external fluid-phase diffusion appears to predominate in the latter during the early stages of breakthrough. Rate determinations confirmed the applicability of gas-pore diffusional control in the absence of helium carrier gas. Static and dynamic equilibrium data were obtained for both xenon and krypton on charcoal at low temperatures. Good agreement was obtained between the results, which can also be fitted to the Dubinin-Polanyi thermodynamic adsorption concept. It is considered probable that a "characteristic curve" can be constructed for a particular adsorbent, which can be used to predict adsorption equilibrium data for other gases for which experimental data are not available. It was concluded that sufficient margin exists in the present design of the fission-product removal plant of Dragon to permit considerable optimization for commercial development.

424 (SC-T-64-617) ON THE RADON ADSORPTION WITH ACTIVATED CHARCOAL IN THE OPEN SYSTEM OF THE FREE GAS FLOW. O. Guebeli and M. Stoeri. Translated by Marcel I. Weinreich (Sandia Corp., Albuquerque, N. Mex.), from Helv. Chim. Acta, 38: 180-4(1955). 8p.

The adsorption of radon on activated charcoal was investigated by a dynamic method of free flow of gas through the system, utilizing distribution curves. Also the desorption of radium emanation in a nitrogen flux was examined for a maple wood charcoal of medium activity, using the quantitative approach. (P.C.H.)

Separation by Centrifugation

See the following reference:

54

Separation by Foaming Agents

See the following references:

388, 428

Separation by Liquefication and Distillation

See the following references:

73, 421

Separation by Metal Reactors (Ag and Cu)

See also the following references:

19, 170, 208, 396

425 (ORNL-3591(p.185-90)) BEHAVIOR OF RADIOIODINE. (Oak Ridge National Lab., Tenn.).

An investigation was made of the origin, identity, and behavior of iodine compounds which appear when air and iodine vapor are mixed. Information is needed on these points in order to design radioiodine removal systems. Rates of deposition of iodine compounds were measured by use of successive diffusion tubes having surfaces of silver, rubber, and activated carbon. Elemental iodine deposited on the silver with its characteristic diffusion coefficient of about 0.08 cm^2/sec; a compound or group of compounds deposited on the rubber to give an indicated diffusion coefficient of about 0.05 cm^2/sec, suggestive of a molecular weight of the order of 300 or higher; other iodine materials deposited on the activated carbon to yield a diffusion coefficient of 0.10 cm^2/sec, implying a molecular weight somewhat lower than that of I_2. This behavior was typical of iodine from two greatly differing sources. As the iodine concentration in air decreased, the proportion of iodine present in forms other than molecular iodine increased. The sorption characteristics of fission products are of significance in predicting their deposition behavior in flowing gas systems. The adsorption of iodine on commonly used stainless steels was explored with metal temperatures ranging from 24 to 700°C and with iodine vapor concentrations varying from 0.0004 to 2200 mg/m^3. Iodine vapor was found to react chemically with stainless steel surfaces to form metal iodides. The capacity of the surface varied with the condition of the adsorbing surface. Passivity towards iodine adsorption was frequently observed after a small amount of material was sorbed. The kinetics of the chemisorption process also varied, apparently with changing surface conditions, even during the course of a single experiment. The iodine sources used were analyzed for forms other than I_2. Methyl iodide was found, but it constituted less than 0.1% of the total iodine. Fractional sublimation of the iodine sources in vacuum ensured that only I_2 was used in the adsorption experiments. Charcoal traps remove molecular iodine from gases with a high degree of efficiency, but they are susceptible to destruction when exposed to oxidizing gases at high temperatures, conditions which could conceivably exist in a reactor accident. Preliminary experiments to explore alternate trapping materials indicated that a bed of platinized alumina (a hydroforming catalyst) would serve only to delay the passage of iodine when the bed is exposed to flowing air at temperatures above 400°C. Trapping of radioiodine by high-temperature exchange with nonradioactive iodine in a bed of potassium iodide appeared to be feasible up

to temperatures at which significant oxidation of the inorganic iodine occurs (600°C or higher). (auth)

426 IMPROVEMENTS RELATING TO RADIOACTIVE GAS DECONTAMINATION. John Francis William Smith and Ralph Hugh Alfred Crawley (to English Electric Co. Ltd.). British Patent 985,550. Mar. 10, 1965. Filed July 21, 1960.

A means for removing radioactive iodine from a flowing gas includes a multi-layer mesh of silver-coated copper continuous wire filaments woven together in the flow path of the gas. The silver may be deposited on the copper by electroplating or by direct immersion process. The mesh is particularly applicable to use in the effluent gas duct of a gas cooled reactor. The knitted or woven mesh is stacked in substantially parallel layers in the duct compressing the layers of the bed so that its depth is about 9 in. (B.L.M.)

Separation by Scrubbing

See also the following references:

19, 185, 208, 960

427 (HW-55735) REMOVAL OF RADIO-IODINE FROM PUREX OFF-GAS WITH NITRIC ACID AND WITH NITRIC ACID-MERCURIC NITRATE SOLUTIONS. R. W. Stromatt (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Apr. 15, 1958. Contract AT(45-1)-1350. 10p.

The comparison of the removal of iodine vapor from a gas stream by nitric acid solutions with and without mercuric ion was studied under the conditions obtained when the iodine concentration in the gas stream, the flow rate of the gas stream, and the concentration of mercury in the scrub solutions were varied. Also this removal was studied with and without nitric oxide in the gas stream, at the acid concentration limits expected in the scrub solution, and at 40°C and at room temperature. The greater the concentration of mercury in the scrub solution, the less effect the flow rate had upon iodine removal. However, at iodine concentrations approaching the concentration expected in the plant gas stream, the flow rate did not effect the iodine removal by scrub solutions in the absence of mercury, at least over the range investigated. At high iodine concentrations, the mercury scrub solution appeared more effective at 40° than at room temperature, while the opposite appeared true in the absence of mercury. At low iodine concentrations, there was little difference in the iodine removal by any of the scrub solutions at the two temperatures. Little difference was found in iodine removal by any of the scrub solutions at concentrations of five and ten percent nitric acid. The presence or absence of nitric oxide in the gas did not noticeably alter the iodine removal by the scrub solutions. The mercury solutions were effective at high iodine concentrations if the mercury were in excess over the iodine. At low concentrations of iodine, the acid solution alone was as effective as the mercury solution in removing the iodine from the gas. An air sparge of nitric acid solutions of iodine showed greater than 99.5 percent retention of iodine if the iodine was less than 10^{-6} M. For the mercuric solution to be effective in removing iodine from the gas and retaining it, the molarity of the mercury should be greater than the iodine molarity (as I_2) in the scrub solution. This indicated that higher iodine complexes, such as the HgI_4^{2-} complex ion, were not very stable in the system studied. However, nitric acid solutions without mercury were about as effective as with mercury in the concentration range of iodine in the plant gas stream. Therefore, there appears to be little advantage to the use of mercury except, perhaps, to control large bursts of iodine. (auth)

428 (RL-SA-50) REMOVING FISSION GASES AND IODINE FROM REACTOR EXHAUST AIR BY FOG SPRAY AND FOAM. Smith, M. L. (General Electric Co., Richland, Wash. Hanford Atomic Products Operation). Aug. 2, 1965. Contract AT(45-1)-1350. 3p. (CONF-651101-35). Dep. mn; CFSTI $1.00 cy, $0.50 mn.

From 13th Conference on Remote Systems Technology, Washington, D. C.

Two tests were conducted on the ability of a fog spray system to remove ^{128}I from the exhaust air at one of the Hanford production reactors. The penetrations of 100-Curie releases of ^{128}I into the components of the system are given. A test was also conducted on the trapping of ^{41}Ar from the exhaust air by foam.

429 REMOVAL OF IODINE FROM THE ATMOSPHERE BY CONDENSING STEAM. Stinchcombe, R. A.; Goldsmith, P. (Atomic Energy Research Establishment, Harwell, Eng.). J. Nucl. Energy, Pt. A/B, 20: 261-75 (Apr. 1966).

Some experiments have been carried out on the removal of iodine at low concentrations (less than 1 $\mu g/m^3$) from air by condensing steam from a mixture of steam and the air in a 2 in. diameter water-cooled tube. The results indicated that with an airflow of 10 l/min, steam condensing at a rate of 3–5 g/min, could effect the removal of about 97% of the elemental iodine, 50% or less of gaseous iodine compounds, and about 95 to 99% of iodine attached to particulate material.

430 STUDY ON REMOVAL OF IODINE FROM ATMOSPHERE BY SPRAY. I. REMOVAL OF IODINE BY SPRAY UNDER ATMOSPHERIC PRESSURE. Maekawa, Tatsuo; Nishizawa, Yukinori; Shibayama, Tetsuo; Kawaguchi, Osamu (Mitsubishi Atomic Power Industries, Inc., Japan). Nippon Genshiryoku Gakkaishi., 7: 563-9(1965). (In Japanese).

The removal of elementary iodine from the atmosphere by spray was studied using a steel drum fitted with an iodine generating system, a spray nozzle, an electric heater, and a gas-phase sampling system. The experiments were carried out over a temperature range of 25 to 80°C under atmospheric pressure, and the removal rate of iodine was estimated by gas chromatography. The results are expressed by the equation, $-dC/dt = (k_p + \alpha F)C$, where C is concentration of iodine in air, t is spraying time, k_p is plateout rate constant, F is spray rate, and α is washout rate constant. The plateout rate constant increased with increasing temperature, but the washout rate constant was independent of temperature.

METHODS OF DISPOSAL

Decay Holdup

See the following references:

80, 237

Incorporation in Solid Media

See also the following reference:

68

431 RADIOACTIVE GAS DISPOSAL. Maue, Marilyn J. (to Pullman Inc.). U. S. Patent 3,206,409. Sept. 14, 1965. Filed Sept. 10, 1959.

A process for the production of water-unleachable rock containing radioactive components is proposed as a process for the disposal of radioactive gases, such as halogens and the noble gases. A solid inorganic oxide of a metal, which is polyvalent in the oxide composition, is reacted with a solid elemental metal in the presence of the gaseous radioactive component. The elemental metal is characterized as being polyvalent in a combined state and as having a higher exothermic heat of oxide formation per oxygen atom than the monoxide of the metal oxide reactant and is selected from the group of metals consisting of Al, Si, Zr, and Mg. The metal oxide is selected from the group consisting of the oxides of Cr, V, Cd, Fe, Cu, Ti, Mn, Pb, W, In, Ga, Bi, Ni, Sn, Co, and Mo. The reaction is initiated at a temperature above 200°C, and the exothermic reaction is allowed to proceed to completion after which the reactor and contents are gradually cooled to form a water-unleachable rock-like product containing the gaseous radioactive components in the interstices of the crystalline lattice.

Stacks

See also the following references:

13, 19, 25, 32, 34, 35, 80, 88, 95, 97, 98, 129, 163, 164, 174, 182, 183, 184, 196, 201, 211, 235, 237, 238, 247, 264, 273, 279, 281, 285, 289, 304, 315, 354, 356, 357, 361, 366, 390, 392, 395, 525, 855, 880

432 (A/CONF.28/P/27) A MEASUREMENT OF THE OCCURRENCE OF MINIMUM DILUTION RATES FOR STACK EFFLUENTS IN THE ATMOSPHERE. P. J. Barry (Atomic Energy of Canada Ltd. Chalk River Nuclear Labs., Chalk River, Ont.). May 1964. 7p.

Atmospheric diffusion rates were measured continuously at four sampling positions in the vicinity of a single stack at the Chalk River Nuclear Laboratories. The dilution of the cloud or plume of radioactive pollutants as it passes downwind from the source was analyzed. Information is given on site, sampling instruments and methods, and results obtained. (N.W.R.)

433 (A/CONF.28/P/714) RECENT DEVELOPMENTS IN THE APPLICATION OF METEOROLOGY TO REACTOR SAFETY. D. H. Pack, J. K. Angell, I. Van der Hoven, and D. H. Slade (Weather Bureau, Washington, D. C.). May 1964. 20p.

Meteorological approaches to reactor siting are described. Nation-wide analyses of vertical lapse rate and methods for determining the persistence of wind regimes are considered. The use of conventional meteorological observations to infer diffusion regimes is described and illustrated for several locations. Use of wind fluctuations to obtain diffusion coefficients was improved and empirical techniques developed that permit use of conventional wind systems. Smoke photography techniques that permit quantitative analysis are described, together with illustrations of the field use of particulate tracer techniques in special problems. One of the newest meteorological techniques for application to nuclear sites is the use of balloon borne radar beacons to obtain air trajectories and diffusion data over ranges up to 100 miles. This technique, and its potential, is described. (auth)

434 (AAEC-TM-307) THE ESTIMATION OF EFFLUENT CONCENTRATIONS FROM AN ELEVATED CONTINUOUS SOURCE. Charash, E. (Australian Atomic Energy Commission Research Establishment, Lucas Heights). Dec. 1965. 26p. Dep. mn.

From fundamental theory and empirical knowledge of the structure of atmospheric turbulence spectra, diffusion equations were developed with parameters that are adjustable so as to reflect the influence of varying surface roughness as well as a wide range of stability conditions. Instructions are given for practical applications of the diffusion equations which should be valid for travel distances of about a mile at heights greater than fifty feet above ground.

435 (ACNP-63628) ERR STACK MONITOR STUDY. Sarigianis, Perry (Allis-Chalmers Mfg. Co., Washington, D. C. Atomic Energy Div.). Dec. 1963. Contract AT(11-1)-654. 15p. Dep. mn; CFSTI $1.00 cy, $0.50 mn.

Two tests were performed at the Elk River site to determine how best to increase the range of the stack monitor to cover the proposed activity release limits. Results of these tests determine the required changes to extend the overall radiation monitoring range. The results are presented.

436 (AEC-tr-6128(p.231-47)) ON THE ROLE OF TURBULENT SCATTERING ALONG THE DIRECTION OF THE WIND IN A SEMI-EMPIRICAL THEORY OF ATMOSPHERIC TURBULENT DIFFUSTION. I. L. Karol'. (Translation)

In the semi-empirical theory of turbulent diffusion in the atmosphere, the scattering of the contaminant in the direction of the wind is usually ignored. The validity of this assumption is studied by estimating the effect of turbulent diffusion in the wind direction on the distributions of the volumetric and surface concentrations. The case of diffusion of a heterogeneous aerosol is also studied. It is concluded that the effect of turbulent scattering along the wind direction is insignificant at adequately large distances from the source. (D.L.C.)

437 (AEC-tr-6365) DERIVATION OF THE GRAPHS FOR THE DETERMINATION OF THE PLUME RISE. F. H. Schmidt (Netherlands. Koninklijk Nederlandsch Meteorologisch Institut, De Bilt). Translated for Univ. of California, Lawrence Radiation Lab., Berkeley. Nov. 18,1963. 15p.

It is assumed that the discharge velocity of a hot plume from the outlet of a stack is caused entirely by the temperature of the gases, which is expressed in terms of potential temperature. The calculation of the plume rise is considered for adiabatic and stable atmospheres, and graphs are presented. The calculation of the maximum ground concentration from the plume rise is considered. (D.L.C.)

438 (AECL-2043) ESTIMATION OF DOWNWIND CONCENTRATION OF AIRBORNE EFFLUENTS DISCHARGED IN THE NEIGHBOURHOOD OF BUILDINGS. P. J. Barry (Atomic Energy of Canada Ltd., Chalk River, Ont.). July 1964. 17p. Dep.; $0.50(AECL).

The air flow in the neighborhood of buildings is briefly described and compared with that assumed for the usual atmospheric diffusion equations. The literature is reviewed and empirical formulas which have been proposed are listed and compared.

439 (AEEW-R-227) MODEL TESTS ON FLOW FROM A BUILDING VENTILATION STACK. A. L. Skinner (United Kingdom Atomic Energy Authority. Reactor Group. Atomic Energy Establishment, Winfrith, Dorset, England). Nov. 1962. 21p.

Wind tunnel tests were made on a model of a building for handling radioactive materials at the Atomic Energy Establishment, Winfrith. The object was to assess the minimum requirements for a stack that would effectively disperse the ventilation air clear of the building wind eddies and also avoid recirculation into the inlet grill. The factors of wind direction and speed, the detail design of the stack top, its height and the discharge velocity were studied, using still and ciné photography. The results suggested that just sufficient clearance of the plume could be expected for normal conditions with a stack height of 90 ft. This equals 1.7 times the height of the highest part of the roof or 2.25 times the average roof height. A high discharge velocity and a cone or ring at the stack outlet all gave a marked improvement in the flow. (auth)

440 (AHSB(RP)-R-42) METHODS OF ESTIMATION OF THE DISPERSION OF WINDBORNE MATERIAL AND DATA TO ASSIST IN THEIR APPLICATION. Pamela M. Bryant (United Kingdom Atomic Energy Authority. Authority Health and Safety Branch. Radiological Protection Div., Harwell, Berks, England). May 1964. 36p.

Methods are described for use in the estimation of the dispersion of windborne material released to atmosphere at ground level or from stacks of various heights. The methods take account of a number of practical situations including the dispersion resulting from short, prolonged, and continuous releases and the effect on the dispersion of downdraft, efflux velocity, buoyancy, elevation of the place of interest, and prevailing wind. To assist in the application of the methods, relevant quantitative data are provided graphically where possible and instructions are given to facilitate use of the graphs as ready-reckoners without reference to the detailed description. (auth)

441 (ANL-6938(p.209-41)) STACK PLUME RISE DETERMINATIONS AS AFFECTED BY METEOROLOGICAL AND ENGINEERING FACTORS. Harry Moses, Gordon H. Strom, and James E. Carson (Argonne National Lab., Ill,).

Empirical plume-rise formulas of Davidson-Bryant and Holland, and the semitheoretical formula of Bosanquet are discussed. Critical examination of these formulas indicates that a better one is needed. Stümke's generalized formula may be accepted after further tests are made with independent data. It appears that for large stacks with appreciable buoyancy such as those at Gernsheim or Duisburg, Holland's formula with a Stümke correction factor of about 3 may be used, i.e., $\Delta h = 3(1.5\ V_s d + 4 \times 10^{-5}\ Q_H)/U$. For smaller stacks with little buoyancy such as the Argonne stack, a Stümke K factor of about 2 may be used. The effect of stability on plume rise can be taken into account by adding 10 to 20% of the computed value, under lapse conditions, and subtracting the same amount under stable stratifications. Terrain effects, such as nearby buildings, hills and valleys or forested areas, must be taken into account. In most cases, this can only be done by wind-tunnel tests. Meteorological conditions may also materially affect plume rise.

442 (ANL-Trans-76) A CALCULATION OF THE MAXIMUM CONCENTRATION ON THE GROUND AND ITS DISTANCE FROM THE SOURCE. K. Imai. Translated for Argonne National Lab., Ill., from p.171-2 of report JAERI-5006. 1962. 7p.

Formulas are presented for use in calculations of the maximum concentration of radioactivity on the ground at various down-wind distances from a reactor stack. (C.H.)

443 (ANL-Trans-126) LAWS OF TURBULENT GAS DIFFUSION AS A FOUNDATION FOR THE DETERMINATION OF MINIMUM CHIMNEY ELEVATIONS. Julius Drimmel, Heinz Fortak, Ernst Geisler, Guenther Hollmann, Werner Klug, Heinz Reuter, Hermann Stuemke, and Friedrich Wippermann. Translated by Elmar K. Wilip (Argonne National Lab., Ill.), from VDI (Vere. Deut. Ingr.)-Forschungsh., No. 483, 1-?(1961). 124p. Dep.(mn); $10.10(fs), $3.92(mf) JCL.

Factory chimneys in industrialized areas ought to have a certain minimum height which guarantees that the flue gas concentration on the ground does not exceed a definite limit still safe for human beings and plants. Methods to evaluate this minimum height are necessarily based on the laws of turbulent gas diffusion and governed by a series of meteorological parameters. Several investigations, independent of each other, were carried out in order to refine and improve the existing method based on the theory developed by O. G. Sutton. These refinements consist in taking into account the vertical elevation of the flue gases over the chimney mouth, the influence of the measuring time and of the time of immission, the variability of the stability of the atmosphere with height, an improved determination of the exponent in the Lagrangian correlation function, the extension of the method to low wind velocities, and the possibilities to evaluate the meteorological parameters entering the problem.

444 (BLG-365(Vol.1)) DIFFUSION DES REJETS ATMOSPHERIQUES DANS LA REGION DE MOL. (Diffusion of Atmospheric Effluents in the Vicinity of Mol). Bultynck, H.; de Maere, X.; van der Parren, J.; Malet, L. (Centre d'Etude de l'Energie Nucleaire, Brussels (Belgium)). Dec. 1965. 68p. Dep. mn.

The general theory on atmospheric diffusion was applied to discharges at the nuclear establishment in Mol, taking into account the characteristics of the site. Five categories of thermal equilibrium were adopted and a separate approach is proposed for the cases of fumigation. The loss of material in the plumes due to settling, dry deposition, and wash-out was considered and appropriate correction factors were adopted. A mathematical model was developed to treat the problems of possible contamination of the atmosphere and of the surface in the environment, in case of releases of long duration. This model led to the construction of polar diagrams of integrated concentrations in the air or average deposition on the ground. A comprehensive set of diagrams is given in order to facilitate determination of the dilution factors.

445 (BNL-859) A DIFFUSION-DEPOSITION TRACER SYSTEM. Gilbert S. Raynor and Maynard E. Smith (Brookhaven National Lab., Upton, N. Y.). Apr. 1964. Contract AT-30-2-GEN-16. 19p.

A dual tracer technique has been developed for use in experiments in low-level atmospheric dispersion and deposition. The system is considered unique in two respects. First, it permits definition of the airborne concentrations of a cloud of small particles representative of a ture aerosol and simultaneously the deposition of larger particles on the soil and vegetation beneath. Second, the deposition tracer consisting of ^{64}Cu particles in the 1 to 10-micron range can be detected directly on the surface without the intervention of any artificial collecting mechanism or the necessity of inferring deposition by mass balance techniques. Techniques, instruments, and facilities for the preparation, activation, emission, sampling, and measurement of both tracers are described in sufficient detail to permit reproduction. (auth)

446 (BNL-914) CONFERENCE ON AEC METEOROLOGICAL ACTIVITIES, MAY 19-22, 1964. Singer, Irving A.; Smith, Maynard E.; Bierly, Eugene W. (eds.) (Brookhaven National Lab., Upton, N. Y.). Contract AT(30-2)-Gen-16. 189p. (CONF-640511). Dep. mn; CFSTI $5.00 cy, $1.00 mn.

A total of 45 papers, with group discussions, is presented which gives descriptions of site meteorological programs and which discusses major technical problems. Separate abstracts were prepared for 43 of the papers.

447 (BNL-914, pp 4-13) THE ARGONNE METEOROLOGY PROGRAM. Moses, Harry (Argonne National Lab., Ill.).

Measurements in the climatological program are described. Soil and rainfall data are given. The possible use of a high-energy proton beam as a meteorological tool is discussed. Some results are presented on radon tracer, smoke plume rise, and wind profile studies.

448 (BNL-914, pp 14-17) SCALE MODEL DIFFUSION EXPERIMENTS. Strom, Gordon (New York Univ., N. Y. School of Engineering and Science).

Experiments are being conducted on a model smoke stack in the Air Pollution Wind Tunnel at New York University, in order to model the atmospheric motions which are significant to diffusion including convective turbulence. It was found that the turbulence in the tunnel was inadequate, and roughness plates were used to remedy this problem. Similarity criteria are discussed. Plume characteristic data are presented for a few of the experiments.

449 (BNL-914, pp 24-5) THE METEOROLOGY PROGRAM AT THE HANFORD ATOMIC PRODUCTS OPERATION. Fuquay, James J. (General Electric Co., Richland, Wash. Hanford Atomic Products Operation).

A brief description is given of a study on the prediction of exposures from ground sources, and of a comparison of diffusion experiments from sources near ground level. Tracer development and studies are also discussed.

450 (BNL-914, pp 26-7) DIFFUSION FROM ELEVATED SOURCES. Elderkin, Charles E. (General Electric Co., Richland, Wash. Hanford Atomic Products Operation).

Results from the 8 best tests conducted in 1962 are presented in the form of graphs of standard deviation of crosswind exposure distribution, normalized peak exposure, and normalized crosswind-integrated exposure versus time

451 (BNL-914, pp 57-64) PROGRAM REVIEW AND SUMMARY OF RECENT ACCOMPLISHMENTS AT NRTS. Islitzer, Norman F. (Weather Bureau, Idaho Falls, Idaho).

Some of the results of diffusion studies from an elevated source and a ground level continuous point source are reviewed. A discussion is given of present research projects: studies of meso-scale transport using tetroons and radar-transponders, meteorological support of reactor distract tests, monitoring of ^{106}Ru from the Waste Calciner, Controlled Environmental Radioiodine Test, and tracer studies of boundary layer turbulence and diffusion. Operations and service of the U. S. Weather Bureau are also considered. (D.L.C.)

452 (BNL-914, pp 153-5) DIFFUSION STUDIES AT HANFORD. Simpson, Charles L. (General Electric Co., Richland, Wash. Hanford Atomic Products Operation).

The exposure data from 46 experiments are presented in the form of plots of normalized peak exposure vs time. The curves are grouped, according to the Richardson number and the lateral growth parameter, into stable, slightly stable, and unstable groups. The Hanford curves are compared with curves from other ground source experiments.

453 (BNWL-36, pp 1.1-84) ATMOSPHERIC PHYSICS. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.).

Eight papers are included. Topics discussed are: diffusion and transport model for continuous release in stable atmosphere; dispersion measurements from continuous elevated point source in unstable atmosphere; atmospheric turbulence; area dosage relations downwind of point source; precipitation scavenging; sampler for recording concentration of airborne zinc sulfide on real time scale; fluorescein-zinc sulfide dual atmospheric tracer technique; micrometeorological instrumentation.

454 (BNWL-SA-123) DEVELOPMENT OF AN ATMOSPHERIC TRANSPORT MODEL. Fuquay, J. J.; Simpson, C. L. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). Mar. 5, 1965. Contract AT(45-1)-1830. 25p. (CONF-650407-10) Dep.(mn), $1.00(cy), 1(mn) CFSTI

From International Symposium on Fission Product Release and Transport Under Accident Conditions, Oak Ridge, Tenn.

The dilution of material released to the atmosphere depends on the time of travel and the lateral and vertical growth of the cloud. It is shown that diffusion data stratify well in terms of stability when plotted in terms of the travel time, a significant improvement on the usual distance relationships. The lateral growth, σ_y, varies as the first power of time, t', initially and varies as $t^{1/2}$ in the limit. σ_y can be uniquely determined from the standard deviation of the wind direction and the mean wind speed, statistics which are routinely recorded. These results permit the assessment of the consequences of rather long releases in which exposure isopleths may take complex forms.

455 (BNWL-SA-429) RECENT HANFORD DIFFUSION RESULTS SIGNIFICANT FOR ESTIMATING POLLUTION POTENTIAL. Fuquay, J. J.; Elderkin, C. E. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). Nov. 23, 1965. Contract AT(45-1)-1830. 21p. (CONF-660401-1). Dep. mn; CFSTI $1.00 cy, $0.50 mn.

For presentation at American Meteorological Society, American Geophysical Union, Annual Joint Meeting, Washington, D. C.

Experimental programs designed to collect data on dispersion of ground sources for several miles are discussed. The data obtained from both continuous ground and elevated sources have been organized most conveniently as a function of travel time for providing estimates of exposures from potentially hazardous releases at Hanford. Measured lateral turbulence parameters, and stability measurements indicating the degree of vertical turbulence, are used to stratify the plume width and peak exposure estimates.

456 (CEND-512) ANALYSIS OF FISSION PRODUCT ACTIVITY EFFECTS FOR THE ELK RIVER REACTOR. Klotz, R. J. (Combustion Engineering, Inc., Dunedin, Fla. Nuclear Div.). Oct. 1964. Contract AT(11-1)-651. 56p. Dep. mn; CFSTI $3.00 cy, $0.50 mn.

The maximum permissible stack activity release rates for the Elk River Reactor (ERR) were determined using the best available meteorological data available. Included are separate dilution factors or atmosphere dispersion factors for iodine and other fission products. With the maximum permissible stack release rates determined, the consequences of a fuel element failure were examined. The gaseous fission product activities released from a failed fuel element by a burst, and from continuous operation following a failure, were calculated for a maximum-power and an average-power fuel element in the ERR core. If a maximum-power element should fail after operating continuously for several days, and if all the activity accumulated in the space between the fuel and cladding should be released instantaneously under the worst possible meteorological conditions, and if the effect of stack height were negligible, a person at the nearest boundary could receive one-tenth of his yearly dose. It is concluded that activity release from a fuel element failure should not limit the operation of the plant. The analysis indicates also that it is possible to continue operation with many maximum-power elements failed if the off-gas system is operating as designed. Studies of the gaseous and particulate fission product activity in the containment building following a fuel element failure were made to determine the effect on plant operation. The results indicate that a cladding failure during reactor operation, at a time when the off-gas system is not operating, can lead to excessive airborne activity levels in the containment building. Operation of the off-gas system does reduce the airborne activity within the containment building but the amount of reduction is a strong function of the ability of the off-gas system to strip non-condensible gases out of the primary system.

457 MONITORING AND DETECTION OF FISSION PRODUCT RELEASE AT A 5MW RESEARCH REACTOR. Kenneth W. Skrable (Columbia Univ., Plainsboro, N. J.). 32p. (CONF-243-33)

From American Nuclear Society Conference on Problems of Operating Research and Power Reactors, Ottawa, Oct. 1963.

The reactor cooling and off-gas systems and the equipment and procedures with which airborne radioactivity is detected and monitored are described. Results on the identification of airborne fission products and the levels of activities show that monitoring of airborne activity for fission products is a very sensitive means for detecting fission product release at the reactor. Information on the origin of activities and their relation to U^{235} exposure is included. (D.C.W.)

458 RESULTS OF EXPERIMENTS ON ATMOSPHERIC DIFFUSION. R. Le Quinio and J. Hugon. 14p. (CONF-337-23). (In French)

From International Conference on Radioactive Pollution of Gaseous Media, Saclay, France, Nov. 1963.

Studies on atmospheric diffusion were made at six French sites, each having different characteristics, but none flat and homogeneous. Site and meteorological factors affecting the diffusion were determined. The results are briefly discussed.

459 (CONF-337-41) COMPUTATION OF ATMOSPHERIC DISPERSION PARAMETERS AT REACTOR SITES BY A SMOKE PLUME RATIO METHOD. F. A. Gifford, Jr., Walter M. Culkowski, and William F. Hilsmeier (Weather Bureau, Oak Ridge, Tenn.). Apr. 1963. 13p.

From International Conference on Radioactive Pollution of Gaseous Media, Saclay, France, Nov. 1963.

Formulas previously obtained, for the determination of Sutton diffusion parameters from simple measurements of smoke plume length-width ratios, are summarized. Using these, diffusion parameters at or near four U. S. nuclear reactor installations were determined; in each instance, the requisite plume length-width ratio information was obtained from published photographs. For several of the resulting estimates of diffusion parameters, an independent estimate based on actual measured air concentration measurements is available. The "observed" diffusion parameter values (i.e., those inferred from air concentration measurements) and the values predicted by the present method are in reasonable agreement, although the ratio method values are somewhat larger. The ratio method is distinguished by its simplicity and extreme economy. On the theoretical side, it is based on exactly the same diffusion models that are ordinarily employed in more elaborate reactor site diffusion studies. The suggestion is that it should prove to be a useful, practical method by which to determine diffusion parameters at reactor sites, particularly sites, where either a large, permanent meteorological office is not contemplated, or where the expense of more elaborate atmospheric sampling studies of diffusion does not seem to be warranted. (auth)

460 CONTROL OF IODINE-131 IN IRRADIATED FUEL PROCESSING PLANTS. R. Estournel and F. Gallissian. 19p. (CONF-337-54)

From International Conference on Radioactive Pollution of Gaseous Media, Saclay, France, Nov. 1963.

During the processing of irradiated fuel elements, the fission products can be released to the atmosphere in the form of gases or powders. At Marcoule these gaseous fission products are mixed with air and released into the atmosphere by an 80-m chimney after filtration. The origin and conditions for the release of ^{131}I at Marcoule are reviewed. The filtration efficiency of the radioiodine and the characteristics of the different outlet filters were determined. The results of the monitoring of the gases showed that the processing of irradiated uranium after a sufficiently long storage causes only a very low emission of ^{131}I. (J.S.R.)

461 RADIOACTIVE COPPER AS AN ATMOSPHERIC TRACER. Gilbert S. Raynor (Brookhaven National Lab., Upton, N. Y.). St. Joseph, Mich., American Society of Agricultural Engineers, 1965, Preprint Paper No. 64-823, 36p. (CONF-641207-2). (BNL-8645)

From American Society of Agricultural Engineers, Winter Meeting, New Orleans.

Techniques and instruments were developed for the use of radioactive copper as a tracer in studies of diffusion and dispersion through the atmosphere and of deposition on the ground and vegetation. The copper, in the form of solid, spherical particles, 2 to 12 μ in diameter, or in a soluble compound, is activated by exposure to a slow-neutron flux to produce ^{64}Cu. The particles suspended in a liquid and the compounds in solution are emitted into the air by atomizing nozzles. Air concentrations are measured by filtered or rotating impactor samplers. Material deposited on the ground and vegetation is measured by mobile radiation counters, by sticky paper collectors, by autoradiographs on x ray film or by actual soil and plant samples.

Both beta and positron annihilation gamma counting are utilized in analyzing the filters and other samples. Instrument calibration techniques and safety procedures were established.

462 MEASURING THE WIND IN NUCLEAR PLANTS. Claudio Gandino (C.C.R. EURATOM, Ispra, Italy). Ing. Nucl., 5: No. 2, 1-12(1964). (In Italian) (EUR-2137.i)

The wind is the most important meteorological quantity influencing the diffusion of the radioactive gas discharged from nuclear establishments. Researches on the wind are briefly discussed, and some features are indicated. Also some anemographs are described.

463 (HW-62455) METEOROLOGICAL INSTRUMENTATION OF THE HANFORD AREA. W. A. Stone (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Mar. 1964. Contract AT(45-1)-1350. 24p.

This report lists the meteorological instruments in use in the Hanford Area and gives a brief explanation of how they work. It also includes installation dates, maintenance information, and some discussion of special problems. (auth)

464 (HW-80204) ESTIMATES OF GROUND-LEVEL AIR EXPOSURES RESULTING FROM PROTRACTED EMISSIONS FROM 70-METER STACKS AT HANFORD. James J. Fuquay, Charles L. Simpson, and W. Ted Hinds (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Jan. 2, 1964. Contract AT(45-1)-1350. 29p.

Data from theoretical and experimental studies of atmospheric diffusion were used for estimating the consequences of discharging radioactive and chemically toxic materials from 70-m stacks at Hanford on environmental exposure at ground level. Hanford diffusion data obtained in field experiments were used to determine the parameters of Taylor's equation for lateral growth. It was found that the lateral growth of a plume could be expressed solely in terms of wind observations taken near the source. This result led to the derivation of air exposure values from the bivariate normal model accounting for the wind variability characteristic to the Hanford site. The air exposures expected from prolonged releases of 2, 4, 8, and 24-hr from a Hanford 70-m stack during conditions of atmospheric stability were calculated using wind data from the Hanford tower. Results indicated that radiation exposure estimates for the emission periods were generally much lower than those previously derived for Hanford. (C.H.)

465 (HW-81746(p.1.1-19)) PREDICTION OF ENVIRONMENTAL EXPOSURES FROM SOURCES NEAR THE GROUND, BASED ON HANFORD EXPERIMENTAL DATA. J. J. Fuquay, C. L. Simpson, and W. T. Hinds (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). (HW-SA-3229)

Values of peak exposure and standard deviations of exposure distributions downwind from a continuous point source are presented for 46 Hanford ground source diffusion experiments. Exposure data were found to order in terms of atmospheric stability when plotted as a function of the travel time. The crosswind variances of the exposure distributions are expressed in terms of the travel time and the product of the standard deviation of the wind direction distribution and the mean wind speed $\sigma_\theta \bar{u}$, in an equation resulting from G. I. Taylor's work in 1921. Prediction methods developed from these concepts permitted extrapolation of the results obtained from short releases to much longer release periods. Good agreement between predicted and observed exposure distributions was obtained from these models, using independent data.

466 (HW-81746(p.1.29-36)) DISPERSION FROM ELEVATED SOURCES. C. E. Elderkin, W. T. Hinds, and N. E. Nutley (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.).

Surface measurements of the atmospheric diffusion from continuous point source releases at 56.4 m were continued at Hanford with 14 tests performed during 1963. Analyses of eight earlier tests compared well with results anticipated from a bivariate normal model, but consistent deviations from the model were noted.

467 (HW-81746(p.1.43-52)) ATMOSPHERIC TRACER TECHNOLOGY. P. W. Nickola and M. F. Scoggins (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.).

The Rankin technique for assay of bulk field samples of airborne fluorescent tracer continued to yield good results. Recent recounts were in good agreement with counts made in 1959. The possibility of replacing water with trichloroethane as the carrier for particulate tracers during generation is considered promising. Larger tracer emission rates would be possible. Tests disclosed that the effects of extreme heat and of solar radiation on one fluorescent tracer (2210) were negligible. Field models of the zinc sulfide real-time sampler disclosed an improvement of approximately four in sensitivity over the prototype model. Results from two field experiments involving tracers other than the standard 2210 are presented.

468 (HW-SA-3176) USE OF METEOROLOGICAL MEASUREMENTS FOR PREDICTING DISPERSION FROM RELEASES NEAR GROUND LEVEL. James J. Fuquay and Charles L. Simpson (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Oct. 1963. Contract AT(45-1)-1350. 24p. (CONF-337-1)

From International Conference on Radioactive Pollution of Gaseous Media, Saclay, France, Nov. 1963.

Atmospheric dispersion studies conducted at Hanford are briefly described as to experimental technique and scope. The theoretical basis for summarization of exposure data in terms of travel time are reviewed. Results are presented for the decrease of maximum exposure and horizontal plume growth as a function of travel time. Prediction parameters are developed from meteorological measurements of vertical temperature profiles in the boundary layers and the wind speed and direction near release height. The local Richardson Number and the product of the wind speed and standard deviation of wind direction are shown to correlate well with the experimental data. A method is described for predicting exposures from protracted releases in which multimodal, flattened, or skewed distributions are often observed. (auth)

469 (HW-SA-3192) FORECASTING DISPERSION FROM A SOURCE NEAR GROUND LEVEL. C. L. Simpson, J. J. Fuquay, and W. T. Hinds (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Jan. 29, 1964. Contract AT(45-1)-1350. 14p. (CONF-173-3)

From Symposium on Ceramics in Aerospace Technology, Los Angeles, Jan. 1964.

Hanford ground source diffusion data have shown the importance of both wind direction variability and atmospheric stability in determining the level of exposure due

to a ground release. The cross-wind spread of a plume has been expressed solely as a function of wind direction variability and mean wind speed through the parameter $\sigma_\theta \bar{U}$. The decrease in (normalized) peak exposure with time has been shown to depend upon $\sigma_\theta \bar{U}$ as well as atmospheric stability. Thus, the plume from a ground source has been described in terms of temperature distribution and wind velocity. Verification of the technique out to eight miles from the source point has been encouraging, and it appears that this method holds promise for a variety of uses, both theoretical and applied. (auth)

470 (HW-SA-3220) COMPARISON OF RESULTS OF ATMOSPHERIC DIFFUSION EXPERIMENTS WITH CALCULATIONS FROM PREDICTION MODELS. James J. Fuquay and Charles L. Simpson (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Sept. 1963. Contract AT(45-1)-1350. 20p. (CONF-255- 9)

From 8th Air Cleaning Seminar, Oak Ridge, Tenn., Oct. 1963.

Results of 340 diffusion experiments utilizing a ground source and encompassing a wide range of atmospheric stabilities are summarized. Peak exposures normalized for wind speed and source strength when plotted against the travel-time show stratification when stability and growth parameters are factored into the analysis. Differences in the results between sites are greatly reduced when diffusion is considered time-dependent, rather than being computed from the distance-from-source concept. In general, exposure limits for unstable, neutral, moderately stable, and stable meteorological conditions are established, and are compared with the predictions from some of the diffusion models. Whereas, the agreement between predicted and observed exposure values are good for diffusion in neutral and unstable atmospheres, there are significant differences noted in the stable case. Results of the Hanford model, which attempts to account for depletion of the plume through deposition, are presented.

471 (HW-SA-3338) ENVIRONMENTAL EVALUATION OF AN ACUTE RELEASE OF I-131 TO THE ATMOSPHERE J. K. Soldat (General Electric Co., Richland, Wash. Hanford Atomic Products Operation). June 10, 1964. Contract AT(45-1)-1350. 19p. (CONF-642-29). Dep.(mn); $1.00(cy), 1(mn) CFSTI.

From Health Physics Society, Cincinnati, June 1964.

The inadvertent release of 60 curies of ^{131}I from a Hanford Separations plant stack occurred in September, 1963. Plant operations were shutdown as soon as the abnormal release was detected, and a comprehensive program of environmental surveillance was undertaken to define the extent and magnitude of the ^{131}I deposition. Initial sampling efforts were guided by rapid calculations of the probable region of maximum deposition made by plant meteorologists. The results of the initial data collected allowed a more precise definition of the area of maximum ^{131}I deposition and the location of the individual who probably received the maximum thyroid radiation dose resulting from the accidental ^{131}I release. This individual was a child residing at the small farm where the maximum ^{131}I concentrations in milk were measured. The estimated maximum total thyroid dose to this child of 0.035 rem was supported by direct measurement of the thyroid burden made in the Hanford Whole Body Monitor. For comparison, the Radiation Protection Guide published by the Federal Radiation Council for individuals is 1.5 rem per year to the thyroid. The results of the surveillance program also supported previously derived parameters for the behavior of ^{131}I in various media in the Hanford environs and provided guidance for rapid evaluation of future accidental releases.

472 (HW-SA-3683) RADIOACTIVE AIRBORNE POLLUTANTS— EFFECTIVE CONTROL IN A NUCLEAR POWER ECONOMY. L. C. Schwendiman (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Oct. 19, 1964. Contract AT(45-1)-1350 19p. (CONF-641103-1). Dep.(mn); $1.00(cy), 1(mn) OTS.

From Air Pollution Control Assn. Annual Meeting Northwest Section, Portland, Oreg.

The potential for air pollution from the various steps in the fuel cycle is discussed. Experience in the nuclear field to date is utilized to draw conclusions as to future trends in air pollution problems resulting from the expanding nuclear industry. Two potentials for radioactive contamination of the environment must be investigated in the processing or applications of radioactive materials. The first is the potential for significant releases of airborne radioactive materials in the day-to-day routine operation. The second is the potential for and consequences of an accidental release of airborne contaminants. A conclusion is reached that the proven technology of air pollution control in processing radioactive materials and the anticipated progress from research underway will permit very low release limits for radioactive materials to be realized in routine operations. The licensing and control requirements for guarding against serious releases in accidents will reduce the probability of damage to individuals and property to a point which should not prove a deterrent to the continued growth of nuclear power and extensive applications of radioactive materials.

473 (HW-SA-3685) LESS-FAMILIAR FACTORS IMPORTANT IN AIR POLLUTION EVALUATIONS. J. J. Fuquay (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). [196?]. Contract AT(45-1)-1350. 14p. (CONF-652-2)

From American Industrial Hygiene Association Pacific Northwest Section Meeting, Richland, Wash., Aug. 1964.

It was shown that there are several factors important in evaluation of industrial air pollution problems which are not generally considered, although mention of them has been made in the literature for several years. Specification of air quality criteria compatible with the manner in which pollution is dispersed by the atmosphere and the nature of use of surrounding land was shown to significantly add to the value of air pollution evaluations. Dilution in the stack was treated on a basis compatible with the rate of release concept basic to atmospheric dispersion calculations. It was shown that specification of maximum concentration permissible in the environs requires also an appraisal of the probability of exceeding this maximum and the probability of overexposure based on numbers of persons potentially exposed. Utilization of these concepts in evaluation of multiple source problems in an industrial complex was discussed. (auth)

474 (IDO-12024) AN APPRAISAL OF GASEOUS WASTE DISPOSAL INTO THE LITHOSPHERE AT THE NATIONAL REACTOR TESTING STATION, IDAHO. Mudra, Paul J.; Schmalz, Bruce L. (Idaho Operations Office (AEC), Idaho Falls. Health and Safety Div.). Sept. 1965. 141p. Dep. mn; CFSTI $4.00 cy, $1.00 mn.

A series of gas-injection field tests were performed at the National Reactor Testing Station, Idaho, during the summer of 1964, to determine the feasibility of discharging radioactive gaseous waste to the lithosphere. Specific objectives included the determination of the: (a) storage capacity of the lithosphere, (b) direction and velocity of migration of the injected gas, (c) dilution and dispersion effects on the initial concentration of the injected gas, and (d) pressures required for injection. A cost estimate using this concept under hypothetical circumstances was also developed. Helium was used as a tracer gas to identify

approximately 33 million standard cubic feet of air injected into porous zones both above and below the regional water table within the subsurface basalts underlying the station and into the unconsolidated sandy regolith materials. The maximum depth of injection was 400 feet below land surface. A mean pressure of 15 psig was required to maintain an injection rate of 1000 cubic feet per minute into the unsaturated basalt zones. Observations indicated that no finite geological reservoir existed in the subsurface and that the regolith materials offered the most restrictive barrier to vertical movement. Although injected gas could eventually arrive at land surface, the amount of dispersion and dilution observed indicates that lithospheric gas disposal is feasible at the NRTS for single injection volumes of up to 6 million standard cubic feet. The cost for a lithospheric disposal system is estimated at a maximum of $250,000. (auth)

475 (IDO-12041) AERODYNAMIC EFFECTS OF LARGE REACTOR COMPLEXES UPON ATMOSPHERIC TURBULENCE AND DIFFUSION. Islitzer, Norman F. (Weather Bureau, Idaho Falls, Idaho). May 1965. 34p. Dep. mn; CFSTI, $2.00 cy, $0.50 mn.

The additional turbulence immediately downwind of a large reactor complex due to aerodynamic effects from the buildings was measured. Wind stations consisting of a Beckman and Whitley 3-cup anemometer and a sensitive bivane were established at upwind and downwind locations around the MTR-ETR complex. It was found that the increase of the product of vertical and horizontal turbulence downwind of the structures over that measured on the upwind side could exceed a factor of 10 for individual 5-min periods and averaged out to a factor of 3.5 in strongly stable atmospheres. A counteracting measure is the reduction of wind speed downwind of the structures which is about 30 percent. Direct measurements of diffusion to 850 m downwind were made using a fluorescent tracer, uranine dye in solution, released just downwind of the main MTR reactor building. Several crosswind sampling lines were established as well as a line of four meteorological stations oriented along the center of the grid. There was a total of 13 releases, 11 in moderately unstable atmospheres and 2 during inversion hours, for 30-min periods. The crosswind spread of the particles and the centerline dosages were measured and compared to those experienced over a flat plain from previous experiments. It was found that the building effects reduced concentrations of the tracer by about a factor of 5 at 100 m downwind and a factor of 2 at 850 m downwind on the average during lapse conditions. Occasional ten-fold reductions in concentration were measured near the buildings. It appears that during unstable conditions the additional building turbulence is not very significant beyond about 800 m. The results of the inversion tests were inconclusive because it was difficult to maintain a wind that would carry the effluent in a well-defined plume over the sampling grid during such conditions. One test indicated that the reduction of concentration from building turbulence could be as high as a factor of 50.

476 (JPRS-28343) PROBLEMS OF ATMOSPHERIC DIFFUSION AND AIR POLLUTION. Translated from Tr. Gl. Geofiz. Observ., No. 158, 3-40; 46-101; 101-19; 123-35(1964). 166p. (TT-65-30148). $5.00(OTS).

Fifteen articles are presented on the following subjects: pollution by power plants, diffusion in stratification conditions, turbulent exchange, smoke propagation from stacks, fallout, radiometers, sounding radiation meters, radioactivity of air and dust, etc.

477 (LA-DC-6106) SOURCE MEASUREMENT: DETERMINING THE RELEASE FROM A POINT SOURCE BY REMOTELY LOCATED SAMPLERS. Richard W. Henderson and Jesse W. T. Meadows (Los Alamos Scientific Lab., N. Mex.). [nd]. Contract [W-7405-eng-36]. 9p.

The effluent from some chemical and nuclear processes may be offensive or constitute a health hazard. Proper control requires a knowledge of the source characteristics. Often, measurement of the source cannot be accomplished by a direct technique. By employing a distant, circular sampling array, Sutton's equation can be modified for the calculation of the release from any such inaccessible point source. (auth)

478 (LAMS-2937) RADIOACTIVE GASEOUS EFFLUENTS FROM A HOMOGENEOUS REACTOR. Dale E. Hankins (Los Alamos Scientific Lab., N. Mex.). June 1963. Contract W-7405-eng-36. 45p.

Presented are the radioactive isotopes and the concentration of these isotopes in the effluent gases released from the Water Boiler reactor stack. The effect of prior operation of the Water Boiler reactor on the relative concentrations of the various isotopes is discussed. An evaluation is made of the results of an additional delay in the stack line. The curie output through the stack for each major isotope following a normal reactor operating day is given. (auth)

479 (NP-13687) INVESTIGATION OF THE DIFFUSION OF THE GASEOUS WASTE DISPERSED FROM STACKS IN TOKAI-MURA. (Japan Atomic Power Co., Tokyo). July 1963. 53p.

In Japanese.

Various investigations have been conducted from May 1958 through June 1959 in an attempt to obtain information on the diffusion of gaseous waste given off continuously from stacks in Tokai-mura. In addition to general meteorological observations on the ground, some observations using a captive balloon, rawin-sondes, and pilot balloons have been carried out in order to survey the atmospheric condition at high altitude. Climatological observations on many items including inversion layer in the atmosphere and the flow pattern of the air current over Tokai-mura area, which affect the waste diffusion, have also been carried out extensively. The concentration of the smoke, which is generated from lead iodide and dispersed from a 65m high stack in JAERI, was measured with monitors placed throughout the area around Tokai-mura, Mito, Hitachi. Conclusions obtained from this experiment are as follows: (1) When the smoke is dispersed at the rate of 1 unit/min, monthly average concentration on the ground shows a max of 1.4×10^{-8} unit/m^3 in June and July, in the direction of south-west of Tokai-mura, in the distance of 1.0 ~ 1.5 km. The concentration is higher than 10^{-8} unit/m^3 in Jan., Apr., Aug., and Dec. besides June and July, between SE and W of Tokai-mura, between 500m and 1.5 km. (2) Average concentration for the year is comparatively high between SE and SW, having its max of 6×10^{-8} unit/m^3 at 1 km. These conclusions were compared with the results obtained by calculation on semi-empirical formulae both of British weather bureau and of Sutton, and showed a good agreement. (auth)

480 (NP-15754) THE MINIMUM ATMOSPHERIC DILUTION OF LONG-TERM AND SHORT-TERM EMISSIONS FROM THE EUROCHEMIC CHIMNEY. Alderhout, J. J. H. (European Company for the Chemical Processing of Irradiated Fuels, Mol (Belgium)). June 1965. 27p. (ETR-173). Dep. mn. CFSTI $2.00 cy, $0.50 mn.

An estimate was made of the minimum long-term average atmospheric dilution (for routine emissions) and the minimum short-term atmospheric dilution (for accidental re-

lease) for the Eurochemic main plant chimney. The estimates were made with the aid of the Gaussian plume model. The results showed that a non-depositing, long-term, continuous emission of one curie per second from the Eurochemic plant chimney can be estimated to give rise to an average air concentration at ground level of a maximum of approximately 2×10^{-7} C/m^3 at the most unfavorable spot. This spot lies in the direction 240° (ENE) at some 3 kilometers from the Eurochemic stack. A non-depositing, short-term emission of one curie from the Eurochemic plant chimney can be estimated to give rise to a total integrated concentration of a maximum of approximately 3×10^{-5} C s/m^3 during the most unfavorable weather conditions (fumigation). The place of maximum total integrated concentration lies at about 0.5 km from the Eurochemic stack. It is thought that the values found are realistic though conservative ones.

481 (ORNL-3578(p.37-45)) RADIATION MONITORING SYSTEMS. (Oak Ridge National Lab., Tenn.).

A fallout-monitoring system is described that consists of 44 stations located at strategic points in the Laboratory area. Each station consists of an alpha monitor and a beta-gamma monitor. Two pairs of telephone lines transmit analog signals, which indicate radiation level, and an a-c tone, which indicates the condition of the alarm levels. All alarm indicators are displayed at a central point. Only 12 of the analog signals are monitored; these are randomly selected until an alarm signal indicates that a certain station or stations should be selected. An iodine detector was developed for use with stack or duct particulate monitors. The detector consists of a charcoal trap and shielded G-M tubes. A detector installed at the 3039 stack has a good sensitivity for iodine and a fair sensitivity for xenon. A status summary is given of the installation of a central display panel and instruments and controllers for containment of radiochemical processing buildings. A tone telemetry system is described. With this system, a tone of fixed frequency is inserted onto a line carrying other signals and is extracted at a receiving end where display is made by means of lights. A memory permits sampling of many units on the receiving end. (auth)

482 (ORNL-NSIC-3) CURRENT PRACTICES IN THE RELEASE AND MONITORING OF ^{131}I AT NRTS, HANFORD, SAVANNAH RIVER, AND ORNL. K. E. Cowser (Oak Ridge National Lab., Tenn.). Aug. 11, 1964. Contract W-7405-eng-26. 111p.

A study was made of the current practices in the release and monitoring of ^{131}I at the National Reactor Testing Station, Hanford Atomic Products Operation, Savannah River Plant, and Oak Ridge National Laboratory. Analyses were made of the principal sources of ^{131}I, methods of treatment prior to release, quantities released to the environment, criteria for release, sampling equipment and analytical techniques, and sensitivity of analytical techniques, and estimates were made of human exposure based on quantities detected (including nuclear test fallout) in environmental media. During 1962, the average release of ^{131}I to the atmosphere ranged from 0.045 to 0.35 curies per day. The average annual concentrations of ^{131}I in the atmosphere and in milk from cows grazing along the perimeter of the controlled areas were $<10^{-7}$ $\mu\mu$c/cc and <100 $\mu\mu$c/liter, respectively; fallout of nuclear debris from weapons tests was the major contributor of ^{131}I. Estimates of human intake did not exceed the 100 $\mu\mu$c/day limit established by the radiation protection guides published by the Federal Radiation Council. The installations are described and conclusions are drawn, based on these findings and the work of others, regarding the release of radioiodine, the chemical and physical forms at the point of release, and changes occurring in form during sojourn in the environment. The monitoring techniques investigated, in use, and most applicable for radioiodine are discussed, and information is presented on the deposition of radioiodine from contaminated clouds and subsequent intake by man. (auth)

483 (ORNL-TM-657) SEPARATION OF CERIUM-144 FROM OTHER RARE EARTHS BY ELECTROLYTIC OXIDATION AND DIFFERENTIAL EXTRACTION. R. E. McHenry (Oak Ridge National Lab., Tenn.). Feb. 1965. Contract W-7405-eng-26. 14p. Dep.(mn); $1.00(cy), 1(mn) OTS.

Cerium-144 was separated with 99.75% yield on a continuous basis from other rare earth fission products by electrolytic oxidation of the cerium(III) ion to the cerium(IV) ion, followed by differential extraction with di(2-ethylhexyl)-phosphoric acid in Amsco.

484 (ORNL-TM-851) MANAGEMENT OF FISSION PRODUCT TRITIUM IN FUEL PROCESSING WASTES. J. O. Blomeke (Oak Ridge National Lab., Tenn.). May 4, 1964. Contract W-7405-eng-26. 13p.

Tritium, a fission product of very low yield, may merit special consideration from the standpoint of its management in wastes from fuel processing. In the case of present solvent extraction plants, at least 75% of the tritium in the irradiated fuel is discharged to the environment in low-level aqueous wastes. Future plants situated less remotely may be restricted in the quantity they can release to their immediate environs. It was estimated that, under present regulations, relatively little tritium could be released to surface streams, but plants processing fuel at rates equivalent to as much as 100,000 thermal megawatts of reactor power might disperse the tritium safely to the atmosphere. Larger plants must consider containment of the tritium by internal recycle of aqueous streams. (auth)

485 (RISO-94) INVESTIGATION OF SOME FILTRATION PROBLEMS IN A HIGH-ACTIVITY HANDLING BUILDING. I. RADIO-IODINE AND -CAESIUM SAMPLING FROM THE VENTILATION AIR OF A HIGH-ACTIVITY HANDLING BUILDING. II. THE PARTICLE-FILTRATION EFFICIENCY OF INSTALLED FILTERS IN THE HIGH-ACTIVITY HANDLING BUILDING 459 AT HARWELL. Hans Flyger (Atomenergikommissionen, Risø (Denmark). Forsøgsinstitut). Nov. 1964. 24p. Dep.(mn).

I. The efficiency of different filters towards radioiodine and -cesium in the ventilation air from a high-activity handling building is reported. The iodine activity was released during the processing of a rig containing uranium carbide and was drawn through a 10 cm bed of granulated activated carbon before it was sampled. Millipore and glass-fiber filters and charcoal paper were used. Data are tabulated. II. An estimate of the size distribution of dust particles present as permanent atmospheric impurities in the intake air of the high-activity handling building is also given. An attempt was made to evaluate the particle-filtration efficiency of an installed filter bank from a combination of this estimate with a filtration theory and a photoelectric counting of the numbers of particles present at the same time on either side of the filter bank. A comparison of the results found with corresponding methylene-blue test results led to a refinement of the final analysis. Photoelectric countings of the particle numbers in the air inside the ventilation system handling the air from the high-activity handling cells showed conspicuously the presence of 100 times the number of particles that are present in the air entering the building.

486 (UCRL-Trans-1229(L)) STUDIES OF THE DIFFUSION PHENOMENA IN THE ATMOSPHERIC SURFACE LAYER. PART 2. DIFFUSION OF SMOKE PUFFS OF THE FLOATING INSTANTANEOUS-SOURCE TYPE. Inoue, Eiichi. Translated by S. Rundle (Univ. of California Lawrence Radiation Lab., Livermore) from Nogyo Kisho, 16: 20-6(1961). 22p. Dep. mn; JCL $2.60 fs, $0.89 mf.

An attempt was made to establish a connection between floating-source and fixed-source diffusion on the basis of the point of view that the differences that exist could be explained by taking into account the oscillation of the center of gravity of the smoke puff itself in the case of diffusion of the floating-source type. A firm method based on the Similarity Theory is proposed. The results showed that the characteristic figure that differentiates the two types of diffusion is the lifetime of the largest coupling eddy existing within the turbulent area brought about on diffusion. It was also found that the lifetime was characterized by the meandering of the line of smoke and by the change in movement of the degree of density of the smoke at a given point. It is tentatively suggested that the theoretical results and experimental results show a satisfactory degree of agreement.

487 ATMOSPHERIC DIFFUSION-DEPOSITION STUDIES OVER FLAT TERRAIN. N. F. Islitzer and R. K. Dumbauld (Weather Bureau Research Station, Idaho Falls, Idaho). Air Water Pollution, 7: 999-1022(Dec. 1963).

Extensive measurements of particle dispersion from a ground-level source to 3 to 2 km were carried out covering a wide range of atmospheric stabilities. A fluorescent tracer, uranine dye in small-particulate form, was released for 60-min. periods over a grid 60° wide with sampling arcs operated at 100, 200, 400, 800, 1600, and 3200 m downwind during some of the tests. An additional series of tracer releases were conducted with sampling near the ground to 800 m downwind and in the vertical direction on 30-m towers located 200 or 400 m downwind. Measurements of the vertical temperature gradient, wind velocities and gustiness were made near the tracer-release point. The test site was quite flat and covered with sagebrush elements about 50 cm high. The horizontal particle dispersion was found to be well predicted in unstable atmospheres by the standard deviation of the horizontal wind direction after averaging the wind direction record by a time dependent upon the travel time from the source to the particular sampling arc. A value of 5 for β, the ratio of the Lagrangian to Eulerian time scales, gave the best overall agreement between measured and predicted horizontal dispersion during unstable conditions. A constant value for β did not achieve satisfactory results in stable atmospheres, but a 10-sec averaging time on the horizontal wind direction record regardless of the travel times involved seemed suitable for the horizontal particle dispersion predictions to 3200 m distance. Deposition rates of the tracer on the surface were determined by material-balance computation from the air concentration measurements on the row of 30-m towers. Up to a 50 per cent loss of tracer in a distance of 400 m was measured. Deposition rates measured on specially made flat plate detectors were usually at least an order of magnitude less than those determined from material-balance calculations, which may have been attributable to the dissimilarity between the flat plates and the grid surface. Vertical dispersion, as computed from ground-level air concentration and the equation of continuity, was not in good agreement with predictions from time-averaged vertical wind direction records. The vertical dispersion determined from the air concentration measurements on the towers at one distance was found to be in reasonable agreement with vertical gustiness data, but this conclusion must be considered tentative until additional measurements have been made. Air concentrations measured in unstable atmospheres were in good agreement with values predicted from a model employing horizontal and vertical gustiness and a deposition velosity to account for the removal of tracer from the air to the ground. The agreement between predicted and measured values was less satisfactory for inversion cases. (auth)

488 THE MAXIMUM PERMISSIBLE ACTIVITY CONTAINED IN THE EXHAUST AIR FROM A NUCLEAR PLANT. [PART] I. L. A. Koenig and S. Zehme (Gesellschaft fur Kernforschung GmbH, Karlsruhe, Ger.). Atompraxis, 11: 191-8(Apr. 1965). (In German)

A method was developed for calculating the limits of the permissible amount of activity discharge from a nuclear plant. A distinction was made between continuous and short-term discharge. For particularly important radionuclides, the limits were calculated on the basis of the recommendations of the German First Radiation Protection Ordinance (1. SSVO) and of the ICRP.

489 PROTECTION AGAINST RADIATION ORIGINATING FROM THE KEMA SUSPENSION TEST REACTOR (KSTR). [PART] II. Hub. Wijker. Atoomenergie Haar Toepassingen, 5: 199-206(Oct. 1963). (In Dutch)

The dispersal of gases from the stack of the KEMA Suspension Test Reactor was discussed. The containment and storage of radioactivity are briefly considered. (tr-auth)

490 WASTE GAS DIFFUSION EXPERIMENT USING SILVER IODIDE SMOKE. K. Ito. Genshiryoku Kogyo, 9: No. 2, 33-5(1963). (In Japanese)

Determination of the atmospheric distribution of radioactive waste gas by a smoke of silver iodide particles is described. Silver iodide particles were generated by burning a 5% acetone solution of silver iodide in a generator at 600 to 900°C. The particle smoke thus produced was sent up into the air through a 5-m-long chimney. An ice box and air contamination meter of the paper filter type were used for collection of silver iodide particle smoke and detection of ice crystals. Smoke samples were collected in a 1-liter metal cylinder (ice box) which was cooled by dry ice to −17.5°C. Fine crystals formed in the ice box with the iodide particles as the nuclei fell into a supercooled saturated sugar solution at the bottom of the cylinder, and the total number of ice crystals growing to a size visible by naked eye in a short time, was counted. A paper filter type air contamination meter was used for counting. The sample was made to pass at certain rate through a paper filter for a certain length of time and the subsequent collection of the ice crystals on the paper filter was counted as above. Another method was also used, which involved exposing a piece of filter paper in the air for the adsorption of particles to take place. The smoke generator was operated at 860°C and the concentration of particles was 5×10^6/cc while the height at the smoke source was 20 m and the direction and speed of the wind was 7.0 m/sec. While the number of smoke particles in the air recorded by the ice-box method was merely a transient value, the paper filter method recorded the total number throughout the period of sampling. The results revealed that, in addition to the influence of wind speed and direction and temperature, the diffusion of silver iodide particles was considerably affected by the terrain and roughness (houses and trees etc.) of the ground surface.

491 THE DECONTAMINATION OF RADIOACTIVE WASTE WATER AND POLLUTED AIR FROM RADIOLOGI-

CAL CONTROL AREAS. H. Mende. Gesundh.-Ingr., 84: 90-6(1963).

Handling and disposing of radioactive waste water originating from radiological institutes and clinics are reviewed with particular reference to the methods used for decontamination of radioactive waste discharged into atmosphere and waste water and their safety margins. Details of radiation-proof apparatus, decontamination of isotopes, chemical decontamination, and a specially designed air filter are described. It is emphasized that decontamination should take place before discharge of the wastes. (Public Health Eng. Abstr., 44: No. 10, Oct. 1964.)

492 DERIVATION OF WORKING LIMITS FOR CONTINUOUS RELEASE RATES OF IODINE-131 TO ATMOSPHERE IN A MILK PRODUCING AREA. Pamela M. Bryant (United Kingdom Atomic Energy Authority, Harwell, Berks, Eng.). Health Phys., 10: 249-57(Apr. 1964).

An assessment of derived working limits for continuous iodine-131 release rates is based on a provisional working limit for the concentration in milk of 0.4 μC/l. The value was derived by the Medical Research Council on the basis that the radiation dose to the thyroid of an infant during its first year should not exceed 3 rems. The meteorological and agricultural conditions assumed are applicable as an annual average to England and Wales. A derived working limit of 20 mC/day of iodine-131 is suggested as the basis of a design figure for an installation, assumed to be the only source of iodine-131 in the area, having a stack of 20 m height at a distance of 500 m from the nearest pasture. The results for other heights and distances are shown graphically. Discharges should be made as uniformly as practicable, and if it is operationally necessary to exceed the continuous release rate over short periods, the quantity released over a few hr should not exceed that permitted for 1 week unless meteorological or agricultural conditions are particularly favorable at the time. Due allowance should be made for multiple releases, and final proof of satisfactory control of the releases always rests on the measurement of the iodine-131 content of milk. (auth)

493 EXPERIMENTS ON THE BEHAVIOUR OF EFFLUENT EMITTED FROM STACKS AT OR NEAR THE ROOF LEVEL OF TALL REACTOR BUILDINGS. P. O. A. L. Davies and D. J. Moore (Central Electricity Research Labs., Leatherhead, Surrey, Eng.). Intern. J. Air Water Pollution, 8: 515-33(Oct. 1964).

The behavior of effluent emitted from stacks projecting a few feet above the roof level of reactor buildings at Berkeley and Bradwell nuclear power stations was studied using models in a wind tunnel and a water tunnel. Full-scale plumes from these buildings were also observed under a limited range of conditions; the behavior of model and full-scale plumes was similar when the efflux conditions were similar. The experimental work indicated that the assumption of fairly simple effluent distributions for the purpose of calculating long-term γ-dose rates would not lead to serious errors. (auth)

494 RESULTS OF RECENT FIELD PROGRAMS IN ATMOSPHERIC DIFFUSION. James J. Fuquay (General Electric Co., Richland, Wash.), Charles L. Simpson, Morton L. Barad, and John H. Taylor. J. Appl. Meteorol., 2: 122-8(Feb. 1963).

During the summer of 1959, the Green Glow program, consisting of 26 diffusion experiments during nocturnal inversions, was conducted at the Atomic Energy Commission's Hanford Site near Richland, Wash. The tracer, zinc sulfide, was released near ground level. Samplers were placed at 1.5 m above ground at 533 positions on six sampling arcs, the radii of which were 200 and 800 m and 1.6, 3.2, 12.8, and 25.6 km. In addition to the ground sampling network, poles or towers were erected at 5 points, 8° apart, on each of the 4 inner arcs. Fifteen samplers were mounted on each pole or tower, the top level increasing from 27 m on the 200-m arc to 62 m on the 1.6 and 3.2-km arcs. General aspects of the experimental design and tracer technique are discussed along with terrain characteristics and meteorological conditions pertinent to these experiments. Experimental results are presented showing the increase in horizontal plume width and decrease of maximum exposure with distance from the source. An analysis of the area enclosed within a given exposure isopleth is summarized. The effect of significant wind direction shear on the vertical distributions of exposure is discussed. Results from the Green Glow experiments are compared with those from earlier diffusion experiments at O'Neill, Nebr., and later experiments at Hanford. (auth)

495 A METEOROLOGICALLY OPERATED STACK CONTROL SYSTEM. Gerald C. Gill and Eugene W. Bierly (Univ. of Michigan, Ann Arbor). J. Appl. Meteorol., 2: 431-9(Aug. 1963).

The standard deviation of wind direction and the mean wind speed were incorporated into a system for controlling the release of radioactive gaseous wastes from the 200-ft stack of the Enrico Fermi Atomic Power Plant. The meteorological sensor used in the system is a Bendix-Friez Aerovane, located on top of a 100-ft meteorological tower near the stack. Data from the Aerovane are processed by a small analog computer whose outputs are displayed on the control board of the reactor control room and a recorded on strip chart recorders. The display enables the reactor operator to know whether the stack is open to the atmosphere or whether the gases are being stored in containment tanks. Averaging times of the meteorological variables may be switched to 3, 6, 12, or 24 minute periods. A discussion of several representative records from the computer and the Aerovane, pointing out some of the interesting features of this system, is presented. (auth)

496 A GUIDE TO RELEASE-RATE OF IODINE-131 FROM STACK IN NORMAL OPERATION OF REACTOR. Toshinori Iijima and Ichiro Miyanaga (Japan Atomic Energy Research Inst., Tokyo). J. Nucl. Sci. Technol. (Tokyo), 1: 316-21(Nov. 1964).

For estimation of the release rate of ^{131}I, the effect of individual variations by age in the general population and the additional effect of ingestion of foods contaminated in the neighborhood of a reactor are discussed. The limiting release rate from the 55-m stack of the Japan Power Demonstration Reactor is then estimated. The release rate should be determined in consideration of children of ages one and two since in these subjects the positive effect from increase in breathing-rate exceeds the negative effect due to increase in thyroid-mass. The effect from ingestion with green vegetables would become comparable to that from inhalation with air, when the deposition-velocity is 2cm/sec and the intake-rate is about 100 g/day. The limiting release rate from the JPDR is about 4 mC/hr under meteorologically neutral conditions with a mean wind-speed of 2.5 m/sec, which is the most predominant condition at the Tokai site.

497 MEASUREMENT OF RADIOACTIVE WASTES VENTED BY THE AIR STACK OF THE VVR-S REACTOR

IN THE PERIOD 1959-63. András Németh and Mihály Tóth. Magy. Tud. Akad. Kozp. Fiz. Kut. Int. Kozlemen., 12: 25-41(1964). (In Hungarian)

The method for measuring the radioactive wastes vented at the air stack of the VVR-S reactor is described, and the results obtained are given. Results of investigations on the concentrations of radioactive dust particles in the atmosphere are also given. The results are used to estimate the airborne radioactivities to be expected around the reactor. (auth)

498 WASTE GAS HANDLING. SYSTEM FOR 300 Mw DIRECT CYCLE BWR. E. E. Duke (Aerojet-General Corp., Azusa, Calif.). Nucl. Eng., 9: 245-8(July 1964).

The results of an analysis of gas release in a 300-Mw (thermal), direct-cycle boiling water reactor are presented. Available processing methods and gaseous waste disposal requirements are summarized, and two absorption systems are proposed for reducing the effluent gases to less than 0.1 the maximum permissible concentration taken as 10^{-10} $\mu C/cm^3$ in the stack. (D.C.W.)

499 CLIMATOLOGICAL ESTIMATES OF DIFFUSION CONDITIONS IN THE UNITED STATES. C. R. Hosler (Weather Bureau, Washington, D. C.). Nucl. Safety, 5: No. 2, 184-92(Winter 1963-64).

In the evaluation of potential hazards of radioactivity releases, the dilution capacity of the atmosphere must be taken into account. Results of studies of regional patterns of slow atmospheric dilution are collated to indicate areas where air pollution problems are probable. The parameters studied include wind speed, low-level inversions, subsidence inversion and stagnation episodes, mixing depths, and time-space considerations. It is concluded that atmospheric dilution is not an important criterion in reactor siting studies. (D.L.C.)

500 EFFECTS OF METEOROLOGICAL AND ENGINEERING FACTORS ON STACK PLUME RISE. Harry Moses, Gordon H. Strom, and James E. Carson. Nucl. Safety, 6: No. 1, 1–19(1964).

Two empirical formulas, those of Davidson-Bryant and of Holland, and the semitheoretical formula of Bosanquel, for stack plume rise are discussed. Critical examination indicated that for large stacks with appreciable buoyancy, Holland's formula may be used with a Stümke correction factor of about 3. For smaller stack with little buoyancy, a Stümke factor of about 2 may be used. The effect of stability on plume rise was taken into account by adding 10 to 20% of the computed value under lapse conditions and substracting the same amount under stable stratifications. It was found that terrain effects, such as nearby buildings, hills, valleys, and forested areas, must be taken into account. It was also found that meteorological conditions may materially affect plume rise. For example, under sunny conditions with winds of 2 mph or less, intermittent gusts sometimes brought the effluent to the ground within one or two stack heights. Moderate to heavy rain in association with gusty thunderstorm winds produced wide deviations from expected results. Practical considerations for minimizing ground-level concentrations are discussed. (47 references)

501 ATMOSPHERIC POLLUTION FROM NUCLEAR POWER PLANTS. J. W. Flora and G. H. Smith. Nucl. Safety, 6: No. 1, 81-7(1964).

Radioactive polution of the atmosphere by nuclear power plants was studied to ascertain whether any condition exists at the facility that may be detrimental to the health and safety of the public. Sources of radioactive effluents, measures taken to limit release of air contaminants, bases for commission limits on releases, and the degree to which effluent limits are met are discussed.

502 RELEASE AND MONITORING OF ^{131}I AT NRTS, HANFORD, SAVANNAH RIVER, AND ORNL. K. E. Cowser (Oak Ridge National Lab., Tenn.). Nucl. Safety, 6: No. 2, 212-18(Winter 1964-65).

Studies were made of the release and monitoring of ^{131}I and the ultimate radiation exposure that man may receive. Installations used in the study were the National Reactor Testing Station (NRTS), Hanford Atomic Products Operation (HAPO), Savannah River Plant (SRP), and Oak Ridge National Laboratory (ORNL). Atmospheric release, terrestrial and surface-water release, environmental concentrations, estimated exposure of man, and empirical relations of ^{131}I in the environment are discussed. Data obtained on chemical and physical forms of radioiodine, monitoring for ^{131}I, deposition of ^{131}I, and intake by man are presented.

503 DEFLECTOR TO PREVENT SUCTION OF SMOKE INTO CHIMNEY SLIPSTREAMS AT HIGH WIND SPEEDS. R. Sallé (Commissariat à l'Énergie Atomique, Paris). Trans. Am. Nucl. Soc., 7: 462-3(Nov. 1964).

504 METHODS PREVENTING THE DIFFUSION OF RADIOACTIVE MATERIALS FROM NUCLEAR REACTORS AND HOT LABORATORIES. Pelser, J. Water, Bodem, Lucht, 54: No. 4, 67-73(1964).

Methods employed for the safe operation of nuclear reactors and hot laboratories by preventing release of radioactive materials are reviewed. In the case of nuclear reactors leakage of the fuel cladding, the primary cooling circuit, and the containment have to be prevented. The disposal of gaseous radioactive waste is an important problem. In hot laboratories a good ventilation system is necessary; several precautions are required for the handling of solid, liquid, and gaseous radioactive waste. (J.V.E.)

505 DEFLECTOR TO PREVENT SUCTION OF SMOKE INTO CHIMNEY SLIPSTREAMS AT HIGH WIND SPEEDS. R. Salle (CEA, Paris). Proc. Conf. Remote Systems Technol., 12th, San Francisco, 1964, 39-44.

Wind tunnel experiments have been made on smoke leaving a chimney under conditions when the smoke velocity is less than that of the cross-wind blowing across the stack. Model chimneys were used, some equipped with special devices for eliminating the drawing-down of smoke into the chimney slipstream that normally occurs in this situation, others with no special devices. On the basis of these studies, a new device has been developed to increase the effective chimney height, this device consisting of two conical collars placed around the top of the chimney, the upper collar having the broad base of the collar facing downward, the lower collar with the broad base facing up. Penetration of the smoke into the slipstream is largely prevented by the continuous layer of air flowing horizontally through the space between the two collar bases, providing that these are properly dimensioned and positioned. Using alcohol vapor as a synthetic smoke, air concentrations have been measured in the lee of the model chimney for various wind speed/smoke speed ratios under the following conditions: (a) unequipped, (b) equipped with previously developed devices and, (c) equipped with the new device. (The instrument used in measuring the relative alcohol concentrations is described.) Aerodynamic forces and currents at the collars of the new device have also been determined.

506 ESTABLISHMENT OF THE HEIGHT OF CHIMNEYS FOR OFF-GASES FROM NUCLEAR ENERGY INSTALLATIONS. U. E. Michaelis (VEB Energieprojek-

tierung Kerntechnische Anlagen, Berlin). Symp. Tech. Zagadnien Ochrony Promieniowaniem, Warsaw, 662-8 (1962). (In German)

The permissible discharge of radioactive gases and aerosols from the chimney was determined as a function of the effective chimney height on the basis of meteorological evaluations and the MPC in the chimney vicinity. Resulting gas masses, gas characteristics, and especially the radioactivities are the decisive factors for the economically most suitable disposal of off-gases on which the selection of the chimney height is directly dependent. After selection of the system for off-gas treatment, such as gas decay containers, and aerosol filters, and off-gas velocity measurements, the minimum chimney height necessary was determined from the data on the necessary effective chimney height. (tr-auth)

507 METEOROLOGICAL PROBLEMS IN THE DISPERSION OF RADIOACTIVE MATERIALS IN THE ATMOSPHERE. W. Bleeker (Koninkjijk Nederlands Meteorologisch Instituut, De Bilt, [Netherlands]. p.32-56 of "Symposium Gezondheidsaspecten bij Radioactieve Besmetting van Levensmiddelen." [The Hague], N. V. Uitgeverij W. P. van Stockum & Zn., [1962]. (In Dutch)

The meteorological problems in the dispersion of radioactive materials in the atmosphere are discussed. The distinction between natural and artificial radioactive materials and the definitions, measurement methods, and units used are described. Tropospheric and stratospheric fallout is considered. The processes depositing radioactive materials in the troposphere and the atmosphere are reviewed. The effects of precipitation and storms on the distribution of radioactive materials are considered. The radioactivity resulting from nuclear reactors and other installations is indicated.

508 SOME POTENTIAL BIOLOGICAL CONSEQUENCES OF A RELEASE OF FISSION PRODUCTS TO THE ATMOSPHERE. Stewart, C. G.; Simpson, S. D. (Atomic Energy of Canada Ltd., Chalk River, Ont.). pp 183-207 of Protection of the Public in the Event of Radiation Accidents. Geneva, World Health Organization, 1965.

products to the atmosphere by the rupture of a reactor fuel core are discussed. There are at least three kinds of hazard to which a population living in the vicinity of a reactor will be exposed in the event of a radiation accident. These include the external radiation dose and the inhalation dose received from the passing cloud, contamination of the ground by fallout from the cloud, and contamination of food produced from land contaminated by radioactive debris. The magnitude of these hazards will be proportional to the kind and amount of radioactivity released. Procedures are described for determining the proportions of fission products released by a reactor accident; the composition of the fission product cloud; the external β or γ dose from the cloud; the external radiation dose from the noble gas fission products in the cloud; the external β or γ dose from the ground due to deposition from the passing cloud; the contamination of food due to deposition of radioactive material from the passing cloud; and the radiation dose from ^{131}I, ^{137}Cs, ^{89}Sr, ^{90}Sr, or tritium oxide released to the environment.

509 TRUDY GLAVNOI GEOFIZICHESKOI OBSERVATORII. VYPUSK 138. VOPROSY TURBULENTNOI DIFFUZII I RASPROSTRANENIYA PRIMESEI V ATMOSFERE. (Reports of the Main Geophysical Observatory. Issue 138. Problems in Turbulent Diffusion and Propagation of Admixtures in Atmosphere). M. E. Berlyand, ed. Tr. Gl. Geofiz. Observ., No. 138, 1-89(1963). Leningrad, Gidrometeorologicheskoe Izd-stvo, 1963. 89p.

Ten articles are presented on distribution of industrial and radioactive aerosols in the atmosphere. Descriptions are given of meteorological conditions conducive to admixture diffusion and turbulent diffusion processes. A separate abstract was prepared for one article of special interest. (R.V.J.)

Subterranean

See the following references:

196, 889

LIQUID WASTES

GENERAL

See also the following references:

18, 28, 33, 153

510 DEVELOPMENT OF RADIOACTIVE WASTE FRACTIONIZATION AND PACKAGING TECHNOLOGY. DeMier, Warren V.; Richardson, Gerald L.; Platt, Allison M.; Swift, Ward H. (General Electric Co., Richland, Wash.). Chem. Eng. Progr., 60: Symp. Ser. No. 53, 50-61(1964).

Solvent extraction, ion exchange, and precipitation processes are described for removing long-lived heat-generating fission products (strontium-90 and cesium-137) from high-level radioactive wastes. Packaging techniques for long-term storage of isolated strontium and cesium employ ion exchange absorption on zeolites followed by thermal stabilization and encapsulation in small-volume, high-integrity containers. (auth)

511 RADIOACTIVITY REMOVAL BY WATER AND WASTE TREATMENT PROCESSES. Public Works, 96: 104-7(May 1965).

Removal from water of low-level radioactivity, such as that entering sewage from hospitals and other installations using radioisotopes, is considered. The treatments for removal of radionuclides described include specific chemical precipitation, sand filtration, lime and soda softening, ion exchange, as well as several nonconventional processes. Iodine, widely used by hospitals, can be removed by the addition of small amounts of copper sulfate, activated carbon, or silver nitrate. Coagulation is unsatisfactory for removal of ^{90}Sr. With an excess of cesium, strontium, or iodine, removal efficiency of coagulation is low; with an excess of trivalent ions of phosphorous, zirconium, and niobium, efficiency is high. Combined coagulation, settling, and filtration gives high removals (up to 99% of the initial activity) of ^{32}P. Lime alone is effective in removing zirconium and niobium. Lime and soda treatment provides >90% reduction in the common radioisotopes of barium, lanthanum, strontium, cadmium, scandium, and yttrium. Treatment plants using zeolite, green sand, or cation and anion exchangers can remove small concentrations of radionuclides readily (over 94% of tungsten, yttrium, scandium, strontium, barium, lanthanum, cesium, barium, cadmium, zirconium, niobium, and iodine). Of the conventional processes of water treatment, ion exchange was found to offer the most promise. Of the nonconventional processes, phosphate coagulation and slurrying with metallic dusts showed decontamination factors as high as 1000. In general, water treatment

processes can provide decontamination factors as high as 10, or slightly higher with modified methods. Of the biological processes of waste treatment, sand filter treatment showed the highest decontamination with factors as high as 100 reported. Since radionuclides remain in sludge after digestion, the use of sludge from radioactively contaminated sewage, for fertilizer creates a problem, especially for radionuclides of long half life. Factors to be considered include the radiation dose from surface contact with the contaminated fertilizer and the transfer of radionuclides from the soil to above-ground portions of the plant. Strontium, which is taken up readily from soil by plants, is not concentrated in sludges. (BBB)

Analysis

See also the following references:

88, 252, 532

512 (AERE-R-4460) THE DETERMINATION OF RADIUM-228 IN HARWELL EFFLUENT. W. M. Henry, B. A. Loveridge, and J. R. Weaver (United Kingdom Atomic Energy Authority. Research Group. Atomic Energy Research Establishment, Harwell, Berks, England). Dec. 1963. 23p.

Radium is recovered from an aliquot of effluent by coprecipitation with lead and barium sulfates. The precipitate is dissolved in an alkaline solution of ethylenediaminetetraacetic acid and the barium–radium sulfate is reprecipitated by careful adjustment of the acidity. Following metathesis to the carbonate and solution in dilute acid, the barium–radium solution is fed to a heated column of ion exchange resin. The radium is eluted free from barium carrier, with 0.5 M citric acid adjusted to pH 6 with ammonia. The actinium-228 daughter is permitted to grow in the radium fraction of the eluate before being extracted with di-(2-ethylhexyl) hydrogen phosphate. The recovered actinium is purified by further ion exchange separations and then beta counted. The analytical procedure is standardized against a solution of natural thorium containing a known quantity of radium-228. (auth)

513 (CEA-R-2553) DOSAGE DES PRODUITS DE FISSION DANS LES EFFLUENTS DU TRAITEMENT DES COMBUSTIBLES IRRADIES (METHODE RADIOCHIMIQUE). (Determination of Fission Products in Irradiated Fuel Treatment Effluents (Radiochemical Method)). Auchapt, Jacqueline M. (Commissariat a l'Energie Atomique, Chusclan (France). Centre de Production de Plutonium de Marcoule). Feb. 1966. 20p. Dep. mn.

Determination methods applicable to relatively long-lived fission products present in the effluents resulting from irradiated fuel treatment processes (Sr–Cs–Ce–Zr–Nb–Ru–I) are reported. The methods are based on the same principle: addition of a carrying-over agent; chemical separation over several purification stages; determination of the chemical yield by calorimetry; and counting of an aliquot liquid portion.

514 (EUR-2118.f) METHODES SPECIALES POUR LA MESURE DE LA RADIOACTIVITE DES EAUX USEES. (Special Methods for the Determination of Waste Water Radioactivity). A. Malvicini (European Atomic Energy Community, Ispra (Italy). Joint Nuclear Research Center. Feb. 1965. 15p. (CONF-640506-1). Dep.(mn).

From Biomathematics Symposium, Houston, Tex.

After a short introduction stressing the importance of the problem, a review is presented of the methods of control and measurement of radioactivity in waste waters. Strontium and cesium determination in such waters is also described in detail. (auth)

515 (MLM-1151) MOUND LABORATORY PROGRESS REPORT FOR FEBRUARY 1963. J. F. Eichelberger, G. R. Grove, L. V. Jones, and E. A. Rembold (Mound Lab., Miamisburg, Ohio). Feb. 30, 1963. Contract AT(33-1)-Gen-53. 28p.

The radioisotopes in a waste stream from the polonium process were identified as Fe^{59}, Co^{60}, Se^{75}, Ag^{110m} and Sb^{124}. Zn^{65}, one of the principal impurities, could be removed by hydroxide precipitation.

516 (ORNL-3744) DETERMINATION OF NUCLIDE CONCENTRATIONS IN SOLUTIONS CONTAINING LOW LEVELS OF RADIOACTIVITY BY LEAST-SQUARES RESOLUTION OF THE GAMMA-RAY SPECTRA. E. Schonfeld, A. H. Kibbey, and W. Davis, Jr. (Oak Ridge National Lab., Tenn.). Jan. 1965. Contract W-7405-eng-26. 59p. Dep.; $3.00(OTS)

One of the most difficult and expensive analyses associated with radioactive-waste disposal is the determination of the amount of each of four to ten radioactive nuclides when present only at the very low radioactivity levels of 10^2 to 10^5 $\mu\mu$C per liter. Gamma-ray spectrum resolution, previously developed to determine higher specific activities, based on data analysis by high speed computer, was tested and found to provide a rapid, semi-automatic, reasonably accurate method for these analyses at a cost of 5 to 25 times less than that for conventional methods. The savings increase with the number of samples. Tests, with two multichannel spectrometers, and data analyses by the method of least squares, using a high speed computer, were made with solutions of ^{60}Co, ^{85}Sr, ^{141}Ce, and ^{144}Ce added as tracers to water containing ^{90}Sr, ^{95}Zr–Nb, ^{106}Ru, ^{134}Cs, and ^{137}Cs. The lowest concentration of radioactivity was about one-tenth of the lowest value previously determined by this technique, that is, down to 10 counts/min. Evaporation by factors up to 20 was also evaluated for improving accuracies of analyses. For comparison, the solutions were also analyzed by standard radiochemical methods. Agreement between the two methods was excellent for ^{60}Co and ^{85}Sr for concentrations down to about 500 $\mu\mu$C/liter (scatter of ±5 to ±20%). Agreement between the two methods was good for ^{137}Cs at activities down to 3000 to 4000 $\mu\mu$C/liter; comparative values from the two methods differed by factors of 5 to 10 for ^{144}Ce except at activities above 1600 $\mu\mu$C/liter. More dilute solutions were successfully analyzed only if they were first concentrated by evaporation. The method is not recommended for determination of ^{106}Ru at activities below 1000 to 2000 $\mu\mu$C/liter, unless this nuclide constitutes a major fraction of the total activity and ^{85}Sr is a minor component. Several contributions to the mathematical techniques of gamma-ray spectrum resolution are reported. These include: data showing that gain shifts up to 5%, and threshold shifts of two channels, can be accurately handled by a computer program; resolution of the errors of concentration estimates into contributions by background and the various radioactive species present; and use of chi-square to test the overall reliability of the multichannel spectrometer. (auth)

517 (ORNL–3782, pp 84-90) PROCESS INSTRUMENTATION AND CONTROLS SYSTEMS ENGINEERING. (Oak Ridge National Lab., Tenn.).

Instrumentation and cell ventilation for the transuranium processing plant are discussed along with instrumentation for thorium-uranium fuel cycle development facility, instrumentation of the radiochemical processing pilot plant, and instrumentation of a waste evaporator. Instrumentation for measurement of liquid level in waste solution storage

tanks is described along with instrumentation of reaction vessels for processing carbon-coated fuels and salt-field instrumentation for waste disposal. (J.R.D.)

518 (ORNL-P-447) DETERMINATION OF NUCLIDE CONCENTRATIONS IN SOLUTIONS CONTAINING LOW LEVELS OF RADIOACTIVITY BY LEAST-SQUARES RESOLUTION OF THE GAMMA-RAY SPECTRA. E. Schonfeld, A. H. Kibbey, and W. Davis, Jr. (Oak Ridge National Lab., Tenn.). [Sept. 3, 1964]. Contract [W-7405-eng-26]. 42p. (CONF-721-2)

From 8th Conference on Analytical Chemistry in Nuclear Technology, Gatlinburg, Tenn., Oct. 1964.

Gamma-ray spectrum resolution, a rapid, cheap method for determining the radionuclide content of solutions, was tested for use in studies of processes concerned with the decontamination of waste water containing only 10^2 to 10^5 $\mu\mu$C of radionuclides per liter. Tests, with two multichannel spectrometers, and associated data analyses by the method of least squares, were made with solutions of ^{60}Co, ^{85}Sr, ^{141}Ce, and ^{144}Ce added as tracers to water containing ^{90}Sr, ^{95}Zr–Nb, ^{106}Ru, ^{134}Cs, and ^{137}Cs. The lowest concentration of radioactivity was about one-tenth of the lowest value previously determined by this technique. Some of the solutions were concentrated by evaporation by factors up to 20 to evaluate this feature for improving accuracies of analyses. For comparison, the solutions were also analyzed by standard radiochemical methods. Agreement between the two methods was excellent for ^{60}Co and ^{85}Sr for concentrations down to about 500 $\mu\mu$C/liter (scatter of ±5 to ±20%). Agreement between the two methods was good for ^{137}Cs at activities down to 3000 to 4000 $\mu\mu$C/liter; comparative values from the two methods differed by factors of 5 to 10 for ^{144}Ce except at activities above 5000 $\mu\mu$C/liter. More dilute solutions were successfully analyzed only if they were first concentrated by evaporation. Because of large statistical fluctuations at ^{106}Ru concentrations as high as 1000 to 2000 $\mu\mu$C/liter, no particular significance can be given to the data for this nuclide. Several contributions to the mathematical techniques of gamma-ray spectrum resolution are reported. These include: data showing that gain shifts up to 5%, and threshold shifts of two channels, can be accurately handled by a computer program; resolution of the errors of concentration estimates into contributions by background and the various radioactive species present; and use of chi-square to test the overall reliability of the multichannel spectrometer. (auth)

519 DETERMINATION OF TRACE QUANTITIES OF URANIUM BY CONTROLLED-POTENTIAL DC POLAROGRAPHY IN A TRI-n-OCTYLPHOSPHINE OXIDE EXTRACT. Belew, W. L.; Fisher, D. J.; Kelley, M. T.; Dean, J. A. (Oak Ridge National Lab., Tenn.). Microchem. J., 10: 301-14(Jan. 1966). (ORNL-P-1540).

The combination of solvent extraction and controlled-potential d-c polarography directly in the organic extract provided a sensitive and selective method for the determination of uranium. The method should be applicable to a wide range of samples containing trace concentrations of uranium, such as sea water and waste streams. Uranium concentrations as low as 2.4 μg in 250 ml of sample can be determined when a 5-ml polarographic cell is used. If a smaller polarographic cell is used, an evaporation step can be added after the extraction and as little as 0.2 μg of uranium determined in 250 ml of sample. The method compares favorably in precision with colorimetric methods for uranium and is more selective and free from interferences.

520 RADIOCHEMICAL DETERMINATION OF ^{210}Pb IN MILL PRODUCTS AND BIOLOGICAL MATERIALS. Sill, Claude W.; Willis, Conrad P. (Atomic Energy Commission, Idaho Falls, Idaho). Anal. Chem., 37: 1661-71 (Dec. 1965).

The determination of ^{210}Pb through its ^{210}Bi daughter was re-examined for application to complex inorganic systems in addition to biological materials. Special emphasis was placed on complete recovery of both lead and bismuth, so that either element can be used in the determination. In the large number of cases in which secular equilibrium is known to exist, the bismuth fraction can be counted and results obtained within a few hours, eliminating the lengthy delays common to many other procedures. Extraction with a solution of diethylammonium diethyldithiocarbamate in chloroform separates lead and bismuth quantitatively from large quantities of other metals in strongly acid solution. Dithizone then eliminates the remaining interferences and separates the two elements from each other prior to beta counting. High activities of gamma-emitting ^{212}Pb and ^{207}Bi tracers were used to detect and identify any source of error larger than about 0.05%. Recovery of both elements through the entire procedure was over 99% from 500 ml of river water or liquid mill effluents, 1 g of ores or mill tailings, 100 g of blood, tissue, feces, or grain, 1500 ml of urine, or large samples of air dusts. Overall recovery was still about 98% even from 100 g of soil or bone ash. Separation of lead and bismuth from each other was better than 99.9%. Several sources of error are discussed.

521 A CONTRIBUTION TO THE DETERMINATION OF ^{90}Sr IN EFFLUENTS IN THE PRESENCE OF ^{89}Sr. Zange, E. (Kernforschungsanlage, Juelich, Ger.). Atompraxis, 11: 365-8(July 1965). (In German).

In spite of the presence of sulfate in the effluent radiostrontium was enriched by way of a carbonate precipitation. Prepurification was carried out by a nitrate precipitation. Both the isolation of pure radiostrontium and separation of the ^{90}Y was done by means of an improved ion exchange method. The high chemical total yield (90%) and the high counting efficiency (54%) permitted very small ^{90}Y growth periods. (auth)

522 POLAROGRAPHIC DETERMINATION OF URANIUM IN THE WASTE SOLUTION OF NUCLEAR FUEL REPROCESSING. Kenji Motojima, Hiroshi Okashita, and Tamotsu Sakamoto (Japan Atomic Energy Research Inst., Ibaraki). Bunseki Kagaku, 13: 1097-1100(Nov. 1964). (In Japanese)

Determination of micro amounts of uranium in the waste solution of nuclear fuel reprocessing (Purex method) was investigated polarographically. The waste solution contains HNO_3, HNO_2, organic substances, fission products, corrosion products, and radiolyzed substances. The uranyl wave is affected seriously by these coexisting substances. Disproportionation effect of U(V) was eliminated by adding $(NH_4)_2SO_4$ to the sample solution. The organic substances were extracted by stirring the sample solution with benzene in the polarographic cell. HNO_2 was removed by passing through the solution with an alcohol-saturated nitrogen gas. The following procedure was suggested: 1$\underline{M}$ $(NH_4)_2SO_4$ and benzene is taken in a polarographic cell. Then the sample solution is poured into the cell and deaerated by a purified nitrogen gas for 10 minutes. If HNO_2 is contained in the sample solution, the solution is bubbled with an alcohol-saturated nitrogen gas for 1 minute. The d-c polarogram is recorded from 0 to −0.5 v. Uranium content is determined by using the calibration curve. This method is considered applicable to the determination of uranium in the waste solution ranging from 5×10^{-5} to $5 \times 10^{-3}\underline{M}$ uranium. (auth)

523 SELECTIVE DETERMINATION OF RUTHENIUM 103 AND RUTHENIUM 106 IN EFFLUENTS AND NATURAL

WATERS. J. Scheidhauer and L. Messainguiral. Chim. Anal. (Paris), 45: 452-62(Sept. 1963). (In French)

Ru radioisotopes are extracted from radioactive effluents by CCl_4. The problem of destruction of certain complexes is solved. Ru^{103} is discriminated from Ru^{106} by a counting technique. The use of the described method for the separation of these two isotopes from natural waters is indicated. (tr-auth)

524 SEPARATION AND MEASUREMENT OF RADIOCERIUM IN RADIOACTIVE EFFLUENTS. J. Scheidhauer, L. Messainguiral, and N. Drogue. Chim. Anal. (Paris), 46: 567-71(Nov. 1964). (In French)

Two methods for determination of radiocerium are described. In one, utilizing a carrier, the cerium is precipitated as the oxalate after a series of decontaminations. In the second with no carrier, the cerium is chemically separated with lanthanum as the coprecipitant and extracted with 2-ethylhexylphosphoric acid. Cerium is finally deposited from hydrochloric acid solution on mylar and counted with a β counter. Utilization of an aluminum screen permits separation of ^{141}Ce and $^{144}Ce + ^{144}Pr$. (tr-auth)

Monitoring

See also the following references:

34, 147, 252, 281, 514, 538, 539

525 (HW-77609(Sect.5)) INSTRUMENTATION. (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). 49p.

Instruments developed or modified include: an automatic data system for atmospheric physics studies; a phosphorescent particle detection instrument for use with ZnS tracer; a monitoring system for protective clothing; a circuit for a liquid effluent radiological monitoring system; a portable radiation survey instrument; a solid state scintillation α detector; and radiation dosimeters with audible or visual signals. A technique for the measurement of neutron dose; development of a contaminated air monitoring technique; a coincidence counting method for Pu contaminated filters; and the development of a detector for airborne Be are presented. (C.H.)

526 (NLCO-970, pp 8.1-4) THE DETERMINATION OF BETA ACTIVITIES IN PLANT EFFLUENTS. Twitty, B. L.; Humphrey, H. W. (National Lead Co. of Ohio, Cincinnati).

Two methods were devised for determining the beta activity of plant effluents containing uranium and thorium decay products. The lower detection level was 3 d/m/ml. of effluent water. The first method, developed for laboratories, utilizes a barium coprecipitation at high pH, followed by gross beta counting with an antracene crystal. The second method, developed for plants, utilizes a detector head consisting of four Geiger-Müller tubes which are submersed for counting in a known volume of effluent. The count is compared with that of a standardized europium-152, -154 solution.

527 (ORNL-TM-786) CALIBRATION OF RADIATION MONITORS FOR PROCESS WASTE WATER. R. L. Shipp (Oak Ridge National Lab., Tenn.). Feb. 12, 1964. Contract [W-7405-eng-26]. 10p.

Two configurations of radiation detectors, used at ORNL for monitoring branch lines of the process waste-water sewer for radioactive contamination, were calibrated to obtain beta- and gamma-energy response curves. These curves, and published data on decay schemes, were used to calculate the response of the detectors to several isotopes. The detectors and calibration procedure are described, and the calibration data are analyzed. (auth)

528 (ORNL-TM-1442) STEAM CONDENSATE BETA-GAMMA MONITOR AT THE FISSION PRODUCTS DEVELOPMENT LABORATORY, BLDG. 3517. Chase, L. H.; Knowles, D. J. (Oak Ridge National Lab., Tenn.). Feb. 17, 1966. Contract W-7405-eng-26. 6p. Dep. mn. CFSTI $1.00 cy, $0.50 mn.

Steam condensate from the Fission Products Development Laboratory is stored in a 2000-gal underground tank, and monitored continuously for beta-gamma radiation, and at intervals is drained into the ORNL Process Waste System. The pumping system for withdrawing samples and the associated radiation monitoring system are described, and a checkout procedure is included.

529 (UJV-1104/64) PRAKTICHESKIE PROBLEMY PO DOZIMETRII RADIOAKTIVNYKH OTKHODOV. (Practical Problems in Dosimetry of Radioactive Wastes). Ya. Ralkova (Ceskoslovenska Akademie Ved. Ustav Jaderneho Vyzkumu, Rez). [1964]. 6p.

Four basic methods of waste concentration; coagulation and coprecipitation, evaporation, ion exchange, and radiochemical separation; are discussed, and their efficiencies and shortcomings are analyzed. Devices for measuring α, β, and γ emission from waste are described, and difficulties in treating the data are discussed. (R.V.J.)

530 METHODS AND TECHNIQUES FOR MEASURING THE RADIOACTIVITY IN WATER AND WASTE WATER. J. Bosch. Ber. Abwassertech. Ver., No. 11, 20-33 (1960). (In German)

Methods and instruments used for the measurement of the concentrations of radioactivity in water and waste water are described. The fundamental factors in radioactivity measurements are presented, and a formula for the relation between radioactivity concentration and measurement device readings is derived. The formula is used in the description of practical measurement set-ups. The G-M counter tube, other counter tube instruments, methane flow-through counters, scintillation counters, and special devices for the measurement of the water activity are discussed individually. (J.S.R.)

531 THE MONITORING OF THE RADIOACTIVITY OF WATERS. PROBLEMS POSED BY THE MEASUREMENT AND THE USE OF THE RESULTS. A. Grauby and P. Raimbault. Eau, L, 50: 307-13(May 1963). (In French)

Standards, quantities, and techniques in the measurement of activity are discussed, and apparatus and stations for monitoring the activity of water for consumption and of radioactive effluents are discribed. (D.C.W.)

532 TECHNICAL SOLUTION TO THE CONTINUOUS MONITORING OF THE RADIOACTIVITY OF LIQUID WASTES AND THEIR RECEIVING WATERS. Pierre Bovard (Commissariat à l'Energie Atomique, Paris). Foederation Europaeischer Gewaesserschutz, Informationsbl., No. 10, 53-5(May 1964). (In French)

The control of radioactive waste water discharge into rivers consists of three stages: continuous sampling, global measurements, and detailed measurements. Equipment for monitoring water is classed in four categories: equipment for direct measurement, integrating resin units, devices recording the radioactivity of dry aerosols, and devices recording the radioactivity of dry extracts. Both radiochemical analysis and gamma spectroscopy can be used for direct measurement. To control effluent discharge, specific overall activity, specific activity of certain elements, amount of effluent discharges, and effective amount of activity discharged measured in curies must be known. Various types of monitoring systems are described. (M.C.G.)

533 ON THE AUTOMATIC MONITORING AND CLASSIFYING APPARATUS OF LOW RADIOACTIVE LIQUIDS. REPORT I. Keiji Miyazaki and Setsuo Kawamoto (Hitachi Shipbuilding and Engineering Co., Ltd., Tokyo). Hitachi Zosen Giho, 25: 188-94(Aug. 1964). (In Japanese)

In a treatment and disposal system for radioactive liquid, it is sometimes required to monitor the radioactivities of liquids and classify liquids according to their intensities. Moreover, it is desirable to do this automatically and continuously. To investigate the technical feasibility of this concept, a continuous-monitoring and automatic-classifying apparatus composed of liquid-dip type radiation detector, a count-rate meter, electrical relays, and electromagnetic valves was made for trial and its performance was tested. Tests indicate that good performance is obtained from an appropriate combination of a dead band around the classifying level with a large time constant of the counting circuit. (auth)

534 METHOD FOR THE MEASUREMENT OF THE RADIOACTIVE CONTENT OF WASTE WATER. F. J. Haughey and R. M. Manganelli. J. Water Pollution Control Federation, 36: 88-95(1964).

The measurement of radioactivity in sewage is complicated by the presence of the solid fractions, and experiments were therefore carried out on the distribution of added iodine-131, phosphorus-32, and thallium-204 among the various solid fractions of domestic sewage. (UKAEA)

535 MONITORING OF THE RADIOACTIVE CONTAMINATION OF WATER. II. MEASUREMENT METHODS AND THEIR APPLICATION. Sigfrid Mueller. Tech. Ueberwach., 4: 412-17(Nov. 1963). (In German)

In the last few years the measurement techniques for radioactivity monitoring of waste water have undergone significant changes. It is now possible to construct reliable instruments with which very low activities can be measured continuously and directly. A brief review is given of the methods and instruments used for the direct measurement of β, α, and γ activity. The measurements after enrichment of radioisotopes are indicated. (J.S.R.)

Sampling

See also the following references:

34, 278, 528, 532

536 (HW-74040) SAMPLING PROCESS WASTE SLUDGE IN UNDERGROUND STORAGE TANKS BY ROTARY CORE DRILLING. J. P. Duckworth, V. P. Kelly, J. D. Lodge, and J. R. Raymond (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). June 29, 1962. Contract AT(45-1)-1350. 15p.

A core driller was developed for coring hard sludge in underground tanks, and sludge samples were taken from the Purex 241-A-103 tank. The development, design, and operation of the drill are described. An electric conductivity probe was used to stop the drill when it contacted the tank bottom. Qualitative observations are given on the physical properties of the sludge in drilling tests. (D.L.C.)

537 (HW-77380) CHEMICAL AND RADIOCHEMICAL ANALYSES OF TK-241-A-103 SLUDGE. Final Report. T. R. McKenzie (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Apr. 10, 1963. Contract AT(45-1)-1350. 8p.

Core samples from top, center, and bottom sections of the Tk-241-A-103 sludge were analyzed. The constituents which were determined included Pu, Np, U, common anions and cations, and various fission products, including five Sr isotopes. (D.L.C.)

538 LIQUID- AND GASEOUS-WASTE-EFFLUENT SAMPLING AND MONITORING. Knowles, D. J. Nucl. Safety, 7: 52-6; 64-5(Fall 1965).

Literature published since 1960 in the field of waste treatment and disposal covering developments in instrumentation and techniques of sampling and monitoring gaseous and liquid waste is reviewed. The references include foreign papers as well as those of U. S. origin. General comments on waste-effluent-monitoring apparatus are made.

539 SAMPLING CONSIDERATIONS IN MONITORING PROCESS-WASTE EFFLUENTS. D. J. Knowles. p.59-62 of "Nuclear Safety. Vol. 5, No. 1."

Techniques of sampling for monitoring process-waste effluents are discussed. Topics covered include: representative sampling, losses in sampling system, accuracy, sampling frequency, and total flow measurements. (M.C.G.)

METHODS OF TREATMENT

Biological

See also the following references:

34, 104

540 (JPRS-23903) THE EFFECT OF STRONTIUM^{89} ON THE MICROFLORA OF BIOLOGICAL FILTERS. B. N. Il'yin. Translated from Gigiena i Sanit., [29]: No. 2, 102-4(Feb. 1964). 5p. (OTS-64-21915)

In deactivation of sewer water containing ^{89}Sr in a concentration of 5×10^{-7} C/l, the maximum isotope content in the biological film reached 1×10^{-5} C/kg of raw weight. This quantity of ^{89}Sr did not have any effect on the biochemical activity of heterotrophic microorganisms in a biological film and on the weight increase in it during the first 8 to 10 months of operation and did not impair

the functioning of the biofilter in decontaminating sewer water. Daily introduction of ^{89}Sr in a concentration of 5×10^{-7} C/l with sewer fluid does not affect the maturation of biofilters under optimum operating conditions. ^{89}Sr in concentrations of 5×10^{-2} to 5×10^{-6} C/l did not have a marked effect on nitrifying bacteria in the biological film of the biofilter. (auth)

541 ACCUMULATION OF RADIOACTIVE STRONTIUM BY HYDROPHYTES AND DETRITUS. A. L. Agre and M. M. Telitchenko. Federation Proc., Transl. Suppl., 22: T1218-19(Nov.-Dec. 1963).

Uptake and accumulation of Sr^{90} was studied in several species of green algae, blue-green algae, diatoms, mosses, and in duckweed. The organisms were placed in experimental solutions made of standing tap water into which Sr^{90} was introduced as $SrCl_2$. Activity of the solutions was 10^{-8} to 10^{-5} Curie per liter. Radioactivity of the organisms was determined every 25 days on a B unit with an SI-2B lead-shielded end-window counter according to standard method. The values of the coefficients of Sr^{90} accumulation by the hydrophytes were found to be inversely proportional to the isotopic concentration of the medium. Algae had higher Sr^{90} accumulation coefficients than did mosses, and the accumulation in algae was more rapid. Experiments showed that organic wastes effected 80% decontamination of the solutions, on the average. It is suggested that hydrophytes and organic wastes may consequently be used for decontamination of water containing Sr^{90}. (H.M.G.)

542 ON THE EFFECTS OF STRONTIUM-89 ON THE MICROFLORA IN BIOLOGICAL FILTERS. B. N. Il'in. Gigiena i Sanit., No. 2, 102-4(Feb. 1964). (In Russian)

Deactivation of waste water containing 5×10^{-7} C/l of ^{89}Sr with the maximum content in the biological layer reaching 10^{-5} C/kg of raw weight, was analyzed. The chemical activity of heterotrophic microorganisms in the biological layer and the growth during the first 8 to 10 months were not affected by the ^{89}Sr. Neither was the performance of a water purification biofilter influenced. Concentrations of ^{89}Sr from 5×10^{-2} to 5×10^{-6} C/l did not effect the bacteria in the biofilter microflora. (R.V.J.)

543 EFFICACY OF ACTIVE MUD IN THE DECONTAMINATION OF RADIOACTIVE WASTE MATERIAL. G. Branca and S. Gagliardi. Igiene Sanita Pubblica (Rome), 19: 496-519(1963).

Research was carried out with the object of ascertaining the efficiency of active mud processes for the decontamination of waste products containing radioactivity. The biological systems used for the purification of domestic waste products, including the process with active mud, are not specifically suitable for the treatment of contaminated waters. Nevertheless these offer to a community that possesses purifying plants a certain degree of protection in those cases in which the waste products are contaminated accidentally. The active mud process is preferred because it has the advantage of making possible the addition to the waste products of coagulant reagents which increase the decontamination. The radioisotopes studied were: ^{137}Cs, ^{90}Sr, ^{106}Ru, ^{131}I and ^{32}P. The results obtained are presented in graphs and tables. (Excerpta Med., Sect. 23, 1: No. 7, July 1964)

544 BIOLOGICAL TREATMENT OF RADIOACTIVE WASTE WATER. Masaharu Imaoka (Inst. of Public Health, Tokyo). Koshu Eiseiin Kenkyu Hokoku, 13: 90-5(June 1964). (In Japanese)

Literature data indicated that the biological treatment methods for municipal sewage, such as activated sludge processes, trickling filters, and oxidation ponds, are applicable to radioactive wastewater. In order to confirm the applicability of the conventional methods to radioactive wastewater, an activated sludge experiment was conducted to see whether the sludge organisms suffer any harm in radioactive water and whether they decontaminate radioactivity. Two samples were placed each in a one liter aeration bottle; one was non-radioactive for control, and the other was radioactive with ^{32}P. Sewage and activated sludge were obtained from a municipal sewage treatment plant. Every day, new sewage and ^{32}P were added after the end of aeration and the sewage was left standing for about thirty minutes, then 800 ml of the supernatant were discarded. Five ml of the centrifuged supernatant were dried on a stainless steel planchet under an infrared lamp and the radioactivity was counted with a G-M counter. Sorption of ^{32}P by the sludge was observed. The radioactivity in the sludge increased day by day. The removal rate of ^{32}P was found to range from 46 to 90%. The total numbers of microorganisms in the sludge in both samples were counted and compared. In short term and at such low levels of ^{32}P as 0.3 to 0.9 μC/l, the activated sludge organisms were found to suffer no harm from the radioactivity of ^{32}P concentrated in the sludge. (auth)

545 TREATMENT OF RADIOACTIVE WASTE WATER WITH ACTIVATED SLUDGE. Masaharu Imaoka (Inst. of Public Health, Tokyo). Koshu Eiselin Kenkyu Hokoku, 13: 96-104(June 1964).

Treating low-level radioactive waste water together with municipal sewage by biological methods is thought to be effective and economical. If no suitable sewage is available, some other nutrients may be added to radioactive waste water. Some experiments were carried out with nutrients such as flour, glucose, and dried sludge from night soil digestion tanks to see if good sludge could be obtained. The test plant consisted of an aeration tank with a capacity of 1 m^3; a settling tank, 0.3 m^3; a reaeration tank or sludge oxidation tank, 1 m^3; and a spare tank, 0.3 m^3. Some test operations were carried out with these nutrients mixed with tap water. First, 3 parts in weight of dried sludge and 1 part of glucose were added as nutrients. When seeded with return sludge from a municipal sewage treatment plant, good activated sludge having very good settleability was obtained. Then dried sludge was substituted with flour to decrease the content of phosphorus in the test plant, and the settleability of the activated sludge became slightly poorer, because dried sludge seemed to contain too much phosphate as nutrient to remove ^{32}P. Removal and holding tests of activated sludge from the test plant and a municipal sewage treatment plant were made in 1-liter aeration bottles. In the first half of the test period, every 24 hr, 800 ml of the supernatant was discarded after 0.5 to 3 hr of standing and fresh artificially radioactive waste water (tap water containing radioactive material and nutrients) was filled in the same amount. In the second half of the test period, aeration was stopped for 0.5 to 3 hr for sampling, and instead of discarding the supernatant, the equivalent amount of the radioactive material was added with or without the equivalent amounts of the nutrients. In some cases, only water was supplemented without adding the radioactive material and the nutrients, and sampling was repeated. Samples were dried on stainless planchets and counted with a G-M counter. More than 99% of ^{32}P was found to be removed from the nutrition-dosed waste water when phosphorus was deficient. As for ^{89}Sr, a better result was obtained with the higher content of dried sludge in the test bottle that contained much phosphorus. Phosphorus-32 and ^{89}Sr were held in the sludge for 2 or 3 weeks, and the radioactive-material removability of the sludges was also

found to be kept good for some 2 weeks if the nutrients added and other conditions were suitable. (Public Health Eng. Abstr., 45: Feb.-Mar. 1965)

Calcination

See also the following references:

7, 13, 15, 19, 24, 26, 32, 36, 64, 68, 79, 98, 106, 122, 123, 132, 178, 187, 188, 189, 190, 191, 192, 193, 194, 195, 197, 198, 199, 200, 201, 202, 203, 204, 209, 210, 213, 217, 220, 221, 222, 226, 228, 229, 230, 231, 232, 242, 243, 244, 245, 246, 247, 248, 249, 370, 372, 537, 696, 703, 704, 708, 737, 753, 771, 778, 878

546 (BNL-832) CONTINUOUS CALCINATION OF HIGH LEVEL RADIOACTIVE WASTES BY MEANS OF A ROTARY BALL KILN. Roy F. Domish and L. P. Hatch (Brookhaven National Lab., Upton, N. Y.). Nov. 1963. Contract AT-30-2-GEN-16. 31p.

Continuous calcination of simulated, high salt content, high level radioactive wastes by means of a rotary ball kiln was demonstrated on a practical scale. The development of process equipment for operation with highly corrosive and dust-laden off-gases, with emphasis on the design features of a special rotary seal, was a central objective. Through condensation of off-gases and exclusion of air from the system, a high degree of control over dust produced in the calcination process was readily achieved. Results are presented of extensive corrosion studies required for the selection of materials of construction for equipment suitable for service in the calcination of wastes containing both fluorides and nitrates. The investigation culminated in the construction of a pilot-scale plant and was terminated after a one-month continuous operation of the plant. (auth)

547 (BNWL-213) RESEARCH AND DEVELOPMENT ACTIVITIES, FIXATION OF RADIOACTIVE RESIDUES. Quarterly Progress Report, October–December 1965. Platt, A. M. (ed.) (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). Feb. 1966. Contract AT(45-1)-1830. 38p. Dep. mn. CFSTI $2.00 cy, $0.50 mn.

Intensive study of one composition melted under various conditions indicated a marked change in properties (solubility and drip temperature) as the processing temperature was raised. At a given temperature, holding periods between 1 and 4 hr have little effect. There appears to be an optimum processing temperature. Thermal expansion of the candidate solids for WSEP was measured and found to be in a range that would be compatible with stainless steel containers. Slope of the temperature reciprocal versus log solubility line in the range 75 to 95°C was found to be −4.68. It was found that at high fission product levels rare earth phosphates crystallize out in the melter. A series of 46 melts was prepared for a statistical study of the effects of additives to PW-2 waste on melt and product properties. None of the specific compositions tried had the desired properties of a high sulfate retention and a stickpoint above 700°C. It was found that nickel, Corronel 230, and Hastelloy G have inadequate corrosion resistance for use as melter construction material. The separation efficiency and performance of wire-wound filters showed that they could be used for the spray calciner. The scope of the WSEP GE-412 computer program was defined and an operating procedures guide for using the computer was written. Design verification tests on the spray calciner were completed by filling an 8-in.-dia container 8 ft long with glass. Retention of sulfate was necessary for good performance of the spray calciner. Containers were filled without stalagmite formation when the wall temperature of the container was within 100°C of the melting point. Expressions were derived for the change of recycle stream compositions with time for WSEP. The cases of continuous and batch recycle were considered. Fission product release studies involving simulated high level ceramic wastes were extended to include different ceramic compositions. Both phosphate and fission product contents were varied. In most instances, release rates at 1100°C for ^{137}Cs were found to range from 0.2 to 0.5%; however, some exceptions were observed. Attempts to measure an entrainment rate for particles (cesium is volatilized into the airstream) resulted in releases below the detection limit of the equipment — less than 10^{-5} released. Studies were concluded on sulfate melts prepared from simulated glass melter condensates. Some types of lead sulfate melts could be prepared and used as storage media, however, the long processing times required makes this method unattractive. Separation of sulfate and recycle of the radionuclides to the primary melter appears to be a practical method for treating the acid condensate.

548 (BNWL-SA-569) HOT-CELL STUDIES OF THE SPRAY AND POT CALCINATION OF FULLY RADIOACTIVE FUEL REPROCESSING WASTES. Moore, R. L. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). Mar. 4, 1966. Contract AT(45-1)-1830. 15p. (CONF-660208-13). Dep. mn. CFSTI $1.00 cy, $0.50 mn.

From Symposium on Solidification and Long-Term Storage of Highly Radioactive Wastes, Richland, Wash.

A small pilot plant for the experimental study of the calcination of full-level radioactive wastes was constructed in a large hot-cell at Hanford and operation initiated in 1962. Both radiant-heat spray calcination and pot calcination were proven operable. The radioactive off-gas was found to be readily decontaminated to safe discharge levels. The results obtained and implications for a full-size plant are discussed, as are techniques used for decontaminating the cell following over two years of operation.

549 CALCINER POT, 8 IN. (Engineering Materials). (Oak Ridge National Lab., Tenn.). (CAPE-1150)

This package contains 5 drawings. For a listing of individual titles see the Engineering Materials List (TID-4100). This mock-up pot is used in liquid waste calcination. It is designed for a maximum pressure of 200 psi at 600°F; a maximum temperature of 1700°F at 10 psi for 100 hr; negligible reduction of wall thickness due to corrosion during calcination; and negligible shell stresses due to contraction. Further investigation is recommended to verify design assumptions, as 304 L stainless steel is used and creep rupture data for this material were not readily available. The pot is 62.5 in. long and 8 in. ID. The reducer has an 8 in. ID at the bottom, a 4 in. ID at the top, and is 6 in. long. (F.S.)

550 FLUIDIZED BED CALCINATION OF HIGH-LEVEL RADIOACTIVE WASTE IN A PLANT-SCALE FACILITY. J. A. Buckham, A. L. Ayers, and J. A. McBride (Phillips Petroleum Co., Idaho Falls, Idaho). New York, American Institute of Chemical Engineers, 1964, Preprint 21d, 20p, $0.50. (CONF-488-7)

From American Institute of Chemical Engineers 53rd National Meeting, Pittsburgh, May 1964.

Operation of the world's first plant-scale facility for converting high-level radioactive waste solutions to a safer solid form began in December 1963, employing the fluidized bed calcination process. Correlated data from initial operation are presented, with emphasis on the highly satisfactory performance of the off-gas cleanup system. (auth)

551 SELECTION AND PERFORMANCE OF NaK HEATING SYSTEMS FOR A FLUIDIZED BED PROCESS. E. S. Grimmett (Phillips Petroleum Co., Idaho Falls, Idaho). New York, American Inst. of Chemical Engineers, 1964, Preprint No. 36, 23p. $0.50. (CONF-644-1)

From 7th National Heat Transfer Conference, Cleveland, Aug. 1964.

The paramount requirements which influenced the choice of the heating system for the fluidized bed waste calcination process were: The amount of off-gas leaving the fluidized bed should be minimized. The system should operate at as high a temperature as possible, subject to the limitations imposed by the materials of construction required by the process. The system should operate at or near atmospheric pressure. The system should be flexible and easy to operate so that the temperature in the fluidized bed can be held at any desired value within the operating range of 300 to 600°C. A liquid metal (NaK) system was selected as the one which best fulfilled these requirements. Operation of a NaK heating system, in a pilot plant calciner and a production scale calciner, proved that the choice of a liquid metal heating system for this process was a good one. Operation of both systems pointed up some factors which caused initial problems but which were eliminated by adopting the following design and operating principles: minimizing thermal and vibrational stresses, maintaining extreme cleanliness within the liquid metal system, and careful inspection and testing of all components and materials prior to installation. (auth)

552 (HW-75223) ANALOG COMPUTER SIMULATION OF A POT CALCINER. Ray D. Benham (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Oct. 1, 1962. Contract AT(45-1)-1350. 38p.

The derivation and analog computer simulation is presented of a mathematical model of the calcination portion of the low temperature evaporation method for waste calcination. The effects of process variables on the time dependence of the radial temperature profiles were studied for pots of different geometry. The results are limited because the chemical reaction rate constants and thermal properties were estimated. Calculated heating periods (2 to 20 hr) are in agreement with literature, and the lag time between initiation of cooling and the occurrence of a hot spot is shown to be appreciable (4 to 20 hr). (D.L.C.)

553 (HW-77979) INSTRUMENTATION FOR CALCINER PRODUCT CANISTERS— LEVEL DETECTION. C. E. Huck (General Electric Co., Richland Wash. Hanford Atomic Products Operation). June 27, 1963. Contract AT(45-1)-1350. 9p. Dep.(mn); $1.00(cy), 1(mn) CFSTI.

Two different level detection methods have been evaluated for use with calciner product canisters. The first method uses MgO insulated, stainless steel-sheathed, chromel-alumel thermocouples spaced four inches apart vertically for power level detection. A usuable drop of 75 to 100°C in thermocouple temperature occurs when preheated calciner powder at a temperature of about 440°C slowly covers a thermocouple which was previously heated by radiation from the canister wall to a temperature of 550 to 600°C. The second method involves the use of electrical conductivity for melt level detection. The measured 6.4 mho/cm electrical conductivity of the melt is very close to the value predicted from earlier fused salt work and to values obtained from the Handbook of Electrochemical Constants, which ranged from 1.74 mho/cm for KBr to 6.74 mho/cm for LiCl. This conductivity value is suitable for actuating an electronic relay as an incremental level readout method but attempts have not been successful to obtain a conductivity signal proportional to melt level. The high viscosity and incomplete fusing of the melt at temperatures of 600 to 700°C as well as encrusted layers and voids below the surface are probably the reasons for the erratic readings obtained. Each of the two methods, thermocouple temperature excursion, and electrical conductivity measurement, appears to be specific but neither will indicate the other variable; that is, thermocouples will not read out melt level and conductivity probes will not read out powder level. It should be noted that the level measurements so far are preliminary and have been made with detectors at fixed locations and therefore proportional signals have not been obtained. (auth)

554 (HW-83896) SOLIDIFICATION OF HIGH-LEVEL RADIOACTIVE FUEL REPROCESSING WASTES BY SPRAY AND POT CALCINATION—HOT-CELL PILOT PLANT STUDIES. R. T. Allemann, F. P. Roberts, and U. L. Upson (General Electric Co., Richland, Wash. Hanford Atomic Products Operation). Sept. 1964. Contract AT(45-1)-1350. 97p. Dep.; $4.00(CFSTI).

When excess sodium was present in the waste, the loss of cesium was increased during pot calcination. The calcined waste containers did not develop any excess pressure upon storage except when excess sodium nitrate was present in the waste. Fission product distribution in the melted wastes was fairly uniform and in the same ratio as in the liquid waste except for ruthenium losses. (auth)

555 (HW-84603(p.8-31)) CALCINATION AND SOLIDIFICATION STUDIES. R. E. Burns, C. R. Cooley, and R. L. Moore (General Electric Co., Richland, Wash. Hanford Atomic Products Operation).

The investigation of the continuous phosphate glass process was continued by three more runs; equipment failure terminated the experiments. With insufficient feed concentration in the evaporator, the sulfuric condensate contained ~5.6% of the ruthenium, and the off-gas was decontaminated by a factor of at least 10^6. Leaching tests were made on phosphate glass and on a Redox borosilicate glass and a Purex phosphate solid produced in the hot spray calciner-melter. The study of phosphate addition indicates an optimum M^+/P ratio in the neighborhood of 2.3 or 1.0. Purex 1WW glass compositions were studied. Materials were evaluated for corrosion resistance to Purex waste–calcium borate at 900°C. The coupled spray calciner and melter was operated with simulated 1WW waste with various fluxes. Various studies in waste storage pot inspection, analog computer solutions, and transient thermal behavior in waste fixation are described. (D.L.C.)

556 (HW-84603(p.32-6)) WASTE SOLIDIFICATION ENGINEERING PROTOTYPES. C. R. Cooley (General Electric Co., Richland, Wash. Hanford Atomic Products Operation).

The design and fabrication status of the equipment is described. Major equipment pieces were installed in the cold semiworks facility for design verification tests. Water boilups on both 8- and 12-in. pots have been made and four 8-in. pot calcination runs were completed using the prototype induction furnace. (D.L.C.)

557 (IDO-12042) BIOLOGICAL AVAILABILITY OF RADIONUCLIDES IN CALCINED WASTE. Echo, John B. (Idaho Operations Office (AEC), Idaho Falls. Health and Safety Div.). June. 1965. 7p. Dep.(mn); $1.00(cy), 1(mn) CFSTI.

The ratio of the amount of soluble compound radioisotope used to give a unit amount of radioactivity in the critical organ was compared to the amount of the same isotope in calcined waste needed to give this same value. In laboratory experiments using rabbits of the Dutch variety these ratios were $1/10$, $1/2$, and $2/3$, for cesium-137, strontium-90, and plutonium-238-239, respectively. (auth)

558 (IDO-14608) DEVELOPMENT OF FLUIDIZED BED CALCINATION OF ALUMINUM NITRATE WASTES IN THE WASTE CALCINING FACILITY. Lakey, L. T.; Lohse, G. E.; Palica, W. P. (Phillips Petroleum Co., Idaho Falls, Idaho. Atomic Energy Div.). May 1965. Contract AT(10-1)-205. 150p. Dep.(mn); $4.00(cy), 3(mn) CFSTI.

Development of the fluidized bed process for converting nuclear fuel reprocessing wastes to granular solids during the 3700 hours of non-radioactive test operation of the plant-scale ICPP Waste Calcining Facility is described. (auth)

559 (IDO-14618) DEVELOPMENT OF A FLUIDIZED BED CALCINATION PROCESS FOR ALUMINUM NITRATE WASTES IN A TWO-FOOT-SQUARE PILOT PLANT CALCINER. PART III. INTERMEDIATE PROCESS STUDIES– RUNS 11 THROUGH 22. B. P. Brown, B. M. Legler, B. R. Wheeler, E. S. Grimmett, and J. A. Buckham (Phillips Petroleum Co. Atomic Energy Div., Idaho Falls, Idaho). Feb. 10, 1964. Contract AT(10-1)-205. 131p.

Intermediate process studies comprising twelve runs were made in a two-foot-square calciner to provide support data for the Waste Calcination Facility (WCF) and to study further the fluidized bed calcination process. For each run, the objectives, conditions, and results are given. Simulated aluminum reactor fuel processing wastes were calcined to a granular free-flowing solid under a variety of operating conditions and with various types of atomizing nozzles. Demonstration of the operability of the feed nozzle selected for use in the large-scale WCF and determination of the factors affecting the intra-particle porosity of alumina product were major results of these studies. The sodium concentration of the feed was shown to have a significant effect on the formation of alpha alumina in the product. Neither the presence nor absence of mercury in the feed had any detectable effect on calciner operation or product properties. (auth)

560 (IDO-14620) ICPP WASTE CALCINING FACILITY SAFETY ANALYSIS REPORT. L. T. Lakey and J. R. Bower, eds. (Phillips Petroleum Co. Atomic Energy Div., Idaho Falls, Idaho). Dec. 1, 1963. Contract AT(10-1)-205. 342p.

A study was made of the radiological hazards associated with operation of the ICPP Waste Calcining Facility. In addition to the normal release of small amounts of radioactivity with the process off-gas, possible accidental releases of larger amounts of radioactivity through operational error or equipment failure were examined. Bypassing of off-gas cleaning elements, uncontrolled chemical reactions, overheating by fission product heat, corrosion, abrasion, vibration, overpressure, utility failures, and major catastrophes were studied as possible causes of accidental radioactivity releases. No major safety deficiencies were found, and the facility appeared satisfactory for processing of high-level radioactive wastes. Details of the safety analysis and information on the process, equipment, and operation procedures are furnished. (auth)

561 (IDO-14622) A COMPARISON OF VARIOUS CALCINATION PROCESSES FOR PROCESSING HIGH-LEVEL RADIOACTIVE WASTES. B. R. Wheeler, J. A. Buckham, and J. A. McBride (Phillips Petroleum Co. Atomic Energy Div., Idaho Falls, Idaho). Apr. 1964. Contract AT(10-1)-205. 76p.

Published information and data relative to the four principal radioactive waste calcination techniques, the fluidized bed, the pot, the radiant-heat spray, and the rotary-ball kiln are summarized on a comparative basis. An ultimate combined capacity of about 1000 gph is expected to be necessary to process generated and accumulated wastes between the years 1965 and 2000. The effects on product properties and on economics of ultimate waste disposal of the type of calciner selected are considered from many standpoints, including the composition, concentration, and activity level of the waste solution and the calcination conditions. Also discussed are creation of glass-like compounds from calciner products, personnel and equipment requirements, and the present status of development of each of the calciners. One or more calciners may be suitable to process any particular waste, depending on the desired type of product and other circumstances. (auth)

562 (IDO-14626) ANALOG SOLUTION OF A MODEL OF THE SOURCES OF ELUTRIATABLE FINES IN THE FLUIDIZED BED CLACINATION PROCESS. Earl S. Grimmett (Phillips Petroleum Co. Atomic Energy Div., Idaho Falls, Idaho). Feb. 28, 1964. Contract AT(10-1)-205. 26p.

A model is proposed which depicts the sources of elutriatable fines in the fluidized bed calcination process as being in two major groups, spray drying mechanisms and attrition mechanisms. Based on this model, equations are derived which express the rate of change of the concentration of a chemical tracer material in the elutriated fines, following introduction of the tracer into the feed and following its removal from the feed. This system has been simulated on an analog computer, and by matching the computer simulation to results from an actual calciner run, the rates of generation of fines by each of the two groups of mechanisms has been determined; the same technique results in an estimate of the amount of these fines remaining in the fluidized bed. Agreement between postulated results and results of actual experimental tests lends credence to the usefulness of this analytical technique. (auth)

563 (IDO-14627) DEVELOPMENT OF A FLUIDIZED BED CALCINATION PROCESS FOR ALUMINUM NITRATE WASTES IN A TWO-FOOT-SQUARE PILOT PLANT CALCINER. PART IV. FINAL PROCESS STUDIES—RUNS 23 THROUGH 37. B. P. Brown, B. M. Legler, and L. T. Lakey (Phillips Petroleum Co. Atomic Energy Div., Idaho Falls, Idaho). June 1964. Contract AT(10-1)-205. 194p.

Final process studies, comprising fifteen runs (Runs 23 through 37), were made on the calcination of aluminum nitrate wastes in a two-foot-square calciner to provide support data for the Idaho Chemical Processing Plant Waste Calcining Facility (WCF) and to study further the fluidized bed calcination process. For each run, the objectives, conditions, and results are given. Final testing of the feed nozzle selected for use in the WCF and demonstration of the control of the crystallinity of the product were major objectives of these studies. Factors affecting the generation of elutriatable solids from the calciner were determined. Properties of the fines elutriated from the calciner and additional product properties were also determined. New experimental techniques were developed for control of particle size and for measurement of the rate of growth of particles. (auth)

564 (IDO-14629) DEVELOPMENT OF AIR DISTRIBUTOR CAPS FOR A FLUIDIZED BED. J. C. Petrie and D. E. Black (Phillips Petroleum Co. Atomic Energy Div., Idaho Falls, Idaho). Apr. 1964. Contract AT(10-1)-205. 20p.

Results are presented of a pilot plant study of factors which affect erosion and performance of caps installed on an air distributor plate in a fluidized bed calciner. The study was initiated because of the excessive erosion experienced on the original air distributor caps installed in the calciner vessel at the ICPP Waste Calcining Facility. Laboratory techniques which considerably reduce the time required to obtain erosion data in tests are described. A theoretical orifice formula is given which accurately predicts the pressure differential across the geometrically complex orifices in the air distributor plate. Details are given of the air distributor cap selected for use in radioactive operation of the Waste Calcining Facility as a result of these tests. (auth)

565 (IDO-14631) PNEUMATIC ATOMIZING NOZZLES IN FLUIDIZED-BED CALCINING. PART II. PROCESS TESTS. B. M. Legler and B. P. Brown (Phillips Petroleum Co. Atomic Energy Div., Idaho Falls, Idaho). July 1964. Contract AT(10-1)-205. 31p.

Results of both pilot plant and plant process tests showed that conventional pneumatic atomizing nozzles constructed of Types 347 or 440-C stainless steel provided satisfactory feed introduction to fluidized bed calciners when the nozzle spray zone was adequately submerged in the fluidized material. When feeding aluminum nitrate solution to a calciner at a given rate, variation of the nozzle air rate was effective in controlling particle size within the fluidized bed without the use of other attrition devices. Erosion of feed nozzles, indicated to be severe in early pilot plant tests, was negligible in the plant-scale calciner at the Waste Calcining Facility. Where erosion was a problem, boron carbide provided complete erosion resistance, but the inherent brittleness of boron carbide often resulted in fracture during installation or disassembly. Temperature perturbations due to the presence of the nozzle spray zone in the fluidized bed presented no serious difficulties, and caking of the bed material was negligible over a wide range of operating conditions. Generation of fines attributable to nozzle effects, although greater than expected, was within acceptable limits. A nozzle feed rate control system employing electromagnetic flowmeters significantly improved the performance of the nozzles in the Waste Calcining Facility. Although all testing in this study involved non-radioactive simulated waste solutions, the ultimate application is for high-level radioactive waste solutions. (auth)

566 (IDO-14633) DRY COLLECTION AND DISPOSITION OF SOLIDS FROM FLUIDIZED-BED OFF-GAS. J. C. Petrie and D. E. Black (Phillips Petroleum Co. Atomic Energy Div., Idaho Falls, Idaho). July 1964. Contract AT(10-1)-205. 27p.

Summarized operating experience is presented for cyclones used in the Waste Calcining Facility and in pilot plant calcination systems at the Idaho Chemical Processing Plant; results of test stand studies of parameters affecting cyclone collection efficiency are also given. Recommended operating conditions for cyclones handling abrasive and friable alumina particles are given. The present method of remote handling and disposition of radioactive material collected by the cyclone at the Waste Calcining Facility is described. Other methods considered for handling and disposition of cyclone fines are also discussed. (auth)

567 (IDO-14653) FLUIDIZED BED CALCINATION OF SIMULATED ZIRCONIUM FLUORIDE WASTE IN EXPLORATORY PILOT PLANT TESTS. Petrie, J. C.; Bailey, E. J.; Rhodes, D. W.; Legler, B. M. (Phillips Petroleum Co., Idaho Falls, Idaho. Atomic Energy Div.). July 1965. Contract AT(10-1)-205. 37p. Dep. mn; CFSTI $2.00 cy, $0.50 mn.

An exploratory pilot plant study of the fluidized bed calcination of zirconium fluoride waste with added calcium nitrate has shown that greater than 99 percent of the fluorides in the waste are retained in the calcined solids. Laboratory studies have indicated that mole ratios of calcium to fluoride considerably below the stoichiometric ratio were effective in suppressing fluoride volatilization. Corrosion tests conducted during the pilot plant study demonstrated that the austenitic stainless steels of which the Waste Calcining Facility is constructed will not be excessively corroded during calcination of zirconium fluoride waste in that facility. (auth)

568 (IDO-14655) ANNUAL REPORT OF DIVISION ANALYTICAL BRANCH FOR 1964. Shank, Ralph C. (Phillips Petroleum Co., Idaho Falls, Idaho. Atomic Energy Div.). Apr. 1965. Contract AT(10-1)-205. 212p. Dep.; $6.00(cy), 4(mn) CFSTI.

569 (IDO-14666) A FLOW DIVERTER-VALVE FOR GRANULAR SOLIDS. Petrie, J. C.; Cooper, E. D. (Phillips Petroleum Co., Idaho Falls, Idaho. Atomic Energy Div.). Mar. 1966. Contract AT(10-1)-205. 17p. Dep. mn. CFSTI $1.00 cy, $0.50 mn.

The design and operating characteristics of a nonmechanical, pneumatically operated, solids flow diverter-valve are described. The device is useful in diverting the flow of granular solids from a common supply header to any one of several terminals. The absence of moving parts makes the device attractive for use where minimum maintenance is important.

570 (IDO-14670) STABILITY STUDIES OF HIGHLY RADIOACTIVE ALUMINA CALCINE DURING HIGH TEMPERATURE STORAGE. Wilding, M. W.; Rhodes, D. W. (Phillips Petroleum Co., Idaho Falls, Idaho. Atomic Energy Div.). Jan. 1966. Contract AT(10-1)-205. 13p. Dep. mn. CFSTI $1.00 cy, $0.50 mn.

Stability of highly radioactive alumina calcine from the Waste Calcining Facility was studied in a temperature range from 600 to 1200°C. Strontium-90 and cerium-144 were not volatile. Cesium-137, ruthenium-106, boron, mercury, and nitrate were volatile to some degree over this temperature range. The alumina produced in the WCF was amorphous but converted to alpha alumina at 800°C and higher temperatures.

571 (ORNL-TM-715) ENGINEERING STUDIES ON POT CALCINATION FOR ULTIMATE DISPOSAL OF NUCLEAR WASTE: FORMALDEHYDE-TREATED PUREX WASTE FOR 1965 (FTW-65). C. W. Hancher, J. C. Suddath, and M. E. Whatley (Oak Ridge National Lab., Tenn.). Jan. 6, 1964. Contract W-7405-eng-26. 182p.

Nine continuous evaporator-calciner tests were made with simulated formaldehyde-treated Purex waste (FTW-65). The average processing rate was 40 liters per hour. The calcined solids had an average bulk density of 1.27 g/cc and a residual nitrate concentration of 100 to 400 ppm. The overall decontamination factor was about 7,000 for

non-volatile ions such as iron, 3,000 for ruthenium, 1,400 for sulfate, and 17 for mercury. Attempts to trap the mercury in the calciner off-gas were not successful. Control of the close-coupled evaporator and calciner was generally good. Processing waste containing degradation products of tributylphosphate caused some control difficulties because of foaming. Severe corrosion of the calciner occurred when fluoride was present in the feed. Continuous addition of calcium nitrate in proportion to the feed rate improved the sulfate decontamination. An attempt to produce a glass failed because of foaming and plugging of the calciner off-gas line. (auth)

572 (ORNL-TM-839) CALCINER POT FOR LIQUID WASTE CALCINATION PROGRAM. Design Report. R. M. Beckers (Oak Ridge National Lab., Tenn.). Apr. 17, 1964. Contract W-7405-eng-26. 3p.

Criteria and assumptions used in design calculations for a 8-in. 304L stainless steel calciner pot are presented. Estimates of the rupture strength of the shell under various conditions are presented. Five engineering drawings are available as CAPE-1150. (M.C.G.)

573 (ORNL-TM-857) STATUS OF THE ORNL POTCAL AND POTGLASS PROCESSES FOR THE CONVERSION OF HIGHLY RADIOACTIVE LIQUID WASTES TO SOLIDS. R. E. Blanco, J. O. Blomeke, W. E. Clark, E. J. Frederick, H. W. Godbee, C. W. Hancher, J. M. Holmes, and J. C. Suddath (Oak Ridge National Lab., Tenn.). May 8, 1964. Contract W-7405-eng-26. 38p.

The POTCAL (pot calcination) and POTGLASS (fixation of wastes in glassy solids) processes for conversion of highly radioactive wastes to solids are being developed. Economic and safety analysis of the processes were made. The equipment is described and corrosion problems discussed. (M.C.G.)

574 (TID-7699(p.1-23)) STATUS OF FLUIDIZED BED CALCINATION DEVELOPMENT AND PLANT OPERATION AT ICPP. J. A. Buckham, G. E. Lohse, and A. L. Ayers (Phillips Petroleum Co. Atomic Energy Div., Idaho Falls, Idaho).

A description is given of the process and equipment development for the Waste Calcining Facility. The final cold testing and initial radioactive operation of the facility are discussed. The properties of the product are listed. Off-gas cleanup and safety analysis are discussed. Volatility tests and preparation of glass from calcined alumina are considered. The future prospects and use of the facility are discussed briefly. (D.L.C.)

575 (TID-7699(p.73-95)) STATUS OF RESEARCH, DEVELOPMENT, AND DEMONSTRATION PROGRAMS ON SOLIDIFICATION OF RADIOACTIVE WASTES. A. M. Platt, R. T. Allemann, C. R. Cooley, J. D. Kaser, J. D. Moore, F. P. Roberts, K. J. Schneider, and U. L. Upson (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.).

A summary is presented of research and development work carried out on waste solidification during the period Sept. 1962 to March 1964. The work includes: small-scale spray and pot calcination of high-level radioactive wastes, large-scale spray solidification of simulated wastes, and development of an engineering-scale prototype for demonstration of three promising solidification processes (pot calcination, spray solidification, and phosphate glass solidification). (D.L.C.)

576 CALCINING TECHNIQUES TO EASE NUCLEAR-WASTE WOES. Chem. Eng., 70: No. 7, 26-8(Apr. 1963).

Four calcination processes under study are described: (1) A rotating kiln that operates at 1500°F. Through the use of aluminum nitrate as a simulated waste, the kiln evaporates water and decomposes the aluminum nitrate to aluminum oxide, which goes into storage. The gases are separated, partially condensed, and scrubbed before they are vented. (2) A fluid-bed calciner that consists of a fluidized bed of previously-calcined material to which fresh feed is supplied. The calcined product is continuously removed. The off-gases are processed by separators, adsorption towers, then scrubbers. (3) Spray tower calcination in which waste slurry is atomized at top of heated tower to evaporate the solution, the particles drying and becoming calcined as they fall. The gases are similarly treated to the rotating kiln process. (4) Pot calcination, a batch procedure in which wastes are held for 20 to 30 hr, the pot serving as a permanent storage vessel. Gases are recycled to an evaporator. (TCO)

577 POT CALCINATION OF SIMULATED RADIOACTIVE WASTE WITH CONTINUOUS EVAPORATION. C. W. Hancher and J. C. Suddath (Oak Ridge National Lab., Tenn.). Chem. Eng. Progr., Symp. Ser., 60: 105-10(1964).

Process equipment and a control system were developed for an evaporation-calciner process for reduction of radioactive waste solutions to solids. A 25-liter evaporator close-coupled to an 8-in. dia by 90-in.-long calciner was operated and controlled successfully. The liquid filling rates for Purex waste and TBP-25 wastes were 20 to 30 liters/hr. The evaporator overhead decontamination factor was about 10^3. (auth)

578 A COMPARISON OF VARIOUS CALCINATION PROCESSES FOR HIGH-LEVEL RADIOACTIVE WASTES. B. R. Wheeler, J. A. Buckham, and J. A. McBride (Phillips Petroleum Co., Idaho Falls, Idaho). Trans. Am. Nucl. Soc., 6: 409-10(Nov. 1963).

579 HOT CELL OPERATING EXPERIENCE IN THE CALCINATION OF LIQUID WASTES. R. T. Allemann, R. L. Moore, F. P. Roberts, and U. L. Upson (General Electric Co., Richland, Wash.). Proc. Conf. Hot Lab. Equip., 11th, New York, 1963, 139-60(1963).

Pilot plant studies of the calcination of high-level radioactive liquid wastes are in progress at Hanford Laboratories. This work is part of a world-wide effort to fix radioactive wastes in stable solid media, both to minimize storage costs by reducing volume and containment requirements, and to provide a safer form for the storage of such wastes, and for the storage and shipment of mixed or separated fission products. The process operations of pumping, spray drying, calcination, melting, condensation, gas purification, and gas and liquid sampling, as well as calorimetry and gamma analysis, are performed within a 7 × 15-ft cell, behind four-foot-thick walls, by remote manipulation. Electrical resistive and inductive heating within the cell totals nearly 50 kilowatts. The nominal throughput of 1 gallon per hour of concentrated waste (typically about 1M in salts and 2.5M nitric acid) corresponds to the waste from about ⅓ ton of irradiated uranium per 24-hour day.

Some of the engineering and operational problems encountered, and some of the experience gained in one year of operation are described. (auth)

Centrifugation

See the following references:

89, 192

Coagulation

See also the following references:

19, 27, 102, 511, 791

580 SEPARATION OF RADIOACTIVE ISOTOPES FROM WASTE WATERS. Rauzen, F. V.; Solov'eva, Z. Ya. At. Energ. (USSR), 18: 623-6(June 1965).

Studies of radioisotope separation from waste waters by coagulation and ion exchange showed that consecutive use of these methods reduced the activity of the waste solutions to permissible levels. It was shown that ion exchange separation of radioisotopes from the waste solutions is a direct function of the salt content in the solution. (R.V.J.)

581 A DECONTAMINATION DEVICE FOR THE REMOVAL OF RADIOACTIVE STRONTIUM FROM WATER. F. Haghiri and G. E. Merva (Ohio Agricultural Experiment Station, Wooster). Soil Sci. Soc. Am. Proc., 28: 132(Jan.-Feb. 1964).

A Sr^{90}-removal device was designed for decontaminating runoff water from contaminated soils. The system is composed of coagulation tank, filter, cation-exchange resin columns, and check tank. Results indicate that the system is suitable for rapid and efficient decontamination of pond waters. (H.M.G.)

Decay Holdup

See the following references:

34, 87, 92, 103, 163, 860, 866

Denitration

See also the following references:

34, 70, 188, 189

582 (HW-76973(Rev.)) DENITRATION OF PUREX WASTES WITH SUGAR. L. A. Bray (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Apr. 1963. Contract AT(45-1)-1350. 25p.

A process was developed for the destruction of HNO_3 in Purex-type waste solutions using sugar. Polyhydroxy compounds such as sucrose, fructose, glucose, and crude syrups such as black strap molasses rapidly and controllably decompose HNO_3 in aqueous solutions at temperatures above approximately 85°C. The number of moles of HNO_3 destroyed per mole of sugar fed is a function of the temperature of reaction and the moles of dissolved multivalent metal cations (such as iron and chromium). The efficiency of HNO_3 destruction does not vary with acid concentration over the range of 0.5 to 8 M. However, destruction of acid does vary with iron concentration. Approximately 12 to 20 moles of HNO_3 are destroyed per mole of sucrose fed as the iron concentration increases from 0 to 1 M. Products of the denitration reaction are all gaseous, consisting of oxides of carbon and nitrogen. The process was successfully tested in the high level hot cells, the cold pilot plant denitration unit, and in the Purex plant. (P.C.H.)

583 (HW-79622) FORMALDEHYDE TREATMENT OF PUREX RADIOACTIVE WASTES. R. C. Forsman and G. C. Oberg (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Oct. 1963. Contract AT(45-1)-1350. 36p.

Continuous denitration of Purex acid waste (PAW) with formaldehyde was successfully demonstrated in a full-scale plant prototype unit in the Hanford Purex plant. The reaction is smooth and is easily and safely controlled. Because severe foaming can occur in the reaction vessel, the use of an antifoam agent is required to attain acceptable plant processing rates. Extensive laboratory and pilot plant work, proper equipment design, and satisfactory operating procedures were the bases for developing adequate safeguards that assure complete control of the reaction at all times. During operation, the formaldehyde and the preheated PAW stream (with 50 to 100 parts antifoam per million parts PAW) are added continuously to the reactor which is maintained at 95°C. From an initial feed concentration of approximately 6.1M, the free acid is reduced to 0.5 to 1.0M in the treated waste. About 2.5 moles of free acid are destroyed per mole of formaldehyde added to the unit for a 60% reaction efficiency. A decontamination factor of 10^4 from feed to recovered acid is typical, which means that the ratio of radioactivity to HNO_3 in the recovered acid is reduced by a factor of 10,000 when compared to the same ratio in the high activity waste (PAW). Formaldehyde denitration of the Purex acid wastes reduces the chemical costs of waste treatment and storage. In addition, the process increases the storage capacity of underground tanks because of less salts in the waste, increases the flexibility of waste treatment equipment, and improves the quality of feed for fission product recovery. (auth)

Electrolytic, Ion Exchange, and Adsorption

See also the following references:

7, 12, 14, 19, 20, 24, 25, 27, 32, 34, 36, 52, 58, 59, 60, 71, 80, 81, 85, 87, 90, 92, 94, 96, 99, 103, 104, 106, 121, 122, 132, 133, 135, 146, 147, 149, 150, 154, 162, 179, 187, 188, 189, 190, 191, 192, 194, 197, 198, 199, 209, 210, 213, 217, 220, 226, 233, 235, 242, 243, 244, 246, 247, 249, 263, 387, 510, 511, 580, 581, 703, 708, 717, 737, 773, 775, 777, 786, 788, 794, 795, 875, 901, 904, 934, 938, 939, 941, 949, 950, 955, 956, 957, 959, 960, 961, 964, 965, 966, 968, 969, 970, 971, 973, 992

584 (A/CONF.28/P/784) RETENTION OF FISSION PRODUCTS BY PEAT HUMIC ACIDS, A NEW POSSIBILITY IN WASTE WATER TREATMENT. A. Szalay and M. Szilágyi (Magyar Tudományos Akadémia Atommag Kutató Intézete, Debrecen). May 1964. 11p.

The authors investigated the retention of fission products on peat preparations in laboratory tracer experiments. The experiments demonstrated that the majority of the fission products, the cations, are strongly fixed on peat humic acids if approximately favorable conditions are secured. The fission products can be classified according to thier fixation on humic acids into anionic and cationic groups. For an efficient binding, the pH of the aqueous phase must be kept between limits of about 4 to 6, optimally at about 5. The elution sequence of the long-living cationic fission products was investigated, and the "breakthrough volume" was determined under optimal laboratory conditions from chemically pure tracer solutions. The sorption capacity of one ton of peat is roughly about one thousand chemical equivalents. This is sufficient for the fixation of megacuries of Sr and Cs. The breakthrough volume—the tolerable amount of waste water—filtrable by a given quantity of peat is about 1000 to 10,000 times its quantity, if optimum conditions are secured. Further investigations concerned the possible interference of various non-radioactive chemical contaminants which might be present in industrial waste waters. The presence of salts, complexing agents and detergents can cause displacement or even remobilization of the sorbed fission products. The breakthrough volume is decreased if their concentration is excessive. The volume and weight of the contaminated peat humic acids can be reduced to a very small fraction by incineration before ultimate disposal. (auth)

585 (AERE-R-4905) INVESTIGATIONS INTO THE USE OF ION EXCHANGE RESINS FOR THE DECONTAMINATION OF RADIOACTIVE EFFLUENT. PART II. PILOT PLANT TRIALS. Clarke, J. H.; White, W. C. J.; Cumberland, R. F.; Wright, T. D. (Atomic Energy Research Establishment, Harwell (England)). Apr. 1965. Dep.; $1.00(BIS); 5s. (HMSO).

An ion exchange process for the treatment of radioactive effluent arising at Harwell, was studied using a 500 gph pilot plant. Average decontamination factors of 1,000 for α, and 1,250 for β activities were maintained during the treatment of 226,750 gallons. By using a modified regeneration scheme, as devised from laboratory work, the waste arisings from the plant plus those from a pre-filtration stage, were only 2.08% of the throughput.

586 (BNL-853) DECONTAMINATION OF SAVANNAH RIVER WASTE SUPERNATE BY ION EXCHANGE. H. Katz and M. Rothbart (Brookhaven National Lab., Upton, N. Y.). Feb. 1964. Contract AT-30-2-GEN-16. 21p.

A flow sheet was developed for the decontamination of Savannah River waste supernate by ion exchange, using Linde AW 400 molecular sieves. Satisfactory decontamination from cesium and strontium could be achieved with this material, the activity in the effluent being mainly ^{106}Ru. The presence of potassium in the waste seriously decreased the cesium capacity of all ion exchange materials studied. Linde AW 400 molecular sieve was the only material investigated having sufficient capacity in the presence of 0.05 molar potassium to be of practical consideration. The cesium capacity was not affected by the ionic composition of the waste except for sodium and potassium. In order to prevent chemical attack of Linde AW 400 molecular sieve by highly alkaline wastes, sodium aluminate must be added to the waste to a concentration of at least 0.04 molar. In order to prevent precipitation of sodium aluminate on the resin after the elution portion of the cycle, an ammonia rinse is necessary prior to the introduction of fresh waste. The proposed flow sheet was demonstrated both with simulated waste on a tracer level and with actual high-level waste supernate in hot cell experiments. Results indicated no difference between simulated and actual waste solutions. (auth)

587 (BNWL-72) RUTHENIUM REMOVAL FROM INTERMEDIATE LEVEL RADIOACTIVE WASTES BY ELECTRODIALYSIS METHODS. L. F. Coleman and J. R. Eliason (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). Apr. 1, 1965. Contract AT(45-1)-1830. 27p. Dep.(mn); $2.00(cy), 1(mn) CFSTI.

Electrodialysis was demonstrated to be capable of removing ruthenium from nitric acid wastes, but the economics for removing greater than 90 percent of the ruthenium activity by only an electrodialysis step are not considered favorable. Electrodeionization through packed resin beds (a special application of electrodialysis) was shown to be a feasible and economical method for removing ruthenium from acid and alkaline process condensates containing on the order of 500 ppm dissolved solids. Laboratory results show that 95 percent of the ruthenium can be removed from acidic and alkaline condensates at up to fiftyfold volume reductions. (auth)

588 (BNWL-180) ADSORPTION OF TRACE IONS FROM INTERMEDIATE-LEVEL RADIOACTIVE WASTES BY ION EXCHANGE. Mercer, B. W. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). Jan. 1966. Contract AT(45-1)-1830. 33p. Dep. mn. CFSTI $2.00 cy, $0.50 mn.

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

589 (CEA-R-2851) SORPTION DU ^{90}Sr SUR UN OXYDE D'ALUMINIUM. (Sorption of ^{90}Sr on an Aluminum Oxide). Rzekiecki, Roger (Commissariat a l'Energie Atomique, Saclay (France). Centre d'Etudes Nucleaires). Nov. 1965. 59p. Dep. mn.

The sorption mechanism of an alkaline earth bivalent cation on activated aluminum oxide was compared to the sorption mechanism of a monovalent ion. The selective retention of the alkaline earth cation was used to remove trace amounts of ^{90}Sr from radioactive waste waters. A kinetic study is reported, leading to the calculation of an industrial column.

590 (CONF-179-1) ION EXCHANGE ABSORPTION OF CESIUM BY POTASSIUM HEXACYANOCOBALT(II) FERRATE(II). W. E. Prout, E. R. Russell, and H. J. Groh (Du Pont de Nemours (E. I.) & Co. Savannah River Lab., Aiken, S. C.). Oct. 23, 1963. Contract AT(07-2)-1. 14p.

From the American Chemical Society Radionuclidic Exchange on Soils, Minerals, and Resins, Philadelphia, Apr. 1964.

A description is given of the preparation and properties of a granular form of potassium hexacyanocobalt(II) ferrate(II) that is a highly selective absorbent for cesium ion. The material is suitable for use in a large-scale ion exchange column, and offers the possibility of isolating and concentrating ^{137}Cs from the fission product waste solutions which arise from the processing of nuclear fuels. (auth)

591 (CONF-179-5) REMOVAL OF RADIOISOTOPES FROM SOLUTION BY EARTH MATERIALS FROM EASTERN IDAHO. M. W. Wilding and D. W. Rhodes (Phillips Petroleum Co. Atomic Energy Div., Idaho Falls, Idaho). [196?]. Contract AT(10-1)-205. 12p.

From American Chemical Society Radionuclidic Exchange on Soils, Minerals, and Resins, Philadelphia, Apr. 1964.

Naturally occurring earth materials from Idaho, primarily from localities near the National Reactor Testing Station (NRTS), were used in laboratory tests for the removal of radioisotopes from aqueous solutions. These earth materials included lignitic deposits, clay-like materials, and specific minerals; ion exchange resins were also considered for a specific application. The aqueous solutions were low-level radioactive cooling water or synthetic solutions made up to represent low-level radioactive wastes at the NRTS. Cation exchange capacities and other properties which affect the removal of radioisotopes from solution were determined. The cation exchange capacities varied from 0.006 to 1.0 meq/g of solid. Earth materials with cation exchange capacities greater than 0.3 meq/g, in general, had distribution coefficients in excess of 1000. The highest distribution coefficients for cesium and strontium occurred in the pH range from 6.0 to 9.0. The possible use of these materials for decontaminating low-level radioactive waste at the NRTS is discussed. The result of laboratory studies using these materials and an organic ion exchange resin for decontaminating a specific NRTS waste are given. A material high in clinoptilolite from a location near the NRTS was considered to be the most promising material for use in large beds or ion exchange-type columns. (auth)

592 TREATMENT OF STRONTIUM-90 BEARING ORGANIC WASTE BY ION EXCHANGE. E. Edgerley, Jr. (Washington Univ., St. Louis) and A. D. Ray. 23p. (CONF-179-10)

From American Chemical Society Radionuclidic Exchange on Soils, Minerals, and Resins, Philadelphia, Apr. 1964.

A laboratory study was undertaken to determine the ability of cation resin to remove a cationic radionuclide form from a waste stream in the presence and in the absence of interfering components. Additional studies were made on the removal of ^{90}Sr from a complex organic waste stream in a small plant treatment facility. Special attention was given to leakage or premature breakthrough phenomena of radiostrontium from columns of cation exchange resin. The relationship between leakage amount and the effective resin contact time in the exchange was determined, and the effect of liquid waste stream characteristics on leakage amounts was ascertained. A strong acid, high capacity, sulfonated polystyrene divinyl benzene resin, Duolite C-20 (Na^+), was used. Decontamination of liquid wastes containing ^{90}Sr by ion exchange is described adequately by ion exchange theory. The presence of organic matter in the liquid waste stream exerts at least a two-fold effect: loss of exchange capacity of the resin; and increase in quantity of leakage of ^{90}Sr. Studies made on a plant scale treatment facility are described briefly. (P.C.H.)

593 SORPTION OF CESIUM FROM DILUTE SOLUTIONS. B. L. Sawhney (Agricultural Experiment Station, New Haven). Soil Sci. Am. Proc., 29: 25-8 (Jan.-Feb. 1965). (CONF-566-2)

Sorption of Cs by a synthetic cation-exchange resin and by several clay minerals, over a range of low Cs concentrations, was studied in the presence of different counterions. An ion-exchange equation was used to predict the sorption of Cs from dilute solutions. Exchange isotherms showed that the sorption of Cs by the resin followed these predictions: Exchange isotherms were straight lines with unit slope. Sorption of Cs by clay minerals, however, did not follow these predictions except when K^+ was the competing cation. When Ca^{2+} and Al^{3+} were the competing cations, the exchange isotherms for montmorillonite were straight lines with less than unit slope, showing that a larger proportion of Cs was sorbed with decreased Cs concentrations. The exchange isotherms of illite and vermiculite, on the other hand, had at least two distinct slopes suggesting that more than one type of exchange site is involved in the sorption of Cs by these minerals. These conclusions were also supported by estimates of the equilibrium constants calculated from Vanselow's equation.

594 (DP-876) REMOVAL OF CESIUM FROM EVAPORATOR CONDENSATE. William E. Prout and Edwin R. Russell (Du Pont de Nemours (E. I.) & Co. Savannah River Lab., Aiken, S. C.). Mar. 1964. Contract AT(07-2)-1. 17p.

An ion exchange process was developed to remove cesium from condensate solutions produced when aged highly radioactive waste solutions are evaporated at the Savannah River Plant. The cesium is removed from the condensate in a single pass through a fixed bed of "Decalso," an aluminosilicate gel. In laboratory tests with actual condensate, about 99% of the cesium was removed from 10,000 bed volumes of feed before regeneration of the bed was required. (auth)

595 (EUR-2199.e) ELECTRODIALYSIS AS AN AUXILIARY IN THE SEPARATION OF Cs AND Sr FROM ACID FISSION PRODUCT SOLUTIONS. A. Guery and L. H. Baetsle (Centre d'Étude de l'Énergie Nucléaire, Mol, Belgium). 1964. 13p. Dep.(mn).

A study was performed on the use of electrodialysis as a means for deacidulation and desalination of high level reprocessing waste solutions. Two types of electrodialysis cells were investigated: the two- and three-compartment cells. Despite the simplicity of the first type, it proved to be inadequate for deacidulation purposes because of the high ammonia formation in the catholyte which impedes subsequent ion exchange processes. A three-compartment cell turned out to be much more suitable if the feed solution does not contain appreciable amounts of salts. In the latter case, an amount of phosphoric acid equivalent to the total amount of bulk ions has to be added in order to avoid precipitation in the feed compartment. A special electrodialysis cell was designed for the desalination of ammonium nitrate solutions being used as elutriant for Cs from min-

eral ion exchangers. The feed solution is brought into the cathode compartment and the semi-permeable membrane is made of fired clay. The passage of current through this membrane with high electrical resistance, carries nitrate ions and heats the cell leaving an ammonia solution which evaporates. (auth)

596 (EUR-2497.e) STUDIES ON THE SEPARATION OF ^{137}Cs AND ^{90}Sr FROM FISSION PRODUCTS IN AN ACID MEDIUM ON MINERAL ION EXCHANGERS. Baetsle, L. H.; Van Deyck, D.; Huys, D.; Guery, A. (Centre d'Etude de l'Energie Nucleaire, Mol (Belgium)). 1965. 70p. Dep. mn.

Studies were made on the recovery of ^{137}Cs and ^{90}Sr by means of new inorganic ion exchangers such as molybdenum ferrocyanide, tungsten ferrocyanide, polyantimonic acid, and the preparation and use conditions of these exchangers. Work was also performed on zirconium phosphate, the deacidification of solutions, and the construction of a ^{137}Cs and ^{90}Sr separation cell.

597 (HW-79174) PROGRESS IN TREATMENT OF A RADIOACTIVE CONDENSATE WASTE. J. M. Skarpelos (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Oct. 1963. Contract AT(45-1)-1350. 80p.

A waste stream decontamination process based on ion exchange was selected for development. The process is centered on ion-exchange technology developed during recent years in the Hanford Laboratories and other atomic energy research centers. Early attempts at small engineering scale application of ion-exchange technology resulted in less than satisfactory results when compared to laboratory data. The difficulties were traced to nonradioactive impurities in the wastes. Steam stripping and filtering successfully removed these impurities, and excellent decontamination of all significant isotopes except ruthenium was achieved by ion exchange. Satisfactory ruthenium decontamination was demonstrated on waste volumes up to about 1000 column volumes, and the capacity for other isotopes ranged from 10,000 to 20,000 column volumes. Improvement in ruthenium decontamination is needed to simplify operations and to improve economics. Although a large-scale integrated process is yet to be demonstrated, a cost estimate of an ion-exchange process was made. The estimate of $7.50/1000 gal was based on a 2 ton/day fuels reprocessing plant which produces about 30,000 gal/day of condensate requiring treatment. The estimate was based on a process for alkaline condensate and it was assumed a process for acid condensate would cost about the same. This estimate is about $10/1000 gal less than the estimated cost for reevaporation. (auth)

598 (HW-SA-2940) SYNTHESIS OF A CLINOPTILOLITE-LIKE ZEOLITE. L. L. Ames, Jr. (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Mar. 4, 1963. Contract AT(45-1)-1350. 14p.

Hydrothermal experiments were run on the systems $SiO_2-Al_2O_3-Li_2O-H_2O$, $SiO_2-Al_2O_3-Na_2O-H_2O$, $SiO_2-Al_2O_3-Na_2O-Li_2O-H_2O$, and $SiO_2-Al_2O_3-K_2O-H_2O$. A clinoptilolite-like phase was formed in the $SiO_2-Al_2O_3-Li_2O-H_2O$ system, but not in any of the other systems, although the products included mordenite, analcime, quartz, and chabazite. X-ray diffraction patterns and heat- and acid-treatments confirmed that the clinoptilolite-like phase was indeed a clinoptilolite. (D.L.C.)

599 (HW-SA-3199) THE ADSORPTION OF CESIUM AND STRONTIUM ON ZEOLITES FROM MULTICOMPONENT SYSTEMS. B. W. Mercer, Jr., and L. L. Ames, Jr. (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Nov. 1, 1963. Contract AT(45-1)-1350. 28p. Dep.(mn); $2.00(cy), 1(mn) OTS.

Mass action quotients for Cs^+ and Sr^{+2} with the synthetic zeolites Linde 4A, 13X, AW-400, AW-500, Norton Zeolon, and the natural zeolite clinoptilolite, were determined in binary systems. A method of using binary exchange data to predict the amount of cations adsorbed by the zeolites from multicomponent systems was developed. Predicted values for Cs^+ and Sr^{+2} adsorption generally agreed with experimental values except for 4A and AW-400 with a quaternary mixture of Na^+, Cs^+, Sr^{+2}, and Ca^{+2}, where Cs^+ adsorption was greater and Sr^{+2} adsorption less than predicted. Experimental values of 1.5 meq of Cs^+ or 2.5 meq of Sr^{+2} adsorbed per gram of zeolite were determined for columns receiving feed solutions containing either Cs^+ or Sr^{+2}. The uptake of both Cs^+ and Sr^{+2} by 4A zeolite from a mixed feed was 0.9 meq of Cs^+ and 1.3 meq of Sr^{+2} per gram of zeolite. Liquid diffusion coefficients for Sr^{+2} in a ternary cation mixture were calculated from electrical conductivity data and found to vary from 7.4×10^{-6} cm^2/sec at 25C to 1.8×10^{-5} cm^2/sec at 80°C. Particle diffusion coefficients for Sr^{+2} in 4A zeolite were calculated from column breakthrough data and the average value found to vary from 1.6×10^{-7} cm^2/sec at 25C to 5.2×10^{-7} cm^2/sec at 80°C.

600 (HW-SA-3233) FIXATION OF RADIOACTIVE FISSION PRODUCTS ON INORGANIC ION EXCHANGE MEDIA. W. V. DeMier (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Oct. 16, 1963. Contract AT(45-1)-1350. 21p. (CONF-196-2)

From Electron Microscope Meeting, Denver, Colo., Aug. 1963.

The in-container ion exchange process was selected for storing cesium and strontium from high-level radioactive wastes at Hanford. The steps of the ion exchange-based packaging process are outlined, and the properties and capacities of several zeolite exchangers are given. The hot-through-gas technique was selected for dehydrating the bed in the storage container prior to storage, and drying rates were determined using air at 800°F. Off-gas moisture measurement is considered. The pressure drop, off-gas particulates, and thermal conductivity of the bed were studied. The problem of sizing the container for adequate heat removal is considered. (D.L.C.)

601 (HW-SA-3333) CHARACTERIZATION AND APPLICATION OF ZEOLITES FOR RADIOACTIVE WASTE TREATMENT. J. L. Nelson, L. L. Ames, and B. W. Mercer (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Feb. 13, 1964. Contract [AT(45-1)-1350]. 30p.

The various ways in which zeolites may be applied to radioactive waste treatment and disposal are described briefly. Some of the properties of zeolites studied are tabulated. Equilibrium and kinetic experiments were run on zeolites with Cs and Sr, and the results are presented. Experiments on actual applications of zeolites to synthetic and actual wastes were also made. Synthesis of zeolites is considered briefly. (D.L.C.)

602 (IDO-14624) REMOVAL OF RADIOISOTOPES FROM SOLUTION BY EARTH MATERIALS FROM EASTERN IDAHO. M. W. Wilding and D. W. Rhodes (Phillips

Petroleum Co. Atomic Energy Div., Idaho Falls, Idaho). Nov. 8, 1963. Contract AT(10-1)-205. 64p.

Naturally occurring earth materials from Idaho, many of which were from localities near the National Reactor Testing Station (NRTS), were used in laboratory tests to determine the most likely candidates for decontaminating low-level radioactive wastes at the NRTS. It was concluded that the material containing a high content of clinoptilolite was the most promising candidate for the following reasons: (1) excellent physical and chemical stability, (2) a cation exchange capacity of about 0.3 milliequivalent per gram, (3) ability to remove radioisotopes from solution over a wide pH range, and (4) the ready availability of thousands of tons of the raw material at a location near the NRTS. This conclusion was based on the results of column breakthrough experiments and batch equilibrium-type tests. The distribution coefficients of this material for cesium and strontium were determined to be greater than 1000 at favorable pH values. A brief study of the effect of complexing agents on the reaction of radioisotopes with the earth materials indicated that citrate depressed the absorption of ruthenium markedly and that EDTA depressed the adsorption of strontium, cobalt, and ruthenium more than the adsorption of cerium and chromium. (auth)

603 (IDO-14657) DECONTAMINATION OF RADIOACTIVE EFFLUENT WITH CLINOPTILOLITE. Rhodes, D. W.; Wilding, M. W. (Phillips Petroleum Co., Idaho Falls, Idaho. Atomic Energy Div.). July 1965. Contract AT(10-1)-205. 14p. Dep. mn; CFSTI $1.00 cy, $0.50 mn.

The removal of strontium-90 and cesium-137 from fuel storage basin effluent by ion exchange using the mineral clinoptilolite as the exchanger is described. The concentration of cesium-137 in the effluent after decontamination was less than its drinking water tolerance, and the concentration of strontium-90 was approximately equal to its tolerance.

604 (INR-609/XIX/D) STUDY ON REMOVAL OF SOME RADIONUCLIDES FROM SEWAGE WATER SOLUTIONS BY SORPTION ON PEAT. Kostyrko, A. (Institute of Nuclear Research, Warsaw (Poland)). Feb. 1965. 45p. Dep.

Studies were made on the interaction between peat moss and the following cations: Na^+, Ca^{2+}, Co^{2+}, Zn^{2+}, Sr^{2+}, Y^{3+}, Ru^{3+}, Cs^+, Ce^{3+}, Eu^{3+}, Tl^+, Hg^{2+}, and Pb^{2+}. Some comparative tests were performed also with humus-rich peat specimens, as well as with peat moss pretreated with 1N HCl. Peat moss was found to be hydraulically suitable as a column material. As was expected, distribution coefficients [K_d] for peat moss were less pH dependent than those for humus-rich peats. Total sorption characteristics [5-50 meq/100 g of peat] were distinguished from ion exchange capacities [5-40 meq/100 g of peat], and the order of selective affinity by peat for these nuclides was checked.

605 (INR-636/XIII/J) USUWANIE IZOTOPOW PROMIENIOTWORCZYCH Z ROZTWOROW WODNYCH PRZY UZYCIU ZELU KRZEMOWEGO. (Removal of Radioisotopes from the Water Solutions by the Use of Silica Gel). Horski, Jozef (Institute of Nuclear Research, Warsaw (Poland)). May 1965. 83p. Dep.

Studies were made on the use of silica gel for the removal of radioisotopes from aqueous solutions. The following radioisotopes separately or all together in mixture were used: ^{90}Sr, ^{95}Zr, ^{106}Ru, ^{137}Cs, ^{144}Ce. In some experiments ^{131}I in the presence of stabile isotopes was also used. In some cases nonisotopic electrolytes or capillary active agents were added. HCl, CO_2, H_3PO_4, Na_2HPO_4, or $FeCl_3$ were used for precipitation of the silica gel. Three ways for decontamination were proved: the removal of radioisotopes in static or dynamic condition with use of the silica gel previously precipitated and dried, and the third one the precipitation of the silica gel in the decontaminated solutions.

606 (NP-14178) ADSORCION DE PRODUCTOS DE FISION SOBRE UNA BENTONITA DE MENDOZA. Informe No. 106. (Adsorption of Fission Products on Bentonites From Mendoza. Report No. 106). Leopoldo Jose Anghileri (Argentina. Comision Nacional de Energia Atomica, Buenos Aires). 1964. 10p.

The efficiency of bentonites in the decontamination of radioactive liquid wastes by the adsorption of fission products was investigated. The effects of the bentonite concentration, pH of the solution, contact time, grain size, and temperature on the adsorption were determined. Separations of the order of 98% were obtained. The adsorbed activity could be fixed in the bentonite by thermal treatment. Thermal treatment assures fixation even at acid pH. (J.S.R.)

607 (NYO-2911-1) THE DEVELOPMENT OF AN EFFECTIVE ALUMINO-SILICATE ADSORBENT FOR RADIOACTIVE WASTE DISPOSAL. Final Report, November 1, 1961–October 31, 1963. W. L. Haden, Jr. (Minerals and Chemicals Phillipp Corp., Menlo Park, N. J.). July 1964. Contract AT(30-1)-2911. 48p.

A natural mineral exchange material may vary in composition and always contains certain amounts of extraneous minerals, which can affect the exchange characteristics of the overall mineral. When used in radioactive waste treatment, these "impurities" are an unknown variable. The development of a mineral synthesized from its basic components, as attempted in this study, would serve (1) to determine the role of extraneous minerals and the mineral's structure on stability and kinetics, (2) to determine the role of the exchangeable cation, (3) to determine the role of the process parameters used to synthesize the mineral, and (4) to provide some insight into the geologic history of the natural minerals. Synthetic zeolite materials were prepared from the $Na_2O-Al_2O_3-SiO_2$ system as both amorphous and crystalline products. No novel or especially useful products resulted. As clinoptilolite appeared to have many desirable properties, it was decided that clinoptilolite and related minerals should be the objective of the synthesis work. In the $Li_2O-Al_2O_3-SiO_2$ system, emphasis was placed on the high-silica region. Hydrothermal syntheses of two new mineral phases, designated MCP-II and MCP-III, were accomplished. The former is an acid-resistant zeolite having a characteristic x-ray pattern, while the latter is a layer-type mineral of the montmorillonite type, possibly of the composition $Li_2O \cdot (2Li_2O \cdot Al_2O_3) \cdot 8SiO_2 \cdot xH_2O$, where the lithium appears both as the exchangeable ion and also in the octahedral layer. Another zeolite designated MCP-I may be novel, although there are similarities to a zeolite referred to by Barrer as Species H. Also in this composition range, the synthetic clinoptilolite-like phase obtained by Ames was confirmed. In addition to the usually considered variables of oxide ratios and temperature, the structure of the raw materials is indicated as a very important variable in zeolite synthesis. In addition to the synthesis work, a procedure was developed for pelletizing natural clinoptilolite in the form of acid-resistant pellets and beads. No particular advantage over natural granulated clinoptilolite was found, however. (auth)

608 (ORNL-3585) PROPOSED HEAD-END TREATMENTS OF PUREX-1WW WASTE FOR FISSION PRODUCT RECOVERY BY ION EXCHANGE. W. C. Yee and W. Davis, Jr. (Oak Ridge National Lab., Tenn.). June 1964. Contract W-7405-eng-26. 35p.

Two methods are described for preparing Purex 1WW radioactive waste for recovering strontium and/or rare earths by cation exchange. One method involves complexing the major cationic impurities with citrate ion at pH 2.5, and processing a twentyfold diluted and treated waste solution through a column of resin (pre-equilibrated to pH 2.5 with ammonia) where the uncomplexed alkaline earth and rare earth elements are sorbed. The other method involves precipitating sulfate ion (as ferric sulfate) from Purex 1WW waste at 55 to 60% HNO_3, and precipitating strontium nitrate from the resulting nearly-sulfate-free solution from 85% HNO_3. Cation exchange is used to purify this precipitate further. Pretreatment (complexing) with citrate ion results in a fourfold and threefold improvement in resin capacity for strontium and rare earths, respectively, over that for untreated and tenfold diluted waste. Less than 1% of the ferric ion in the waste is sorbed on the resin. Pretreatment by a double precipitation concentrates the strontium by a factor of 2.5 to 3.0 and reduces by more than 200-fold the ratio of total trivalent cationic impurities to strontium. Resin capacity for strontium is improved by a factor of 20 over that for the citrate-treated waste. Radiation degradation of the resin in both processes is discussed in terms of expected resin life. If citrate complexing is used as the basis for separating strontium and/or rare earth elements from impurity ions, then operating time to 10% resin capacity loss would be about 2500 hr or 100 hr, depending on whether just strontium or strontium and rare earth elements were recovered. If the head-end precipitation process is used for removing impurities, then the time to 10% loss of resin capacity in the final purification would be only about 15 and 10 hr, respectively, depending on whether strontium or strontium and rare earth elements were recovered. (auth)

609 (ORNL-3697(p.49-53)) MINERAL EXCHANGE STUDIES. Tsuneo Tamura, D. G. Jacobs, Tohru Murano, and W. P. Bonner (Oak Ridge National Lab., Tenn.).

In sorbing trace concentrations of strontium, ideal ion exchange behavior, based on mass action expressions, was exhibited by several of the clay minerals under special conditions. Hydrous sesquioxides sorbed strontium preferentially, and the strontium sorption curve did not yield the predicted slope for ideal behavior. Conasauga shale was shown to behave according to the mass action law when the free iron oxide was removed. Ethylenediaminetetraacetate (EDTA) forms a more stable complex with calcium than with strontium. In a waste solution containing calcium as a major ionic component, the major break-through of strontium through an ion exchange column can be delayed by complexing part of the calcium with EDTA. However, the delayed break-through of strontium is gained at the expense of increased leakage of strontium through the ion exchange column.

610 RADIUM REMOVAL FROM URANIUM MILL EFFLUENTS WITH INORGANIC ION EXCHANGERS. Arnold, W. D.; Crouse, D. J. (Oak Ridge National Lab., Tenn.). Ind. Eng. Chem., Process Design Develop., 4: 333-7(July 1965). (ORNL-P-131)

The concentration of ^{226}Ra in uranium mill waste solutions is normally too high to permit their direct discharge to the environment. Radium was efficiently removed from simulated lime-neutralized acid waste by adsorption on a number of inorganic ion exchange materials, including barytes, Decalso (synthetic zeolite), and clinoptilolite (natural zeolite). In column tests, radium breakthrough was not significant until many thousands of bed volumes of aqueous feed had passed through the column. Adsorption efficiency decreased with increases in exchanger mesh size and in aqueous throughput rate. However, the radium capacity of the column was not changed appreciably by a fivefold change of radium concentration in the aqueous feed. Radium adsorption from highly contaminated acid wastes was not efficient. Ammonium nitrate (2M) eluted radium readily from clinoptilolite but less completely from barytes and Decalso. (auth)

611 (PAN-453/XIX) ION EXCHANGE COLLECTORS—A NEW TYPE OF ION EXCHANGER BED. A. Kostyrko (Polish Academy of Sciences. Inst. of Nuclear Research, Warsaw). Aug. 1963. 24p.

Method of producing the ion exchange collectors, their characteristics, and possibility of use is discussed. The results of experiments on $Sr^{90} + Y^{90}$ and Cs^{137} ion exchange in collector batch systems at different conditions are given after theoretical introduction. It is evident, from these data, that the collectors represent a convenient tool in ion exchanger investigations. It was found out that inhomogeneous magnetic fields exert a strong influence on course of sorption processes. First experimental results concerning this problem are given. (auth)

612 PRELIMINARY STUDY ON THE BEHAVIOR OF CLAY MATERIALS PASSING RADIOACTIVE SOLUTIONS. Guido Branca, Giuseppe Lenzi, and Renato Sinno (Università, Naples). Rend. Acad. Sci. Fis. Mat., Soc. Nazl. Sci. Napoli, (4), 29: 259-84(1962). (RT/PROT(63)12). (In Italian)

The possibility of making use of a montmorillonitic flegrea pozzuolana was examined for the decontamination of radioactive liquid wastes. The ^{90}Sr distribution coefficient of clay material was studied with regard to the particle size and to the moisture. Furthermore, the removal of such nuclide was determined depending on flow rate in the column. Afterwards the ion exchange capacity of the montmorillonitic component of pozzuolana was measured; thus the absorption mechanism interpretation is given and the occupied position of ^{90}Sr in the lattice defined on the basis of the results obtained by the x-ray data and the thermic-differential analysis. (auth)

613 (RT/PROT(65)19) A STUDY ON THE SORPTION PROPERTIES OF NATURAL TUFFS OCCURRING IN THE LAKE BRACCIANO REGION (ROME). Boenzi, D.; Dlouhy, Z., Lenzi, G. (Comitato Nazionale per l'Energia Nucleare, Rome (Italy)). June 1965. 18p. Dep. mn.

This study includes a preliminary classification of natural tuffs from the Lake Bracciano Region, according to their ability to take in microamounts of cesium and strontium. The materials studied were classified by means of criteria of distribution coefficients for both radionuclides. The most promising among these sorbents were studied in more detail. The influence of granulometry, of pH, and of the concentration of competing Na^+ and Ca^{++} ions, as well as the sorption kinetics, were the main criteria for comparing the selected materials. (auth)

614 (RT/PROT(65)29) A STUDY ON THE SORPTION PROPERTIES OF THE NATURAL TUFFS OCCURRING IN THE LAKE BRACCIANO REGION (ROME). Report No. 2. Boenzi, D.; Dlouhy, Z.; Lenzi, G. (Comitato Nazionale per l'Energia Nucleare, Rome (Italy)). Oct. 1965. 16p. Dep. mn.

Samples of 85 different materials were studied with respect to their sorption capacity for microamounts of ^{137}Cs and ^{89}Sr from low-activity waste solutions. The materials studied were classified according to certain criteria, chiefly as regards the influence on sorption of grain size, of pH, of the contact time, and of the concentration of competing Na^+ and Ca^{2+} ions. On a limited number of particularly interesting samples, sorption kinetics were also briefly studied. The classification of a total of 175 samples, tuffs (90 in the first and 85 in this second part of the work), has led to the identification of a number of very promising materials, predominantly volcanic tuffs, which meet the requirements of good natural sorbents and which may be used to advantage in the treatment of contaminated liquids.

615 (STI/DOC/10/28(p.9-20)) RADIOACTIVE WASTE MANAGEMENT AND ENVIRONMENTAL RESEARCH. (International Atomic Energy Agency, Vienna).

Experimental methods and results are reported from investigations of the possibility of using wood as a raw material for the preparation of inexpensive ion exchange materials for use in the treatment of radioactive waste. Cation exchange substances with relatively high capacities were produced at little cost but the cost of anion exchangers was not competitive with that of commercial resins. The influence of radioactive wastes on biological conditions in a river was investigated under carefully controlled laboratory conditions. Data are presented on the sorption of ^{95}Zr, ^{95}Nb, ^{91}Y, ^{144}Ce, and ^{32}P on river sediments; the uptake by algae of ^{32}P, ^{134}Cs, ^{137}Cs, ^{131}I, ^{89}Sr, ^{144}Ce, ^{95}Zr, ^{95}Nb, ^{103}Ru, ^{106}Ru, and U; and the uptake of ^{32}P and ^{137}Cs by fish. The uptake, accumulation, and exchange of ^{90}Sr, ^{137}Cs, ^{90}Y, and ^{89}Sr by various open-sea phytoplankton were measured. (C.H.)

616 (TID-20662) FIXATION OF CESIUM-137 BY SOIL CLAYS. Progress Report, 1963–64. (Connecticut. Agricultural Experiment Station, New Haven). Contract AT(30-1)-2955. 5p.

Sorption of Cs from a range of dilute solutions by a synthetic cation-exchange resin and by several clay minerals in the presence of competing cations was studied. The sorption of Cs from dilute solutions by some typical soils of Connecticut was also investigated. The sorption behavior of the soil clays is described in terms of their mineralogical composition using the exchange isotherms obtained with pure clay minerals. Selectivity functions for the clays were calculated. (M.C.G.)

617 (TID-20980) PART I. INORGANIC SORBENTS. B. L. Baker and T. H. Wilson. PART II. SOIL PERCOLATION STUDIES. E. R. Conway. Progress Report No. 9. (South Carolina. Univ., Columbia). July 30, 1964. Contract AT(38-1)-147. 90p.

618 (TID-21955) FIXATION OF CESIUM 137 BY SOIL CLAYS. Progress Report, 1962-1965. (Connecticut Agricultural Experiment Station, New Haven). Contract AT(30-1)-2955. 8p. Dep. mn; CFSTI, $1.00 cy, $0.50 mn.

The sorption of Cs from dilute solutions by silicate minerals, resins, and clay minerals in the presence of different competing cations was studied. The sorption of Cs by illite–vermiculite mixtures was studied in detail, and the effects of Al-interlayers were considered. Al-interlayers were determined by means of titration curves of H-saturated soil clays, Al salt solutions, H- and Al-saturated montmorillonite, and H- and Al-saturated resin. Kinetics of Cs sorption by illite and vermiculite were also studied.

619 (TID–22511) IMPROVED ALUMINO SILICATE ION EXCHANGERS. Final Technical Report. Calmon, C.; Grundner, W. T. (Ionac Chemical Co., Birmingham, N. J.). Oct. 1962. Contract W-7405-eng-26, Subcontract No. 2133. 63p. (IONAC–5002). Dep. mn. CFSTI $3.00 cy, $0.75 mn. For Oak Ridge National Lab., Tenn.

An investigation was conducted to determine the possibility of producing alumino-silicate ion exchangers with superior characteristics for the removal and/or recovery of cesium from radioactive wastes. The effect of changing certain variables in the preparation of these materials on the characteristics of the product, including such properties as the ion exchange capacity, selectivity for cesium, physical strength, stability to low and high pH, etc., was studied. Several materials were prepared which exhibit higher capacities than the standard commercially available materials. In addition to improved physical and chemical stability they also have improved (by factors as high as 7.5) selectivity for cesium present in very low concentrations in the presence of large amounts of sodium.

620 (UCD–472-112, pp 141-7) ION EXCHANGE REMOVAL OF ^{90}Sr FROM BIOLOGIC WASTES. Anderson, R. P.; Goldman, M.; Taie, M. (California Univ., Davis. School of Veterinary Medicine).

The fourth year of operation of the Sr-90 waste decontamination plant is reported. The average concentration of radioactivity per batch, 50 pC/ml, remained approximately the same as for 1963. The waste influent rate of 700 gal/day totalling 250,000 gallons during 1964 more than doubled that of 1963. Forty-eight mC were removed by ion exchange with an average decontamination factor exceeding 3000. A total discharge of 50 μC Sr-90 was made to the underground leaching field. (auth)

621 (UJV-936/63) AKTIVIROVANNYI SERNOKISLYI BARII V KACHESTVE SORBENTA MIKROSTRONTSIYA. CHAST I. (Activated Barium Sulfate Acting as a Sorbent for Microstrontium. Part I). Lubomir Berak (Ceskoslovenská Akademie Věd. Ústav Jaderného Výzkumu, Rež). June 28, 1963. 36p.

Decontamination of waters containing radiostrontium is complicated by the presence of geochemically similar soluble calcium salts. Experiments were conducted to evaluate the separation of microstrontium from calcium salts by activated $BaSO_4$. Maximum activation was affected by calcium sulfates. Calcium sulfate impurities interfered with the normal formation of $BaSO_4$ crystallites, promoting a submicroscopic particle mass of high internal depression. When placed in a solution, the $CaSO_4$ is rapidly leached leaving a large sorption active surface. Isomorphous cations present in Sr^{2+} solution fill the anomalous vacancies in the $BaSO_4$ lattice forming a mixed crystal. With Sr^{2+} concentrations $<10^{-3}$M the secondary sorption slowed the $BaSO_4$ recrystallization, and strontium retention was achieved by cation exchange between Ca^{2+} and Sr^{2+}. The method is considered efficient for decontamination of drinking water. (R.V.J.)

622 (UJV-973/63) ZPŮSOB DESAKTIVACE RADIOAKTIVNÍCH ODPADNÍCH VOD PROTIPROUDNOU SORPCÍ NA ZKOAGULOVANÉ SUSPENSI PŘÍRODNÍHO SORBENTU. (A Method of Deactivation of Radioactive Waste Waters by Countercurrent Adsorption of Coagulated Suspensions of Natural Adsorbents). Vladimír Veselý (Československá Akademie Věd. Ústav Jaderného Výzkumu, Řež). [1963]. 106p.

Research work is described on the development of an original method of deactivating radioactive waste waters

contaminated with artificial radioisotopes, principally ^{90}Sr and ^{137}Cs. For the deactivation, countercurrent contact of the contaminated waste water with a coagulated slurry of natural adsorbents is used. The most convenient natural material from a series of materials investigated proved to be the rhyodacitic tufa from Lower Hrabovce in Slovakia. This material also has the optimum sedimentation characteristic in the form of a suspended powder with particle size up to 0.1 mm. Investigation of the countercurrent adsorption showed that it is possible to apply the well-known laws of extraction to the system. The distribution coefficients for tracer cesium are quite high. For tracer strontium with a small calcium content a lower value is attained; however, the overall efficiency of separation in a three stage system will be at least one order higher than with the use of the classical single stage system. The efficiency of separation of tracer strontium is caused to increase by means of a preformed phosphate precipitate. A proposed technological scheme and some data for construction of a device with a capacity of 3.6 m^3 of waste water per hour are presented.

623 (UJV-1106/64) K PROBLEMATIKE PRIRODNYKH SORBENTOV V KHSSR I IKH PRIMENENIYA DLYA DEKONTAMINATSII OTBROSHYKH RADIOAKTIVNYKH VOD. (On the Problems of Natural Sorbents in Czech SSR and on the Applications of the Sorbents for Decontamination of Radioactive Liquid Waste). L. Berak and Z. Dlougi (Československá Akademie Věd. Ústav Jaderného Výzkumu, Řež). [1964]. 13p. Dep.(mn).

Aluminosilicates, such as bentonite and vermiculite, must possess high selectivity and high sorptive capacity, in order to qualify as a natural decontaminating agent for radioactive waste liquors. Therefore, a number of naturally available materials were tested as to their selective sorption with respect to ^{137}Cs and ^{90}Sr. It was found that rhyodacite tuff from the Lower Grabowitz district possesses excellent sorption properties. The pumice rhyolite specimens had a simple structure and without exhibiting any capillarity, were able to separate the radioactive contamination from waste liquors containing large amounts of dissolved salts. Its sorptive capacity, as expressed by its separation factor, varies between 5×10^4 to 1.4×10^5 ml/g. In actual use, the flow rate of the liquid through the absorption column, the length of the column and the concentration of the inactive ions in the liquid must be taken into account when designing a decontamination system.

624 (UJV-1108/64) OBORUDOVANIE DLYA PROTIVOTOCHNOI SORBTSII NA SUSPENZII ESTESTVENNYKH SORBENTOV. (Equipment for a Counterflow Sorption Using Natural Sorbent Suspensions). V. Vesely, I. Napravnik, S. Gaderka, M. Paar, and L. Mazel (Československá Akademie Věd Ústav Jaderného Výzkumu, Řež). [1964]. 14p. Dep.(mn).

A method for countercurrent sorption of radionuclides from aqueous waste, using bentonite and tuff was developed on a laboratory scale. The tuff showed relatively high extraction coefficients, especially for Cs. Values for Sr were lower and depended on Ca ion concentration. An industrial process carried out in conical-bottomed mixers is outlined. (IIC-M.G.)

625 (UJV-1112/64) OBEZVREZHIVANIE RADIOAKTIVIYKH SBROSNYKH VOD METODOM ELEKTROKHIMICHESKOGO ANODNOTO RASTVORENIYA METALLICHESKOGO ALYUMINIYA I ZHELEZA. (Decontamination of Radioactive Liquid Waste by the Method of Electrochemical Anode Dissolution of Metallic Aluminum and Iron). I. Napravnik, Ya. Krzhizh, and M. Paar (Československá Akademie Věd. Ústav Jaderného Výzkumu, Řež). [1959]. 14p. Dep.(mn).

The most widely used purification processes of waste waters possessing low-level radioactivity are based on sorption and precipitation mechanisms. Previous studies revealed that the hydrated oxides of Al and Fe cannot remove the radioiodine from the waste solutions but may be employed for the precipitation of fission-product mixtures; therefore, the otherwise little-used electrochemical dissolution of Al and Fe was adapted for this purpose. Anodic dissolution of these metals yields their hydrated oxides which adsorb certain ions and then move toward one or the other electrode, depending on their charge, where they are precipitated. The efficiency of the method depends on electrolytic and electrophoretic phenomena. The potential applicability of the hydrated oxides of Zn and Cu was also examined. The method was found useful for separating various radionuclides on phosphate deposits. Comparison of the properties of the chemically precipitated oxides with those obtained by anodic dissolution revealed that the latter ones are less sensitive toward the pH of the solution and may be used in a broader range. In the tests, carried out with solutions containing 1 milliequivalent/100 ml of solution, the biologically dangerous nuclides, such as radioactive Sr, Ru, and fission product mixtures were considered. The separation factor was found to be higher in the case of the anodically prepared material. A pilot plant with a capacity able to treat 1 m^3/hr of waste liquids is under construction for the purpose of determining the effect of various factors on the separation efficiency. (TTT)

626 (UJV-1296/65) THE SORPTION OF FISSION PRODUCTS ON THE SUSPENSION OF THE NATURAL SORBENTS COAGULATED IN AN ELECTRIC FIELD. Napravnik, J.; Kepak, F. (Ustav Jaderneho Vyzkumu, Ceskoslovenska Akademie Ved, Rez). Feb. 1965. 35p. Dep. mn.

Results are reported of the laboratory and pilot plant experiments of the sorption of fission products on the suspension of natural aluminosilicates coagulated in an electric field. This method proved to be very suitable for natural sorbents, because no losses of their sorption capacity occurred in contradiction with the chemical coagulation. The influence of the nature of electrodes, current density, temperature, and of pH on the electrocoagulation was determined. The influence of the mode of the coagulation on the sorption properties of used sorbents was investigated. Attention was also paid to the sedimentation time and to the volume of the resulting sludge. (auth)

627 (ZFK-RCH-2) ANWENDUNGSPROBLEME EINES TORF-TON-AUSTAUSCHERS BEI DER REINIGUNG RADIOAKTIVER LÖSUNGEN. (Use of a Peat-Clay Exchanger for the Purification of Radioactive Solutions). Guenter Sachse and Reinhart Winkler (Germany. Zentralinstitut für Kernphysik, Dresden). May 1963. 23p.

From montmorillonite clays and peat was prepared a cation exchanger with stable granular structure. The selective properties of the components made the exchanger suitable for the separation of cationically existing fission products from waste waters. The raw materials for the preparation of the exchanger are cheap and easily obtainable. Therefore, it is possible to forego regeneration and reuse. (auth)

628 (ZFK-RN-17(p.165-78)) DIE VERWENDUNG NATURLICHER KATIONENAUSTAUSCHER FUR DIE REINIGUNG RADIOAKTIVER ABWASSER. (The Application of Natural Cation Exchangers for the Purification of Radioactive Wastes). Winkler, R.; Sachse, G. (VEB Atomkraftwerk, Rheinsberg (East Germany)).

A bibliographic review is given of the applicability of natural cation exchangers for the purification of radioactive water. Mineral ion exchangers, fixation of radio-

active cations on the exchangers, the addition of clay in precipitation methods, natural organic exchangers, organo-mineral exchangers, and the use of natural ion exchangers in the purification of reactor water are considered. (J.S.R.)

629 ION EXCHANGE. Robert Kunin and F. X. McGarvey (Rohm and Haas Co., Philadelphia). Anal. Chem., 36: 142R-4R(Apr. 1964).

A review on ion exchange is presented, including discussion on theory, equilibrium and selectivity, concentration of dilute solutions, inorganic analysis, organic and biochemical analysis, and ion exchange papers. 75 references are included for the period Nov. 1961 to Nov. 1963. (P.C.H.)

630 DECONTAMINATION OF RADIOACTIVE WASTE WATER BY HUMIC ACID COLUMN. Matsumura, Takashi; Ishiyama, Toshio (Radiation Center of Osaka Prefecture). Ann. Rept. Radiation Center Osaka Prefect., 5: 33-6(Mar. 1965).

Humic acid has a characteristic cation exchange property. Decontamination of radioactive waste water by a humic acid column was tested. When only very small amounts of foreign matter are present in the water to be decontaminated, ^{89}Sr and $^{137}Cs-^{137}Ba$ are satisfactorily removed from neutral solution; $^{95}Zr-^{95}Nb$, $^{144}Ce-^{144}Pr$, and FP are effectively removed by acidification of the liquid, but nitrato-nitrosylruthenium is not removed effectively. When an appreciable amount of carrier ions is present, the removals of ^{89}Sr and $^{137}Cs-^{137}Ba$ decrease relatively quickly with the concentration of carrier ions. It seems that the cation exchange property of humic acid works mainly on the surface of the humic acid preparation. (auth)

631 EXTRACTION OF RADIUM FROM LIQUID WASTE BY MEANS OF MANGANESE DIOXIDE. Tyutrina, A. P.; Zhagin, B. P.; Bakhurov, V. G. At. Energ. (USSR), 18: 487-91(May 1965). (In Russian)

Application of amorphous manganese dioxide and pyrolusite for extraction of radium from uranium industrial waste waters was studied. Artificially prepared manganese dioxide coated on the sawdust or pyrolusite as suspension in a column are recommended as sorbents. The spent pyrolusite can be utilized as oxidizer of uranium ores. (R.V.J.)

632 KINETIC INVESTIGATIONS ON ION EXCHANGE PROCESSES WITH REGARD TO URANIUM FISSION-PRODUCT SEPARATION. Bobleter, O. (Oesterreichische Hochschule, Vienna). Ber. Bunsenges. Phys. Chem., 69: 874-84(1965). (In German).

A linear dependence of the break-through volume on flow rate was established with the aid of break-through curves obtained by loading anion exchangers with mineral acids. The exchange behavior at the front and in a larger region near the front of the electrolyte are mathematically described on the basis of a diffusion model. The equations permit the determination of layer thickness and diffusion coefficients in exchanging systems. Up to a maximum permissible flow rate (v_{max}), which could be theoretically and experimentally demonstrated, it is possible to obtain quantitative adsorption of exchangeable ions. The v_{max} values were determined for a separation process of uranium fission products using aluminum nitrate elution. It is thus shown that the highly active wastes from power reactors can be processed without noticeable radiation damage to the material of the organic exchanger.

633 THE ELECTRODEPOSITION OF RUTHENIUM FROM A RUTHENIUM(III) AND RUTHENIUM(IV) SOLUTION AND A FISSION PRODUCTS SOLUTION. Kobayashi, Yoshii; Yamatera, Hideo; Okuno, Hisateru (St. Paul's Univ., Tokyo). Bull. Chem. Soc. Jap., 38: 1911-15(Nov. 1965).

The electrodeposition of ruthenium from ruthenium chloride and ruthenium nitrosyl chloride solutions was studied and applied to the separation of the radioactive ruthenium contained in fission products. When the amount of ruthenium deposited in a given time was plotted against the cathode potential, the curves for 1×10^{-7} M solutions of chloride and nitrosyl chloride showed a maximum. The decrease in the electrodeposition at lower cathode potentials may be due to the anodic oxidation of ruthenium as well as to the evolution of hydrogen gas at the cathode. The rate of electrodeposition was found to depend on the material used as the cathode, and to be increased by an increase in the temperature and by the addition of platinum or ruthenium carrier. A complete separation of radio-ruthenium from fission products resulted when electrolysis was carried out with a 0.1 M hydrochloric acid solution of fission products containing carrier ruthenium. The ruthenium electrodeposited on the platinum electrode was completely recovered in the solution by anodic oxidation.

634 PROBLEMS OF DECONTAMINATION WITH ION EXCHANGERS. Friedrich Martinola. Tribune CEBEDEAU (Centre Belge Etude Doc. Eaux), 16: 459-64(Oct. 1963).

The use of ion exchangers constitutes a very important procedure for the decontamination of radioactively polluted waters. The simplicity of pile design of the installations and the reduced cost of the preparation due to its high efficiency permit economic exploitation in practice. In permanent installations for treatment of radioactive waste waters, it is advisable to regenerate the exchangers and reuse them a number of times. The exchanger should be employed only a single time, however, if it is necessary to obtain potable water. For salt-free waters, such as those used for cooling and moderating the piles, the ion exchanger is the only substance that, technically and economically, gives complete satisfaction in decontamination. (Public Health Eng. Abstr., 44: No. 8, Aug. 1964).

635 RETENTION OF FISSION PRODUCTS BY THE HUMIC ACIDS OF PEAT—A NEW METHOD FOR THE TREATMENT OF EFFLUENTS. Szalay, Sandor; Szilagyi, Maria (Inst. of Nuclear Studies, Hungarian Academy of Sciences, Debrecen). Fiz. Szemle, 15: 388-91(Dec. 1964). (In Hungarian)

Previous studies carried out at the Institute of Nuclear Studies revealed that the uranyl cation is strongly sorbed on the insoluble humic acids which behave similarly to the carboxylic synthetic ion-exchange resins. The sorption is not specific for UO_2^{2+} ions. The problem was studied by tracer method, examining the behavior of fission products in dilute aqueous HCl solutions. The peat used for the tests was collected from below the water table where it was protected from oxidative decay. It was found that the long half life, biologically dangerous cations, such as rare earths, Zr, Sr and Cs are strongly sorbed while anionic materials—I, Rn, Se, Nb, Sb and Te and the noble gases—are not retained. ^{95}Zr was most strongly held, it could not be eluted even with 1N HCl. But its daughter, ^{95}Nb, is not sorbed at all. The degree of sorption is strongly dependent on the valence of the cation. The method is considered potentially useful for the large-scale treatment of radioactive waste effluents containing fission products, provided that the liquors do not contain strong acids, keeping the acidity between 4 and 6, and are free from dissolved stable cations, such as Zr^{4+}, Fe^{3+} and Al^{3+}; complexing agents should also be absent. Although no large-scale tests have been carried out, it appears feasible to discharge the waste effluents into uninhabited marshy areas containing large amounts of peat. (TTT)

636 REMOVAL OF RADIOACTIVE SUBSTANCES BY KOREAN ACID CLAY. Noriyuki Mishio and Sho-ou Tcho (Ebara Mfg. Co., Ltd., Japan). Genshiryoku Kogyo, 10: No. 4, 34-7(Apr. 1964). (In Japanese)

Four kinds of acid clay from the Kyung-buk district in Korea were used as samples. The main components of these samples were found by x-ray and spectroscopic examination to be montmorillonite, zeolite, and quartz. After adding 1,000 ppm acid clay to a pseudo-waste solution of ^{137}Cs and ^{89}Sr (ca 10^{-2} $\mu C/ml$) and stirring for 30 min, the sample solution was allowed to stand for 30 min and the supernatant was taken out to measure the radioactivity. In general, baking of the samples lowers the decontamination ratio. The existence of $NaNO_3$ does not affect the decontamination of ^{137}Cs to a great extent; the decontamination of ^{89}Sr is, however, likely to be influenced by the existence of Na^+ and the decontamination ratio decreases appreciably. A pH value of 4–11 is suitable. In the treatment of actual waste solutions (5×10^{-6} C/ml), about 80% decontamination ratio was obtained. (NSA of Japan)

637 ON THE POSSIBILITY OF USING SOILS FOR THE PURIFICATION OF LOW RADIOACTIVITY SEWER WATERS. I. ADSORPTION OF Co^{60} BY THE SOIL OF PEAT BOGS. A. I. Berdnikov and V. F. Oreshko (Moscow Medical Inst.). Izv. Vysshikh Uchebn. Zavedenii, Khim. i Khim. Tekhnol., 7: 594-7(1964). (In Russian)

The adsorption by peak soils of ^{60}Co from aqueous solutions was investigated as a function of the nature, degree of decomposition, granulometric composition, concentration of the radioisotope, and pH of the medium. It was found that peat soils readily adsorb ^{60}Co from solution. The maximum rate of sorption was possessed by hypnum and wood-floodplain samples with low (10%) and medium (50%) degrees of decomposition. The radioactivity level of the solution exerted practically no influence on the sorption process. The maximum adsorption fell within the range of pH values from 5.0 to 8.0. The possibility of complexing ^{60}Co with the primary and secondary amino groups of peat, as well as with the sulfo groups in hydrochloric acid medium and with hydroxycarboxylic acids in alkaline medium is suggested as an explanation of the observed pattern of sorption. Peat soils are recommended as cheap natural absorbents in the purification of waters with low radioactivity.

638 THE INVESTIGATION OF ADSORPTION PROPERTIES OF PYROCLASTIC ROCKS FOR THE PURPOSES OF RADIOACTIVE WASTE WATER DECONTAMINATION. I. THE CLASSIFICATION OF PYROCLASTIC ROCKS. Zdenek Dlouhy (Inst. of Nuclear Research, Czechoslovak Academy of Sciences, Rez). Jaderna Energie, 11: 87-91 (Mar. 1965). (In Czech.)

Classification of a great group of natural silicates, pyroclastic rocks, on the basis of their adsorption properties and further criteria for their applicability in radioactive waste water decontamination was carried out. Materials, which maximally met all requirements, were investigated in more details. The best material, rhyodacite tuff from Nižný Hrabovec, was found to be very convenient for radiocesium adsorption from waste waters. The adsorption of radiostrontium under certain conditions [in the absence of calcium ions in treated solutions] may be realized as well. The possibility of the prompt utilization of the above mentioned material in pilot plant or on full operation scale is feasible. (auth)

639 STUDY OF THE SORPTION PROPERTIES OF PYROCLASTIC ROCKS, FOR THE DECONTAMINATION OF RADIOACTIVE WASTE WATER. PART 2. STUDY UNDER STATIC CONDITIONS. Dlouhy, Zdenek (Inst. of Nuclear Research, Czechoslovak Academy of Sciences, Rez). Jaderna Energie, 11: 207-12(June 1965). (In Czech.).

It was previously found that rhyodacite tuff from Nizny Hrabovec is the most suitable pyroclastic rock for the decontamination of radioactive waste water. Therefore, this mineral was investigated in greater detail, and the dependence of the adsorption of the radionuclide ion on the Ca^{2+} and Na^+ ions present was determined. The results obtained were in good agreement with the published data, except for the value of the distribution coefficient for Cs. It was concluded that for Cs the proportionality condition, as reported in the literature, of ion activities and molar fractions of these ions in solid phase is probably invalid. (CFSTI)

640 SORPTION OF STRONTIUM-90 BY SOIL. Hwa-sheng Cheng (Tsing Hua Univ., Hsinchu, China). J. Chinese Chem. Soc. (Taiwan), 11: 24-9(May 31, 1964).

The removal of ^{90}Sr by various clay deposits at pH values ranging from 0 to 14 was found to be slightly less effective than radiocesium uptake. The radiostrontium uptake by red and yellow soil was greatly affected by the foreign cations (0.01N). Yellow soil was particularly suited for the removal of ^{90}Sr. (auth)

641 SORPTION OF RADIOEUROPIUM BY SOIL. Hwa-sheng Cheng (Tsing Hua Univ., Hsinchu, China). J. Chinese Chem. Soc. (Taiwan), 11: 50-5(May 31, 1964).

The removal of radioeuropium (^{152}Eu, ^{154}Eu) was practically complete at pH values greater than 4 by seven clay deposits that were studied. The chemical properties of the radioeuropium rather than the nature of the sorbing matrices were the determining factors in its decontamination. The effects of competition toward radioeuropium uptake from solutions by, and extraction of sorbed radioeuropium from, yellow and red soil by various cations (0.01N) were studied. (auth)

642 ION EXCHANGE PROPERTIES OF VANADIUM FERROCYANIDE. II. DYNAMIC EXPERIMENTS. Krtil, J. (Nuclear Research Inst., Czechoslovak Academy of Sciences, Rez). J. Chromatog., 21: 85-91(Jan. 1966).

The use of vanadium ferrocyanide for the separation and isolation of heavy alkali metals under dynamic conditions was studied. The capacity of this sorbent for Cs, Rb, Sr, and Ce was determined, and the conditions for maximum sorption and elution of these elements were also determined. The application of vanadium ferrocyanide to the isolation of ^{137}Cs from a solution of fission products was tested.

643 EXCHANGE-ADSORPTION OF RADIOACTIVE CATIONS BY ZEOLITIC TUFF. Hiroshi Takahashi and Mutsumi Nishidoi. J. Clay Res. Group Japan, 3: 239-45 (Nov. 1963). (In Japanese)

The exchange-adsorption of radioactive ^{90}Sr and ^{137}Cs in batch system by zeolitic tuff from Kawarago, Shiroishi, Miyagi Prefecture was studied. The relation between the affinity of both radioactive cations and the form of exchangeable cation of adsorbent was investigated in NaCl chloride solution. In general, the mono-valent forms show the selectivity for ^{90}Sr except H-form, while the divalent forms do not show so remarkable selectivity. H-form only shows the highest selectivity for ^{37}Cs. Moreover, zeolitic tuff show the higher capacity for these both cations than other clay minerals, for example, kaolinite or montmorillonite. Since the zeolitic tuff is relatively soft mineral and swells very little in water, it is very effective and economical to apply it for the treatment of radioactive aqueous waste. (auth)

644 A NEW SERIES OF SYNTHETIC ACID STABLE MINERAL ION EXCHANGERS. I. FERROCYANIDE–

MOLYBDATE (FeMo). D. Huys and L. H. Baetslé (Belgian Nuclear Research Center, Mol). J. Inorg. Nucl. Chem., 26: 1329-31(July 1964).

The precipitate from a mixture of molybdate and ferrocyanide was found to possess particularly attractive properties for use as an ion exchange material. On mixing sodium molybdate in HCl solution with $H_4Fe(CN)_6$ a precipitate is obtained if the final ratios of H/Mo and Mo/Fe are higher than 2; below this ratio the efficiency of the precipitation is very poor. The product is reasonably stable in acids up to 0.2 M HCl and HNO_3, but is slowly attacked at higher acidities. A reducing medium helps to keep the material stable even in the presence of 2 M acid. The practical value of the new compound for the separation of fission products is demonstrated by the fact that Cs may be removed from concentrated reprocessing waste solutions.

645 SEPARATION OF CESIUM AND STRONTIUM BY ELECTRODIALYSIS. Hershey, H. C.; Mitchell, R. D.; Webb, W. H. (Univ. of Missouri, Rolla). J. Inorg. Nucl. Chem., 28: 645-9(Feb. 1966).

Electrodialysis methods employing Permutit anion exchange membranes (3148) and cation exchange membranes (3142) were investigated for the separation of strontium from cesium. The methods were based on the formation of strontium chelate complexes with EDTA and DTPA. Separations of radiostrontium and radiocesium in equilibrium with their daughter decay products were made with two-cell and three-cell electrodialysers. The best results were obtained with EDTA at an initial pH of 11.0 employing anion exchange membranes. Data suggest that this method is feasible for the separation and purification of radioactive cesium and strontium.

646 CONTINUOUS PURIFICATION OF WATER-COOLED POWER REACTOR PRIMARY COOLANT CIRCUIT BY ION EXCHANGE. R. Winkler and A. Schoenherr (VEB Atomkraftwerk, Betriebsteil Rheinsberg/Mark, Ger.). Kernenergie, 7: 741-8(Nov. 1964). (In German)

A comprehensive report on the problem and methods of purification of the primary circulation in ion exchangers is presented. The origin of impurities and their harmful effect are discussed in general terms and on the basis of specific examples. Next the chemical and technical aspects of ion exchange are considered with emphasis on impurities whose major parameters are the pH-value and the sizes of crude particles. A discussion follows of mechanical filter design and performance, and treatment of the ion exchanger. Some data are given on operating temperatures and pressures. Forty four references are listed.

647 UTILIZATION OF MULTICLONE DUST FOR DECONTAMINATING SEWAGE WATERS CONTAINING ^{137}Cs AND $^{90}Sr/^{90}Y$. Otto, R.; Hecht, P. (Institut fuer Angewante Radioaktivitaet, Leipzig). Kernenergie, 8: 643-5(Nov. 1965). (In German).

The results of some experiments concerning the use of multiclone dust for decontaminating sewage waters containing ^{137}Cs and $^{90}Sr/^{90}Y$ are given. Multiclone dust is a product of gasifying coke in the Winkler generator. Studies carried out on a laboratory scale using model waters showed that the surface-active material is suitable for the separation of the radionuclides mentioned.

648 EXCHANGE ADSORPTION OF STRONTIUM ION BY CALCINED MONTMORILLONITE. Mutsumi Nishidoi and Zenzo Yokozeki (Kurita Ind. Co., Ltd., Japan). Nippon Genshiryoku Gakkaishi, 5: 985-9(Dec. 1963). (In Japanese)

A new swellingless and large particle exchanger was prepared by calcination of bentonite and its adsorbability for strontium ion was examined. The calcination is not affective to the exchange capacity up to 500°C; it decreases with the rise of calcination temperature above 500°C. The rate of the decrease in its capacity was maximum in the exchanger treated with sodium hydroxide. Although a simple relation between the concentration of acid and the rise of the capacity was not found, there is a semi-logarithmic relation between the concentration of sodium hydroxide and the capacity. Only Na-form exchanger and Natural- and H-form can be used in column for the removal of strontium. It was concluded that Natural-form bentonite has selectivity for cesium ion as the result of very low capacity for strontium comparing with cesium. However, montmorillonite possibly has a high capacity for strontium corresponding to cesium in the treatment with sodium hydroxide. (auth)

649 SEPARATION OF ^{137}Cs FROM FUEL REPROCESSING WASTE WITH FERROCYANIDE EXCHANGER. Noriyuki Mishio (Ebara Mfg. Co., Ltd., Japan). Nippon Genshiryoku Gakkaishi, 6: 2-7(Jan. 1964). (In Japanese)

The highly radioactive waste that arises from the processing of irradiated fuel is too hot to handle and hence, in current practice, is stored without treatment in underground tanks for hundreds of years. For removing ^{137}Cs from the liquid waste, a special solid ion-exchange material, a ferrocyanide-bentonite mixture made by a freezing-thawing method, was developed. This new absorption material has shown its capability of adsorbing Cs exclusively even from 6N nitric acid solution containing various other ions, and ^{137}Cs has been successfully separated from synthetic Purex waste and Zirflex waste with the new material. Elution of 80% of the ^{137}Cs from this ferrocyanide-bentonite exchanger was accomplished by use of silver nitrate solutions. (Nucl. Sci. Abstr. of Japan, 3: No. 2 (Apr. 1964))

650 ION EXCHANGE OF RADIOACTIVE ION WITH ZEOLITIC TUFF. Nishidoi, Mutsumi (Kurita Industry Co., Ltd., Japan). Nippon Genshiryoku Gakkaishi, 7: 298-305(May 1965). (In Japanese).

Study was directed to finding an economical ion exchanger for removing Cs and Sr from radioactive waste. It was found that the ion-exchange capacity of aluminosilicate mineral is closely related to its crystal structure. Work was carried out on zeolitic tuff possessing the three dimensional clay structure. The zeolitic tuff consisted of moldenite crystal and noncrystallized volcanic glass, and these components had different selectivity and exchange capacity. Pretreatment of the zeolitic tuff modified its exchange capacity and selectivity, and this can be attributed to a change in its crystal structure and in some cases in its solid-state properties. Natural zeolitic tuff was successfully applied to the decontamination of radioactive solutions, and it was found that it has good selectivity for Cs ions. (auth)

651 AN ANION EXCHANGE PROCESS FOR AMERICIUM–CURIUM RECOVERY FROM PLUTONIUM PROCESS WASTE. M. H. Lloyd (Oak Ridge National Lab., Tenn.). Nucl. Sci. Eng., 17: 452-6(Nov. 1963).

An anion exchange process for the recovery of americium, curium, and rare earths contained in the waste effluent from plutonium processing was developed and tested on a laboratory scale. In the process the waste, which is a solution of americium, curium, aluminum, and fission products, in concentrated nitric acid, is concentrated by evaporation until a temperature of 140°C is reached. This re-

moves excess acid, and the proper feed concentration of 2.34 M $Al(NO_3)_3$ is obtained by dilution. The americium, curium, and rare earths are sorbed on Dowex 1-10X resin; aluminum is washed from the column with 8 M $LiNO_3$; and the americium, curium, and rare earths are eluted with 0.65 M HNO_3. In laboratory demonstrations of the process made with americium tracer and macro amounts of rare earths, americium losses were undetectable, aluminum decontamination factors were 250, and rare earth concentration in the product was as great as 8.5 gm/liter.

652 THE TRIALS OF PEAT APPLICATION TO SORPTIVE DECONTAMINATION OF WASTES, PRELIMINARY PURIFIED, FORMED AS THE RESULT OF THE SULFURIC ACID TREATMENT ON URANIUM ORES. Rafal Wlodarski (Instytut Badan Jadrowych, Warsaw). Nukleonika, 9: 496-8(1964). (In Polish)

653 ON THE RADIOACTIVE DECONTAMINATION WITH SEVERAL KINDS OF SAND AND CLAY. Goto, Hidehiro; Suzuki, Shin; Inoue, Yasushi; Kudo, Reiko (Tohoku Univ., Japan). Radioisotopes (Tokyo), 14: 225-32 (May 1965).

From the standpoint of radioactive decontamination, the adsorptive properties of several kinds of sand and clay, occurring in various parts of Japan, were investigated by batch methods using fission products as adsorbates. The removal was different from sample to sample and ranged between 40 and 100%. The relation between the removal of fission products and chemical composition was not clear. All the samples rich in calcite adsorbed effectively and resembled each other in their adsorptive properties. The removal of rare earths was high, while the removal of ruthenium was low. In the case of strontium and cesium, decontamination was greatly affected by the nature of materials but there was little difference in the pH dependence between strontium and cesium regardless of materials.

654 REMOVAL OF ^{137}Cs ON A NIIGATA MONTMORILLONITE CLAY BY COLUMN TECHNIQUE. Masao Takizawa (Inst. of Physical and Chemical Research, [Japan]), Kugao Koda, Shigetane Uehara, and Akio Hirokawa. Radioisotopes (Tokyo), 13: 455-61(Nov. 1964). (In Japanese)

Natural clay in rod of 1 mm dia. × 2 – 5 mm was prepared from a predominately Ca saturated montmorillonite (100%), which is a commercial bleaching earth from Ōmi, Niigata prefecture, with polyvinyl acetate (10%) as a binder. Portions of the natural clay in rod were converted to Na- and H-forms by treating with 1N solutions of the corresponding chloride. Through a column (2 cm dia.) of 20 g of natural, H-, or Na-clay in rod, the test solutions containing ^{137}Cs 1.0×10^{-2} μc/ml were percolated at a flow rate of 10 ml/min at room temperatures. The cesium capacities of the three clay columns were measured by the effluent volume (ml/g of clay) at a specific activity ratio (C/C_0) of 0.05. Na-Clay column was incomparatively superior in cesium capacity to natural and H-clay columns, and the pH value of the feed most suitable for ^{137}Cs removal on Na-clay was a range of 6 to 9, as was also indicated for ^{90}Sr removal. Comparative effects of competing various cations on the cesium capacity of Na-clay column are in the order: $K^+ > Na^+ > Ba^{2+} (> Al^{3+}$(pH 2.3)) $> Ca^{2+} > Sr^{2+}$ for neutral 1.0×10^{-1}N aqueous solutions of the corresponding chloride, because the effluent volume values at C/C_0 of 0.05 for the above systems in the presence of these cations were found to be 30, 170, 440, 770 (pH 2.2), 2,400 and 3,900 ml/g, respectively. Effects of increasing K, Na, and Ca concentrations on the cesium capacity of Na-clay column can be expressed as the linear relations between the effluent volume in ml/g at C/C_0 of 0.05, x, and log (the initial concentration of the corresponding chloride in N, y): $\log y = \log(1.07 \times 10^{-1}) - (2.58 \times 10^{-3})x$, for 0.01 – 0.1N KCl, neutral; $\log y = \log(1.06 \times 10^{-1})x$, for 0.05 – 0.1N NaCl, neutral; and $\log y = \log 1.32 - (4.66 \times 10^{-4})x$, for 0.1 – 0.5N $CaCl_2$, neutral.

655 REMOVAL OF FISSION PRODUCTS BY DIATOMACEOUS EARTH OF ISHIKAWA PREFECTURE. Itakura, Tadashi (Public Health Lab., Ishikawa, Japan). Radioisotopes (Tokyo), 15: 4-13(1966). (In Japanese).

Long-lived nuclides could be preferentially separated from fission products by use of refined diatomaceous earth. Raw materials from the Ishikawa Prefecture were treated with an acid mixture of HCl and H_2SO_4 and then calcined at 500°C. These refined materials are commercially employed as a filter-aid material for industrial use, but it was also found that the material behaves as a good adsorbent for radioactive nuclides in water. ^{137}Cs, ^{89}Sr, ^{90}Sr, ^{91}Y, ^{95}Zr, ^{144}Ce, ^{147}Pm, ^{103}Ru, and ^{106}Ru, as well as mixed fission products, were used for the examination. For various concentrations of nuclides, pH, and presence or absence of other salts, the adsorptivity of the earth material was examined. The refined diatomaceous earth (100 mg) was put into 50 ml of an aqueous solution containing a definite amount of radionuclide (less than 1×10^{-3} μCi/ml) and shaken for three minutes. After centrifuging the mixture for five minutes, the radioactivity of the supernatant was measured. The difference of activities between the original solution and the supernatant showed the amount of nuclide removed and the percentage removal was calculated. The nuclides treated may be classified in the following four groups from a point of the adsorptivity of the diatomaceous earth: cationic ^{137}Cs, ^{89}Sr, and ^{90}Sr (in neutral solutions, more than 98% ^{137}Cs could be removed and 90% of the ^{89}Sr or ^{90}Sr was removed when it was carrier free and 20% removed in the presence of other salts, but in alkaline medium the removal percentage increased up to 99%, especially in the presence of iron (III) ion); anionic ^{103}Ru and ^{106}Ru (as much as 95% of these were removed when iron (III) ion was present); hydrolytic ^{91}Y, ^{95}Zr, ^{144}Ce, and ^{147}Pm (these were removed well even when the pH of the solution was about 4 to 5); the fission products (the solution including the above three group nuclides) lost 99% of its activity after treatment with the diatomaceous earth in alkaline medium.

656 SORPTION OF STRONTIUM IONS ON SODIUM MOLECULAR SCREEN. G. V. Tsitsishvili and A. Yu. Krupennikova. Radiokhimiya, 5: 656-60(1963). (In Russian)

New data are presented on ion-exchange properties of synthetic A type zeolite (sodium aluminosilicate) studied with ^{90}Sr as a radioactive tracer and verified by chemical methods. Properties of long-life fission fragments $^{90}Sr-^{90}Y$ were investigated. (R.V.J.)

657 NATURAL INORGANIC DECONTAMINATING AGENTS FOR ^{134}Cs AND A MIXTURE OF ^{90}Sr AND ^{134}Cs IN WATER. Galea, V.; Birsan, E. T.; Uray, Ildiko (Medical-Pharmaceutical Inst., Cluj. Inst. of Hygiene and Labor Safety, Cluj). Rev. Chim. (Bucharest), 16: 456-8 (Sept. 1965).

The decontaminating effectiveness of crude gypsum, commercial superphosphate, and 4 varieties of bentonite for solutions of ^{134}Cs alone or mixed with ^{90}Sr and ^{90}Y was studied. The effects of varying the proportions of decontaminating agents and of varying the pH of the solution was also studied.

658 A COMPARISON OF Mg^{2+}, Ca^{2+}, AND Sr^{2+} AS EXCHANGE IONS FOR MINERALOGICAL STUDIES OF CLAYS. S. B. Weed and R. A. Leonard (North Carolina State Coll., Raleigh). Soil Sci. Soc. Am. Proc., 28: 58-62 (Jan.-Feb. 1964).

Strontium-, Ca^{2+}-, and Mg^{2+}-saturated clay minerals were studied with respect to x-ray diffraction, differential thermal analysis, cation exchange capacity, and glycerol adsorption. In general, 3-layer clay minerals saturated

with Sr^{2+} lost interlayer water at lower temperatures and at higher vapor pressures than did Mg^{2+}-clays; Ca^{2+}-clays were intermediate. A clay mineral-ion-saturation interaction was apparent. Calcium- and Sr^{2+}-saturation of clays yielded similar values for cation exchange capacity. Strontium-saturated 3-layer clays retained more glycerol than did clays saturated with Ca^{2+}. Expansion of vermiculite with glycerol and ethylene glycol was dependent on cation saturation and lattice charge. Strontium-saturated vermiculites expanded to near 16 Å more readily than did Mg^{2+}-clays, and Ca^{2+}-clays were intermediate. Vermiculites with high lattice charge expanded less readily than did vermiculites with relatively low lattice charge. Strontium can satisfactorily substitute for Ca^{2+} and Mg^{2+} in routine mineralogical studies of clays though special care may be needed to control relative humidity. Additional information regarding lattice charge of 3-layer clays may be obtained by using several divalent ion saturations. (auth)

659 SORPTION AND FIXATION OF MICROQUANTITIES OF CESIUM BY CLAY MINERALS: EFFECT OF SATURATING CATIONS. B. L. Sawhney (Connecticut Agricultural Experiment Station, New Haven). Soil Sci. Soc. Am. Proc., 28: 183-6(Mar.-Apr. 1964).

Sorption and fixation of microquantities of Cs by six minerals, as affected by the saturating cations, were studied. Micas and vermiculite sorbed more Cs in the presence of Ca and similar divalent cations than in the presence of K and similar monovalent cations. Montmorillonite and kaolinite, on the other hand, sorbed more Cs in the presence of K than in the presence of Ca. While micas and vermiculite fixed large proportion of the Cs sorbed, montmorillonite and kaolinite did not fix significant amounts of Cs against replacement by neutral salts. Thus, whereas Cs sorption by montmorillonite and kaolinite appeared to be largely electrostatic, the sorption and fixation of Cs by micas and vermiculite was governed by the size and hydration, rather than valence, of the competing cation. Micas particularly exhibited a large fixation of Cs. It is suggested that the fixation of Cs by micas occurs at the crystal edges where Cs may replace K and thus become a part of the stable mica structure. (auth)

660 DECONTAMINATION OF RADIOACTIVE WASTES WITH A TRANSPORTABLE ION EXCHANGE INSTALLATION. A. Dagen. Symp. Tech. Zagadnien Ochrony Promieniowaniem, Warsaw, 542-5(1962). (In German)

In work with radioisotopes, radioactive contaminated wastes occur which are collected in storage vats at the institutions. If the maximum permissible concentration is exceeded for surface waters, these must not be released into the drainage system, but must be processed. The operation of a portable ion exchange installation, which is installed at the individual institutions according to need, is described. (tr-auth)

661 THE EFFECT OF COLLAPSE-INDUCING CATIONS ON THE CESIUM SORPTION PROPERTIES OF HYDROBIOTITE. D. G. Jacobs (Oak Ridge National Lab., Tenn.). p.239-48 of "International Clay Conference, 1963." Pergamon Press, New York, 1963.

The Cs exchange properties of hydrobiotite were improved by collapsing the vermiculite lattices with ammonium ions or with one of the heavy alkali metal cations. Pretreatment with K increased the number of exchange sites at the edges of collapsed basal spacings, where Cs can be incorporated into the lattice by edge fixation. This increased the partial Cs capacity by a factor of 2.7. Addition of NH_4^+, Cs^+, Rb^+, or K^+ to the influent resulted in interlayer fixation of Cs as a result of collapse of the lattice to physically entrap the Cs sorbed at the basal surface exchange sites. (auth)

662 ION-EXCHANGE CONCENTRATION AND SEPARATE ISOLATION OF CESIUM, STRONTIUM, AND CERIUM FROM FRESH WATER. Senyavin, M. M.; Nikitina, N. G. pp 118-24 of Radiokhimicheskie Metody Opredeleniya Mikroelementov. Moscow-Leningrad, Izdatel'stvo Nauka, 1965. (In Russian).

After calculating the optimum concentration of the HCl eluent for each element and the volume of eluent of a given concentration corresponding to the peak on the elution curve (V_{max}), a procedure for separating a mixture of ^{137}Cs, ^{90}Sr, and ^{144}Ce from fresh water was developed. The KU-2 cation-exchange resin in a hydrogenated form was employed. The activity of the filtrates was measured with a β counter. Tap water to which the three isotopes had been added was passed through a column filled with KU-2, then the following amounts of acid were successively poured through the column: 100 ml 0.05 N HCl, 230 ml 0.5 N HCl, and 200 ml 2 N HCl, to remove ^{137}Cs, ^{90}Sr, and ^{144}Ce, respectively. The data indicate that the proposed ion-exchange method permits the separation and a sufficiently accurate determination of the content of cesium, strontium, and cerium in solutions with compositions similar to those of natural waters.

663 PROBLEMS IN USE OF AN ION EXCHANGER MADE FROM PEAT AND BENTONITE CLAY. G. Sachse and R. Winkler (Zentralinstitut für Kernphysik, Rossendorf, Ger.). Symp. Tech. Zagadnien Ochrony Promieniowaniem, Warsaw, 560-79(1962). (In German)

Natural ion exchangers have great possibilities for the purification of radioactive waste solutions. Organic exchangers such as peat, lignite, or other humus-containing substances, and inorganic exchangers, such as silicate clays, are suitable. In the use of exchangers containing humic acid, the most essential disadvantage is the solubility of the alkali humates arising in the exchange process. Clay exchangers, which are partly cesium selective, produce great difficulties because of their high swelling properties. A combination of peat and bentonite clay eliminates these disadvantages to a great extent. The combined exchanger has available sufficiently stable grain structure and reduces the formation of Cs humate when Cs-selective clays are used. The simple preparation of the exchangers and their properties are described. The stability, swelling behavior, and capacity with various clay components are shown. The exchange properties are characterized by examples of ^{137}Cs, ^{90}Sr (or ^{89}Sr), ^{144}Ce, and fission product mixtures. (tr-auth)

664 PURIFICATION OF RADIOACTIVE WASTES WITH NATURAL ORGANIC ION EXCHANGERS. R. Winkler (Zentralinstitut für Kernphysik, Rossendorf, Ger.). Symp. Tech. Zagadnien Ochrony Promieniowaniem, Warsaw, 546-59(1962). (In German)

Natural ion exchangers are often used for the decontamination of radioactive wastes as they can be discarded after one use because of their low cost. The organic materials of this group, such as lignite, peat, and wood, can also be burned so that the adsorbed radioactive impurities can be concentrated in the relatively small ash volume. The use

of natural organic ion exchangers is made difficult by the relatively low alkali resistance and by the weakly acid character of the exchange-active groups. Problems are discussed on the basis of some studies on humic acid and peats. It was proved that natural organic exchangers can be used when the water to be treated has no extreme pH value, contains low levels of inactive salts, and contains the impurities to be removed in cationic and colloidal form. (tr-auth)

665 RESEARCH ON PRODUCTION AND EXCHANGEABILITY OF LIGNIN SULFONIC ACID CONDENSATION PRODUCTS FOR PURIFICATION OF RADIOACTIVELY CONTAMINATED WASTES. S. Riedel (Zentralinstitut für Kernphysik, Rossendorf, Ger.). Symp. Tech. Zagadnien Ochrony Promieniowaniem, Warsaw, 500-16(1962). (In *German)*

Condensation products of lignin sulfonic acids can be refined from the sulfite waste liquor of the cellulose industry by heating under pressure. These lignin sulfonic acid condensation products can be used as economic cation exchangers in the purification of radioactive wastes because of their content of groups with ion exchanger properties. The dependence of the exchange properties on the type of waste liquor and the preparation conditions are described. The uptake of individual fission products from wastes and laboratory investigations on the decontamination of reactor wastes are briefly considered. (tr-auth)

666 SORBENT SCRAP MATERIALS FOR PURIFICATION OF ACTIVE WASTE. Z. Smal (Inst. of Nuclear Research, Warsaw). Symp. Tech. Zagadnien Ochrony Promieniowaniem, Warsaw, 517-41(1962). (In Polish)

The use of scrap materials in the form of blast-furnace slag for the purification of radioactive wastes is described. The results obtained on the absorption of ^{90}Sr, ^{137}Cs, ^{95}Zr, ^{144}Ce, and ^{106}Ru from waste solutions were analyzed. The effects of parameters such as temperature and flow velocity were taken into consideration. The possibility of the appearance of capillary-active materials in the wastes purified by this method was considered. (tr-auth)

667 A STUDY OF SORPTION OF CESIUM BY MONTMORILLONITE [Thesis]. Metzger, Charles Frederick. Urbana, Ill., Univ. of Illinois, 1965. 68p.

In the study of sorption of cesium by montmorillonites, four samples with interlayer cation percentages of Ca_{100}, $Ca_{56}Na_{44}$, $Na_{56}Ca_{44}$, Na_{100} were used in order to determine the effect of the interlayer cation on the sorption of cesium. The sorption studies were carried on by contacting 0.25 grams of each sample with 100 ml of cesium nitrate plus cesium-137 as contacting solution. Master and sample aliquots were then compared at various time intervals by counting the cesium-137 present, to determine the amount of sorption of cesium. One-dimensional Fourier syntheses and electron density projections were made of the untreated and cesium-treated samples, and the interlayer positions compared to see if cesium had replaced the exchangeable interlayer cations of the montmorillonite samples. Results of the sorption analyses indicate the exchangeable cations of the montmorillonites have no appreciable effect on the amount of cesium sorbed. Studies of one sample that had extremely low sorption and a high K_2O content led to the conclusion that the K_2O is present on the edges of the clay platelets and results in a blocking of the cesium causing a decrease in sorption. In comparing the one-dimensional Fourier syntheses and electron density projections of the untreated and cesium-treated montmorillonites, it was seen that no cesium had entered the interlayer position of the clay minerals. Therefore, it is concluded that the cesium fixation is of the edge type rather than interlayer type fixation. A "core-rind" model of the clay platelets is proposed to explain the sorption properties of the montmorillonites. The inner "core" is thought to be a well-crystallized montmorillonite which shows good diffraction and gives data that produced a Fourier projection of a normal well-crystallized montmorillonite, whereas the "rind" is the less ordered portion surrounding the "core," which produces few diffraction data and in which sorption of cesium takes place. This "core-rind" model also accounts for the decrease in sorption when potassium is present. Potassium on the edges of the montmorillonite platelets would cause a collapse of this area, making it difficult for the larger cesium atom to enter, thus causing lower sorption. (Dissertation Abstr.)

668 STUDY OF THE USE OF MINERAL COAL FOR PURIFYING RADIOACTIVE WASTE WATER. J. Makowski (Inst. for Hydrological Construction, Gdansk, Poland). Symp. Tech. Zagadnien Ochrony Promieniowaniem, Warsaw, 489-99(1962). (In Polish)

The results of research on the application of Polish mineral coal to the purification of waters and radioactive wastes are discussed. The study was carried out by means of filtration of the radioactive water solutions in sorption columns filled with various types of mineral coal and on prepared solutions, sea water, river water, tap water, and municipal sewage. The radioisotopes ^{90}Sr, ^{137}Ba, ^{137}Cs, and ^{144}Ce were considered. The most suitable type of mineral coal, the optimum pH value for the solution, the minimum length of the sorption column, and the decontamination coefficients obtainable with this method were reported. (tr-auth)

Evaporation

See also the following references:

12, 19, 24, 32, 34, 64, 80, 81, 82, 83, 85, 87, 89, 92, 98, 99, 103, 104, 107, 108, 121, 135, 137, 141, 146, 148, 150, 162, 163, 194, 237, 245, 517, 594, 811

669 (AD-404279) AN EVAPORATOR-TYPE RADIOACTIVE LIQUID WASTE CONCENTRATOR WITH DISPOSABLE VESSEL. Eliezer Rubin, Arye Reisser, and Algird Karalis (Radiation Applications Inc., Long Island City, N. Y.). Mar. 5, 1963. Contract DA18-108-CML-7196. 69p.

Detailed design specifications and operating instructions are presented for a prototype evaporation unit with disposable vessel for concentrating radioactive liquid wastes. The design is based on the use of drums as evaporation vessels so adapted that the feed inlet, vapor outlet, steam connections, and liquid level control are detachable permitting the drums to be detached from the rest of the equipment and shipped for land disposal. The complete evaporation unit is portable, automatically controlled, and has operating requirements for a supply of 80 to 150 psi steam for the evaporator, cooling water for the condenser, and electricity for operation of the pumps and controls.

670 (BNL-10047) PHOSPHATE GLASS PROCESS: EVAPORATOR STUDIES WITH SIMULATED PUREX WASTES. Strickland, G.; Tuthill, E. J.; Drager, R. F. (Brookhaven National Lab., Upton, N. Y.). [1966]. Con-

tract AT(30-2)-Gen-16. 24p. (CONF-660208-11). Dep. mn. CFSTI $1.00 cy, $0.50 mn.

From Symposium on Solidification and Long-Term Storage of Highly Radioactive Wastes, Richland, Wash.

Preliminary studies were made on the continuous evaporation and denitration of simulated Purex wastes using a tube-bundle evaporator equipped with a turbine agitator. Operating experience and performance data were obtained with prototype design-basis feed at 150°C. Decreases in performance were generally related to the initial heating conditions rather than to the total time of operation. In the concentration of Purex-2 feed at temperatures of 123–145°C, the performance of the evaporator decreased at each higher temperature because of the increased volume reduction and solution viscosity. Recommendations include several design changes and continued testing in the vicinity of 130°C.

671 (NDL-TM-20) CONCENTRATION OF RADIOACTIVELY-CONTAMINATED POTASSIUM TETRABORATE TETRAHYDRATE (PTB) WASTE SOLUTIONS. Goff, David L.; Bloore, Ernest W.; Crisco, Carl (Army Nuclear Defense Lab., Edgewood Arsenal, Md.). May 1965. 69p.

The ability of the Radioactive Liquid Waste Treatment Facility was demonstrated in successfully processing radioactively contaminated aqueous solutions of potassium tetraborate (PTB). Feed solutions contained an average of 50,000 ppm PTB and 10^{-5} to 10^{-2} μC/ml of gross beta activity. Evaporator bottoms approached theoretical saturation while the effluent remained essentially free of chemical and radiological contaminants. (auth)

672 (NP-13289) TESTS FOR THE EUROCHEMIC "HOT WASTE" EVAPORATOR. A. Redon and F. Marcus (European Company for the Chemical Processing of Irradiated Fuels, Mol, Belgium). Mar. 1963. 81p. (ETR-140)

Details are given of the experimental program carried out by A. B. Atomenergi to supply data required for the design of an evaporator for treating "hot wastes" from the Eurochemic plant. The investigations showed that a decontamination factor of between 10^7 and 10^8 (based on uranium) can be attained between the evaporator solution and the condensate. The test results give information about design and operation which will form an essential part of the evaporator specifications. (auth)

673 (NP-15367) CHEMICAL ENGINEERING STUDIES ON THE PROCESSING OF RADIOACTIVE LIQUID WASTES BY EVAPORATION. Progress Report, January 1–June 30, 1964. Yamamoto, Yutaka; Miyauchi, T.; Kunii, D.; Kiyose, R. (Tokyo Univ. (Japan). Faculty of Engineering). Contract 250/RB. 24p. Dep. mn.

The operational characteristics of fluidized-bed spray evaporator for processing radioactive liquid wastes were investigated. Although the principal idea of the process is almost the same as the double-effect evaporator, by using the spray evaporator for final evaporation and solidification, it becomes possible to get the product directly from the evaporator as solid particles and thus to minimize the cost of disposal of radioactive liquid wastes. (P.C.H.)

674 (NP-15371) CHEMICAL ENGINEERING STUDIES ON THE PROCESSING OF RADIOACTIVE LIQUID WASTES BY EVAPORATION. Final Report, January 1, 1964–March 31, 1965. Yamamoto, Yutaka; Miyauchi, T.; Kunii, D.; Kiyose, R. (Tokyo Univ. (Japan). Faculty of Engineering). Contract 250/RB. 27p. Dep. mn.

In studies on the use of the fluidized-bed spray-type evaporator for economical processing of radioactive liquid wastes, cold experiments were conducted to obtain fundamental data on evaporation and solidification processes of solutions sprayed and heated in superheated steam. Heat transfer rate in the preheating stage is quite rapid as the process is considered to be condensation heat transfer. In the vaporization process, sensible heat transfer through the film around the liquid drops is considered dominant and the process is somewhat slower than the preheating stage. Engineering data for the fluidized-bed are correlated. It was concluded that the selection of the position of injection point of the feed liquid is not a critical problem in this type of evaporator. Liquid drop size has, however, an important effect on the operation characteristics of the bed. From a bench scale experiment it was concluded that the minimum recycling steam is necessary to maintain the good fluidizing condition. (P.C.H.)

675 (NP-15662) THE EVAPORATION OF RADIOACTIVE WASTES BY SOLAR ENERGY. Report No. 1. Progress Report, December 1, 1963–August 31, 1964. Amarantos, A.; Hatzikakidis, A. (Democritus Nuclear Research Center, Athens (Greece)). Contract 224/RB. 18p. Dep. mn.

Field experiments with a flat plate collector (black body panels) as the heat source for a solar evaporator are described. From the field experiments conclusions were made in connection with the calculation and design of the evaporator. Reasons are given for choosing to heat the evaporator by heat exchangers instead of passing the wastes through the solar collectors. Calculations on the heat transfer were based on maximum values of received solar radiation. Two separate evaporating systems are described, and power and evaporating capacity for each were determined.

676 (NP-15874) THE EVAPORATION OF RADIOACTIVE WASTES BY SOLAR ENERGY. Report No. 2 (Final). Amarantos, S.; Hatzikakidis, A. ("Democritus" Nuclear Research Center, Athens (Greece)). Dec. 1965. Contract 224/RB. 33p. Dep. mn.

The feasibility of using solar energy in the evaporation of radioactive wastes was studied. A semi-pilot plant device (solar evaporator) was designed and constructed based on the existing information on the various applications of solar energy. The solar evaporator was installed in the open air and operated for almost three months (September to November 1965). The results showed that solar evaporation of radioactive liquids in open air is possible without the danger of contaminating the environment. Also, the following evaporating capacities were received: approximately 20 liters per eight hours (8.00–16.00) when the maximum temperature of the evaporated liquid was approximately 50°C.

677 (RFP-534) AQUEOUS WASTE EVAPORATION. Butler, F. E.; Vance, J. N. (Rocky Flats Div., Dow Chemical Co., Denver, Colo.). Aug. 6, 1965. Contract AT(29-1)-1106. 23p. Dep. mn; CFSTI $2.00 cy, $0.50 mn.

A study of overall heat transfer coefficients in a short vertical tube evaporator for the concentration of salted aqueous wastes was conducted. The variables of solution concentrations, boiling temperature, temperature differential, and their effects on overall heat transfer coefficients were investigated. Decontamination factors were determined in relation to vapor velocities in the deentrainment section of the evaporator. (auth)

678 (UJV-1109/64) OPYT PO EKSPLUATATSII USTANOVOK PO OBESVREZHIVANIYU RADIOAKTIVNYKH OTKHODOV V IYAI V RZHEZHI. (Operation of Waste Decontamination Installations Constructed by the Institute of Nuclear Research in Rzhezhi, Czechoslovakia). I. Zaruba and I. Napravnik (Ceskoslovenska Akademie Ved. Ustav Jaderneho Vyzkumu, Rez). [nd]. 7p.

Descriptions are given of two installations for liquid waste decontamination by evaporation that have a capacity of 125 to 140 kg/hr at 80°C and yield a 100-fold volume reduction. The mean activity of treated material was 10^{-6} C/l, mean dry residue 1.5 g/l, with pH from 2.5 to 4.5. A layer and improved device built in 1961 increased capacity to 8 m^3/hr. Since 1962 the installation has processed 570 m^3 of waste with activities from 10^{-4} to 10^{-8} C/l and dry residue 0.5 to 7 g/l. (R.V.J.)

679 (UJV-1110/64) USTANOVKI PO OBESVREZHIVANIYU PADIOAKTIVNYKH OTKHODOV. (Radioactive Waste Decontamination Installations). V. Vesely, A. Enichek, and I. Yansa (Československá Akademie Věd. Ustav Jaderného Výzkumu, Řež). 1964. 14p. Dep.(mn).

A boiler type evaporator and a small film evaporator used for decontamination of radioactive liquid waste are described. (R.V.J.)

680 EVAPORATORS FOR THE DECONTAMINATION OF EFFLUENT WATER. H. Mende. Atomwirtschaft, 9: 498-504(Oct. 1964). (In German)

The decontamination of radioactive polluted water by distillation is discussed. Special consideration is given to the evaporator used. The purchase price and operating costs of the evaporation plant must stand in a favorable relationship if the decontamination is to be economic. The required heating surface is an important parameter influencing these costs. In this connection the type of evaporator has an essential influence on the price of the distillate. The individual types as well as the energy consumption of single and multi-stage evaporators are compared. A cost comparison showed that in the case of continuous operation a directly heated evaporator shows the highest distillate price, an evaporator with vapor jet compression of the steam a considerably lower price, and a forced circulation evaporator with a electrically driven compressor by far the lowest price. This applies for equal capacities, equal heating costs, and favorable operating conditions.

681 DECONTAMINATION OF RADIOACTIVE EFFLUENTS. Brodsky, Michel; Moneton, Jean-Louis; Pagny, Pierre. Energ. Nucl., 8: 9-22(Jan.-Feb. 1966). (In French).

Following a description of methods used for the decontamination of radioactive effluents and the operation of the evaporation installations, the technical and financial results obtained on these installations are compared according to their treatment capacity. Thus, capital investments and the cost per unit for an effluent treatment in a single-acting evaporator (with or without mechanical recompression of stream) are included in the financial study of the chemical treatment. With the results obtained from this study, a choice can be made, for a future project, of the type of treatment to be used according to the characteristics of the effluents and certain local requirements.

682 RADIOACTIVE WASTE WATER DISTILLATION DATA AND COSTS. R. Rupert Kountz (Pennsylvania State Univ., University Park). Purdue Univ., Eng. Bull., Ext. Ser., No. 112, 409-20(Mar. 1963).

The performance of an evaporator for the purification of radioactive waste waters of varying concentrations of solids, varying acidity, and varying radioactivity was tested. Results indicated that distillation of radioactive waste water at low pressures and low temperatures produces water that is safe for general use. The estimated cost is between $0.20 and $0.50/gal. (C.H.)

683 A PLANT FOR THE DISPOSAL OF RADIOACTIVE WASTE WATER. Vesely, Vladimir (Inst. of Nuclear Research, Czechoslovak Academy of Sciences, Rez); Napravnik, Jiri; Jansa, Jindrich. Jaderna Energie, 11: 213-18(June 1965). (In Czech.).

When work with radioisotopes was begun at the Nuclear Research Institute in Rez, the storage tanks designed originally only for reactor waste water proved inadequate. A waste water disposal plant was built, with a boiler and a film evaporator. The disposal plant is described, and experience with its operation over a period of several years is reviewed. (CFSTI)

684 STUDY OF THE DYNAMIC BEHAVIOR AND THE PROPERTIES CONCERNING AUTOMATIC CONTROL OF A FOUR-STAGE EVAPORATOR FOR TREATING CONTAMINATED WATERS. Mikoleit, W. (VEB Atomkraftwerk, Berlin). Kernenergie, 8: 597-601(Oct. 1965). (In German).

A multistage evaporator for treating contaminated water was investigated by using an analog computer with regard to its dynamic behavior and its properties in an automatic control system, when electronic control elements are applied. The system of equations and some results of the calculations are given.

685 ECONOMIC ASSESSMENT OF VARIOUS EVAPORATOR TYPES. H. Mende (Pintsch Bamag AG, Butzbach, Ger.). Kerntechnik, 6: 159-63(Apr. 1964). (In German)

An evaporator heated by steam or hot vapors that are compressed by a steam injector was compared with an evaporator of the forced recirculation type with an electrically driven hot vapor compressor. The cost of utilities and the operating periods were found to highly affect the cost of the distillate produced. By adding the annual capital expenditure to the operating cost the price of the distillate increased to a very different extent for each of the three evaporator types. Although its capital cost is the highest the evaporator with forced recirculation was found to be the most economical in the long run. Diagrams show the influence of the variables on capital cost and operating cost. The optimum value of the overall cost of the evaporator installation can be determined only for each case by taking into account all operating conditions. (auth)

686 DECONTAMINATION OF RADIOACTIVE WASTES OF THE KARLSRUHE NUCLEAR RESEARCH CENTER FROM 1963 TO 1965. Krause, H.; Nentwich, O. (Kernforschungszentrum, Karlsruhe, Ger.). Kerntechnik, 8: 105-10(Mar. 1966). (In German).

A vapor compression evaporator and an installation for the fixation of radioactive wastes in bitumens were constructed. The amount and activity of the waste processed is given. The procedures followed in the decontamination are outlined and the decontamination factors obtained in each step are discussed.

687 THE INSTALLATION FOR THE EVAPORATION OF RADIOACTIVE LIQUID WASTES. P. Baertschi and H. R. von Gunten (EIR, Wurenlingen, Ger.). Neue Tech., 5: 531-5(Sept. 1963). (In German)

The evaporating system consists of the evaporator itself and the steam-heating circuit. These are described. A series of inactive experiments was performed on the evaporator. The decontamination factors obtainable were calculated, and the results were in agreement with the laboratory-scale research. (J.S.R.)

688 STUDIES ON TREATMENT OF RADIOACTIVE LIQUID WASTES BY FALLING FILM EVAPORATOR. Tokuro Mizushina, Takeichiro Takamatsu, Nobuo Mitsuishi, Tsutomu Fukumoto, and Takashi Yamada (Kyoto Univ.). Nippon Genshiryoku Gakkaishi, 5: 933-8(Nov. 1963). (In Japanese)

A falling film evaporator is proposed for treatment of radioactive waste solutions. A test apparatus was manufactured, and its decontamination capacity was studied using water and CMC aqueous solutions as Newtonian and non-Newtonian fluids, respectively, and LiCl as a tracer. When the liquid flow rates are 200 to 300 l/hr, and vapor mass velocities 100 to 1,000 kg/m^{2}hr, the decontamination factors are about 10^5 for the Newtonian fluid, which coincides well with the results previously obtained with a standard type evaporator, and about 10^4 for the non-Newtonian fluids. The mean retention time for the Newtonian fluid is about 0.26 sec, and that for the non-Newtonian fluid about 3.2 sec, when the liquid flow rate is about 200 l/hr.

689 AN INEXPENSIVE METHOD OF REDUCING THE VOLUME LOW LEVEL RADIOACTIVE SOLUTIONS FOR WASTE DISPOSAL. E. M. Romney, K. A. Flygare, and L. B. Silverman (Univ. of California, Los Angeles). Soil Sci. Soc. Am. Proc., 28: 295-6(Mar.-Apr. 1964).

An inexpensive, practical, and safe nonboiling method is described for evaporating radioactive nutrient solutions to small volume for subsequent waste disposal. The device is an electric immersion heater designed to float at the surface of the liquid waste. Economy, speed, and limitations of the method are discussed. (H.M.G.)

Filtration

See also the following references:

27, 34, 36, 74, 87, 102, 103, 121, 150, 162, 511, 587, 737, 866

690 (AERE-R-3880) FILTRATION OF HARWELL EFFLUENT. Clarke, J. H.; Parsons, C. P.; Wright, T. D. (Atomic Energy Research Establishment, Harwell (England)). Nov. 1961. 26p. Dep. mn.

A two-stage system for the filtration of Harwell effluent was studied on a pilot plant scale. Experiments showed that the waste arisings can be as low as 0.8% of the throughput for effluent containing 3,000 ppM suspended solids and less than 0.1% of the throughput at 50 to 80 ppM.

691 HANDLING AND FILTRATION OF RADIOACTIVE SLUDGES. Pearce, K. W.; Smyth, M. J. (Atomic Energy Research Establishment, Harwell, Eng.). Filtr. Sep., 90-4(Mar.-Apr. 1965).

Two processes for treatment of radioactive sludges were developed: pressure filtration at 30 to 35 psig and freezing-thawing followed by vacuum or centrifugal filtration.

692 ON THE APPLICATION OF FILTER AIDS FOR THE FILTRATION OF SLUDGES FROM THE DECONTAMINATION OF RADIOACTIVE EFFLUENTS. Levi, H. W. (Hahn-Meitner-Institut fuer Kernforschung, Berlin). Kerntechnik, 7: 485-90(Nov. 1965). (In German).

Some of the factors affecting the decontamination of radioactive sludges by filtration using filter aids were investigated. The distribution of the filter aid masses on predepositions and admixtures, aging of the precipitation, and addition of the filter aid before precipitation were the factors studied. The filter aids investigated were Fibra Flo, Filtercel, Celite 545, Celite 503, Hyflo-supercel, Sinofil, peat, sawdust, and vermiculite. Iron phosphate and calcium carbonate were the precipitating agents. Of the filter aids studied, Celite 545 and Celite 503 at a concentration of 4 g/l were optimum for the filtration of iron phosphate sludges. With 8 g/l sawdust, the filtration efficiency is significantly worse, to which must be compared however the lower cost and the possibility of volume reduction by burning. Fibra Flo offers no advantage because of its high price. In the filtration of $CaCO_3$, the addition of a filter aid is necessary only for the stabilization of the filter cooking. Filtercel in a concentration of 2 g/l is suitable for this. Predepositions of a part of the filter aid offers no advantages. Aging of the iron phosphate sludge and addition of the filter aid before precipitation do not affect the filtration characteristics. The decontamination factor for Sr was significantly modified by the choice of the filter aid neither in the case of iron phosphate nor calcium carbonate precipitation.

Fixation in Ceramic and Other Solid Media

See also the following references:

7, 12, 16, 19, 20, 24, 26, 35, 39, 45, 49, 55, 56, 57, 61, 66, 67, 75, 81, 82, 85, 87, 106, 121, 130, 146, 148, 154, 171, 185, 186, 209, 213, 217, 226, 233, 235, 245, 248, 249, 387, 686, 752, 880, 1009

693 (A/CONF.28/P/774) ASPHALT CONDITIONING AND UNDERGROUND STORAGE OF CONCENTRATES OF MEDIUM ACTIVITY. P. Dejonghe, L. Baetsle, N. Van de Voorde, W. Maes, P. Staner, J. Pyck, and J. Souffriau (Centre d'Etude l'Energie Nucleaire, Brussels). May 1964. 16p.

Two studies were carried out on the problem of storage of medium-activity wastes. Insolubilization of Wastes. It was found that concentrates can be insolubilized by violent stirring into hot asphalt at 200°C. Elution tests and pilot plant tests were made. An industrial plant for processing of sludges is described. Migration of Radioelements in Ground. Tracer studies were made of migration in sandy soils and ground-water table. (D.L.C.)

694 (AEET/HP/TH-20) TEMPERATURE DISTRIBUTION IN RADIOACTIVE SOLIDS. PART II. GAMMA-ACTIVE SPHERES. Sundaram, V. K.; Gopinath, D. V.; Ganguly, A. K. (Atomic Energy Establishment, Trombay (India). Health Physics Div.). 1964. 22p. Dep. mn.

One of the ultimate methods proposed for high level radioactive waste disposal is to contain the activity in ceramic bodies and bury them underground or in deep sea. The limit of the activity that can be incorporated in such bodies is decided by the temperature rise in the body due to the absorption of nuclear radiation. The temperature distribution in a solid sphere containing long-lived gamma emitters and buried in a medium having the same thermal properties is discussed. Analytical expressions are given for heat source distribution and for calculations of temperature distribution in and out of a solid sphere containing gamma activity. Numerical values obtained with energies in the range of 0.5 to 3 Mev and in spheres of radii 10, 20, and 30 cm are presented. The results are compared with those of the beta case. (auth)

695 (BNWL-157) TRANSIENT THERMAL BEHAVIOR IN FIXATION OF RADIOACTIVE MATERIALS.

Jansen, G. Jr.; DeMier, W. V. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). Jan. 1966. Contract AT(45-1)-1830. 123p. Dep. mn. CFSTI $4.00 cy, $0.75 mn.

Dimensionless correlations are given for the thermal history of cylindrical, infinite-slab, and annular heat-generating radioactive material containers after fixation at elevated temperatures. The maximum temperatures attained are presented as functions of heat generation rate, initial temperature, container geometry, thermal conductivity, and environmental heat transfer coefficients. The effect of heat losses through the ends of a solid cylindrical container is also presented for steady state conditions. Several examples are given as an aid to calculations. Computer programs for more detailed estimation of thermal histories in slabs, cylinders, and annuli are appended.

696 (BNWL-170) RESEARCH AND DEVELOPMENT ACTIVITIES. FIXATION OF RADIOACTIVE RESIDUES. Quarterly Progress Report, July-September 1965. Platt, A. M. (ed.) (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). Oct. 1965. Contract AT(45-1)-1830. 40p. Dep. mn. CFSTI $2.00 cy, $0.50 mn.

The calcined-waste containers produced during the Hot Cell Pilot Plant experiments in 1962 and 1963 have generated a maximum pressure of 3 psig. Design verification tests with the prototype spray calciner showed satisfactory performance when producing high melting point powders. Later tests on other feeds showed poor atomization and cake buildup in the calciner. Design of the prototype equipment is complete. Procurement and fabrication of prototype equipment is done except for a few cleanup items and final field-fitting of jumpers. Design verification tests on the pot storage station indicated satisfactory calorimeter performance. A GE-412 computer was successfully used in design verification tests for data collection, calculation, and correlations. Studies continued on fixation of sulfates from acid condensates generated in a glass-making process. Lead sulfate-borax melts produced by a continuous feed melter at 900°C appear to be quite fluid, but exhibited a high (22%) evolution of sulfate. Studies were initiated on incorporation of the neutralized condensate into asphalt. Release rates of up to 2%/hr for ^{137}Cs from a simulated phosphate ceramic waste material were observed from exposures to temperatures up to 1200°C. Release rates of other isotopes which appear to be a thousand-fold lower than for cesium established volatilization as the predominant mode of cesium release. A technique for varying the phosphate content of the simulated ceramic wastes was developed.

697 (BNWL-SA-453) VOLATILIZATION OF FISSION PRODUCTS FROM HIGH LEVEL CERAMIC WASTES. Albrethsen, A. E. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). Feb. 8, 1966. Contract AT (45-1)-1830. 17p. (CONF-660208-5). Dep. mn. CFSTI $1.00 cy, $0.50 mn.

From Symposium on Solidification and Long-Term Storage of Highly Radioactive Wastes, Richland, Wash.

Volatilization was established as the predominant mechanism of fission product release from simulated high level solidified waste of differing phosphate content when exposed to high temperatures. Waste representative of fuel ranging from 20,000 to 100,000 Mwd/t exposure was used. Cesium-137 was found to be the isotope of greatest radiological concern from the standpoint of airborne release during a shipping accident fire. Fractional release rates at 1100°C for cerium, ruthenium, and strontium were at the lower detection limits of the experiment (approximately 10^{-3} percent/hour) while average cesium release rates of about ½ percent/hour were observed under the same exposure conditions.

698 (BNWL-SA-467) THE WASTE SOLIDIFICATION ENGINEERING PROTOTYPES TECHNICAL PROGRAM. Cooley, C. R.; Platt, A. M. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). Feb. 1, 1966. Contract AT(45-1)-1830. 25p. (CONF-660208-6). Dep. mn. CFSTI $1.00 cy, $0.50 mn.

From Symposium on Solidification and Long-Term Storage of Highly Radioactive Wastes, Richland, Wash.

The objectives and future plans for the Waste Solidification Engineering Prototypes are discussed. The emphasis on specific safeguard, technical, and economic evaluations and the program steps to achieve demonstration of solidification methods are outlined for an Oak Ridge National Laboratory pot calcination and rising level glass process, a Pacific Northwest Laboratory spray solidification process, and a Brookhaven National Laboratory phosphate glass process. Plans for the first portion of the program are reviewed.

699 (BNWL-SA-514) SURVEY OF METHODS FOR THE TREATMENT AND DISPOSAL OF GLASS MELTER CONDENSATES CONTAINING SULFURIC ACID. Mercer, B. W.; Touhill, C. J.; Coleman, L. F.; Svoboda, C. J. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). Feb. 2, 1966. Contract AT(45-1)-1830. 14p. (CONF-660208-7). Dep. mn. CFSTI $1.00 cy, $0.50 mn.

From Symposium on Solidification and Long-Term Storage of Highly Radioactive Wastes, Richland, Wash.

Methods for the treatment of sulfuric acid condensate from the phosphate glass melter were investigated. Projected fission product concentrations in the condensate were estimated to be more than 100 C/gal (predominantly ruthenium) when processing wastes from 20,000 Mwd/t fuel. Two general approaches were taken for the treatment and disposal of the condensate: conversion of the condensate or the sulfate contained therein to solids which retain all or a portion of the fission products; and separation of the sulfate from the fission products which would be recycled to the glass process.

700 (CEA-2331) ENROBAGE PAR LE BITUME DES BOUES RADIOACTIVES DE LA STATION DE TRAITEMENT DES EFFLUENTS DU DENTRE DE MARCOULE. Mise au point des états d'avancement 1, 2, 3 et 4. (Bitumen Coating of the Radioactive Sludges from the Effluent Treatment Plant at the Marcoule Center. Review of the Progress Reports 1, 2, 3, and 4). J. Rodier, G. Lefillatre, and J. Scheidhauer (France. Commissariat à l'Energie Atomique. Centre de Production de Plutonium de Marcoule, Chusclan). 1963. 135p.

Evaporation, resin concentration, and coprecipitation techniques for the processing of highly active liquid wastes give rise to inorganic sludges with a high water concentration. The preliminary studies undertaken on the bitumen coating of the radioactive sludges from the effluent treatment plant at Marcoule are reviewed. The properties of bitumen and of the radioactive sludge are first described. The principle of the bitumen coating of sludge is discussed. A one-stage and a two-stage process for the coating and the quality of the coating resulting from each process are described. The activity of the liquid wastes and the atmospheric contamination resulting from the bitumen coating are determined for both processes. The problems of bitumen coating are discussed.

701 (CEA-tr-A-1413) PROCEDE DE SOLIDIFICATION, DE DEFORMATION ET DE PROTECTION CONTRE LES RAYONNEMENTS DES DECHETS RADIOACTIFS. (Process of Solidification, of Deformation, and of Protection Against Radiations of Radioactive Waste). H. Wiedmann. Translated into French from German Patent 1,071,246. [nd]. 6p.

A procedure that permits the transformation of radioactive wastes into bodies giving no difficulties in transport or storage is described. The solid, liquid, or sludge radioactive wastes are incorporated into bituminous materials used as binders. The bituminous materials are used in the form of emulsions. Substances that ensure improved solidification and/or that shield against the radiation are added to the radioactive wastes. (J.S.R.)

702 (HW-81829) RESEARCH AND DEVELOPMENT ACTIVITIES [ON] SOLIDIFICATION OF RADIOACTIVE WASTES. Progress Report. A. M. Platt, ed. (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Apr. 15, 1964. Contract AT(45-1)-1350. 26p.

Research and development activities on the solidification of radioactive residues are reported. Work is reported on the following activities: small-scale spray and pot calcination studies with actual high-level radioactive wastes, large-scale spray solidification studies with non-radioactive simulated wastes, and design and fabrication of an engineering-scale prototype for demonstration of three promising processes (pot calcination, spray solidification, and phosphate glass solidification) with actual high-level radioactive wastes. (N.W.R.)

703 (HW-81862) RESEARCH AND DEVELOPMENT ACTIVITIES. FIXATION OF RADIOACTIVE RESIDUES. Quarterly Progress Report, January–March 1964. A. M. Platt, ed. (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Apr. 1964. Contract AT(45-1)-1350. 44p.

Glass samples were prepared from a simulated Purex waste spiked with massive quantities of radiocerium. Laboratory studies on glass formulations showed variations in drip temperature as a function of fission product content and sulfate-phosphate content. A marked phosphate loss occurred with some of the mixtures during calcination and melting. Capacity studies on a mockup of the 14-in. spray calciner have verified the desired capacity of 10 gph. Other tests have indicated unusually high water capacity with an ultrasonic atomizing nozzle. Computer studies show that the maximum size of a container for radioactive material may be limited by transient conditions during fixation-stabilization rather than equilibrium storage conditions. Pilot plant pulse column studies using BAMBP to extract cesium from Purex alkaline waste were completed with satisfactory results. Laboratory studies were conducted to evaluate several methods of eliminating or minimizing ammonium ion in the scrub effluent from zeolites loaded with cesium from Purex supernatant waste. The temperatures that would be experienced in solidifying wastes in large underground tanks were predicted as a function of waste and soil thermal conductivities and waste heat generation rates. Electrodialysis and electrodeionization techniques for removal of ruthenium from intermediate-level wastes are being examined. An investigation of the use of zeolites in neutron irradiation applications was initiated. (auth)

704 (HW-SA-3604) UNIQUE PROBLEMS OF RADIOACTIVE WASTE SOLIDIFICATION. Interim Report. Cooley, C. R.; Schneider, K. J. (General Electric Co., Richland, Wash. Hanford Atomic Products Operation). Sept. 3, 1964. Contract AT(45-1)-1350. 70p. (CONF-650603-2) Dep. mn; CFSTI, $2.00 cy, $0.50 mn.

From Joint Meeting of the American Institute of Chemical Engineering and the British Institution of Chemical Engineers, London.

A pilot plant has been constructed to demonstrate the technical and economic feasibility of converting liquid radioactive wastes to solids for immobilized permanent storage. Special consideration is given to the design problems associated with converting radioactive liquids to solids ranging from powders and calcines to ceramics or glasses. Temperatures as high as 1200°C require design precautions to contain volatile components, to obtain proper materials of construction and to assure non-hazardous operating conditions. Radioactive self-heating and storage characteristics of wastes require design of processing and storage conditions to circumvent overheating. Actual equipment is illustrated, and precautions necessary to cope with processing conditions in a routine and safe manner are discussed. (auth)

705 (KR-86) QUARTERLY PROGRESS REPORT, JULY–SEPTEMBER 1964. (Norway. Institutt for Atomenergi, Kjeller). Dec. 1964. 36p. Dep.(mn).

706 (LA-3150) DISPOSAL OF Am^{241}–Pu^{239} RAFFINATE SOLUTIONS BY FIXATION WITH CEMENT. L. A. Emelity, C. W. Christenson, and E. B. Fowler (Los Alamos Scientific Lab., Univ. of California, N. Mex.). Aug. 1964. Contract W-7405-eng-36. 20p. Dep.; $1.00(cy), 1(mn) CFSTI.

In 1959 it became necessary to develop a means of disposal of a raffinate solution containing ^{241}Am and ^{239}Pu with some parent materials and daughter products. Research led to a decision to neutralize the waste and fix it in cement-vermiculite for land burial. Development of the physical plant to treat the wastes on a continuing basis is described in detail. Data on the first $4\frac{1}{2}$ years operation are presented. For control of the operation, systems for sampling the fresh mortar mixes and for conducting a study of the rate of radioactivity leach from the cured sample blocks were established. The problem of activity plating on the interior surface of the glass leach-test container is discussed and a decision to conduct leach water counting in the acid phase is explained. Leach water activity data are presented which demonstrate the excellent retention of activity by the mortar over periods extending to 2 years. The data verify that leach rate during the first few weeks immersion exceeds that of following periods due to the accessibility of block surface activity. The per gallon cost of material for disposal of the raffinate waste solution is $0.984 (prices as of August 21, 1964). (auth)

707 RADIOACTIVE WASTE DISPOSAL. PART III. C. W. Christenson (Los Alamos Scientific Lab., N. Mex.), Stephen D. Stoddard, Earnest F. Gloyna, Calvin C. Patterson, Warren H. Adams, and Eric B. Fowler. Ceram. Age, 80: No. 8, 31-4; 37-41(Aug. 1964). (LA-DC-6283)

A pilot plant operation that absorbs radioactive waste material in bisqued clay body and then vitrifies the body in an automated tunnel kiln is described. The preparation of a ceramic sponge is included. (P.C.H.)

708 (LADC-6398) FIXATION OF ATOMIC WASTES IN INERT SOLID MATERIALS. Eric B. Fowler (Los Alamos Scientific Lab., N. Mex.). [1963]. Contract W-7405-eng-36. 35p. (CONF-516-1)

From 19th Purdue Industrial Waste Conference, Lafayette, Ind., May 1964.

The fixation of high-level atomic wastes in inert, solid materials is discussed. The following methods of fixation and disposal are described: in-tank solidification, conversion of the waste into glasses and synthetic micas, fixation in montmorillonite clay, calcination, conversion into other ceramics, entrapment in other mate-

rials, and sorption of the radioactive nuclides on synthetic zeolites. (M.C.G.)

709 (ORNL-3824) A COMPUTER CODE FOR STORING AND RETRIEVING DATA AND DEVELOPING IMPROVED SOLIDS FOR FIXATION OF HIGHLY RADIOACTIVE WASTES. Fitzgerald, C. L.; Godbee, H. W.; Davis, M. H.; Clark, W. E. (Oak Ridge National Lab., Tenn.). Mar. 1966. Contract W-7405-eng-26. 106p. Dep. mn. CFSTI $4.00 cy, $0.75 mn.

A computer code was developed for storing, retrieving, and correlating the data generated by programs for the fixation of highly radioactive wastes in stable, solid media. Properties considered important for the design of equipment to produce these solids as well as the design of facilities to store them are included in the code. In addition to basic chemical composition of each mixture or melt, viscosity, thermal conductivity, heat capacity, volatility data, and corrosiveness are among these properties. The experimental data describing a melt is punched on standard 80-column data cards. The code as now programmed requires eight data cards to describe each melt. However, provisions are made so that 99 data cards may be used for each melt to provide for future additions as they are necessary. The mechanics of transferring experimental data to the data sheet is described in detail. A specific example is worked out item by item to illustrate the use of the data sheets. The flowsheet for the code and the FORTRAN II code are presented so that anyone desiring to use the code in toto can do so readily. Nearly five hundred melts were developed and included in the code. The melts were developed from wastes simulating those expected from the various Purex processes, the TBP-25 process, and the Darex process. About three hundred and fifty of the melts were developed for the Purex waste solution, about sixty for TBP-25 waste solutions, and about sixty for Darex waste solutions. A condensed readout of the more important properties for all melts developed at ORNL to date is presented. Any or all of the melt properties can be included in the readout depending on the listing desired. The code can accommodate all data currently generated and provides for possible addition of unforeseen variations in both compositions and properties. It will also provide a convenient way of exchanging data between different groups working on the fixation problem. Use of the code with additional input data to predict optimum melt compositions will be the subject of future work.

710 (ORNL-TM-1343) FIXATION OF INTERMEDIATE LEVEL RADIOACTIVE WASTE IN ASPHALT: HOT-CELL TESTS. Goode, J. H. (Oak Ridge National Lab., Tenn.). Nov. 18, 1965. Contract W-7405-eng-26. 16p. Dep. mn. CFSTI $1.00 cy, $0.50 mn.

Hot-cell tests on the fixation of intermediate-radioactivity-level waste (5 Curies/gal) in emulsified asphalt confirmed earlier successful nonradioactive and tracer tests of the process. Leaching, with static demineralized water, reached an apparent steady-state rate of about 2×10^{-4} g cm^{-2} day^{-1}, which is adequate for permanent disposal by burial in steel drums. Radiolytic-gas generation for five months, during which time the asphalt received an internal dose of about 1.2×10^{7} rad, was not measurable. The success of these tests suggests that the process should be extended to include experiments with more different radioactive wastes.

711 (UJV-1105/64) VVEDENIE RADIOIZOTOPOV V RASPLAVLENYE PORODY. (Radioisotope Incorporation into Molten Rocks). J[aroslav] Saidl (Československá Akademie Věd. Ústav Jaderného Výzkumu, Řež). [1963]. 9p.

Small diffusion coefficients secure safe incorporation of highly radioactive wastes in molten basalt when the casting is buried in underground compartments. The evolution stages prevent the diffusion of moisture and exclude elution of radioactive material. Measurement shows that Cs and Sr diffuse in crystalline basalt at 10^{-17} to 10^{-15} cm/sec^{-1} at normal temperature, guaranteeing safe fixation of even the longer-life radioactive wastes. (R.V.J.)

712 (UJV-1111/64) OTVERZHDENIE RADIOAKTIVNYKH OTKHODOV METODOM TSEMENTIROVANIYA. (Solidification of Radioactive Wastes by Cementing). L. Skheibalova and I. Novotnyi (Ceskoslovenska Akademie Ved. Ustav Jaderneho Vyzkumu, Rez). [1964]. 13p. Dep.(mn)

Solidification of radioactive wastes in concrete blocks was studied considering the mechanical properties of various cementing materials. The best results were obtained with portland cement-350. Procedures for solidification and weight ratios for mixing are given. A diagram of the installation is included. (R.V.J.)

713 CEMENTATION OF HYDROXIDE DEPOSITIONS CONTAINING RADIOACTIVE ELEMENTS. Yu. M. Butt, V. V. Timashev, L. A. Kutsenko, I. E. Kozlova, and A. V. Gordievskii. At. Energ. (USSR), 17: 124-9(Aug. 1964). (In Russian)

The possibility is demonstrated of fixing the following radioactive materials in a cement stone: deposits consisting of hydrate of iron oxide, hydrate of aluminum oxide, hydrate of manganese oxide, sulfate-silica mass, and ashes of burned rags, paper, and wood. Conditions and operation modes for constructing cemented works utilizing the wastes are worked out. Costs of cement as a function of the waste content lies in the 20 to 50% range. Different isotopes are fixed in cement to different degrees: Nb and Ru better than Cs and Sr. (tr-auth)

714 ENCASING RADIOACTIVE SLUDGE IN BITUMEN. J. Rodier, G. Lefillatre, and J. Scheidhauer. Energie Nucl., 6: No. 2, 81-8(Mar.-Apr. 1964). (In French)

The aim of the research is to transform the sludges which come from the treatment of radioactive effluents, by coprecipitation, into a solid, homogeneous product that is inert in the storage environment. The emulsifier and the chosen bitumen are added to the sludge and the water is eliminated by heating. Tests were carried out on the products obtained, in order to evaluate the diffusion of the radioelements in fresh and sea water and their resistance to microorganisms. (auth)

715 THE SAFE BURIAL OF HIGH-LEVEL FISSION PRODUCT SOLUTIONS INCORPORATED INTO GLASS. W. F. Merritt and P. J. Parsons (Atomic Energy of Canada Ltd., Chalk River). Health Phys., 10: 655-64(Sept. 1964).

High-level waste-fission-product solution containing 1000 C was incorporated into glass blocks and the blocks buried for three years in sand beneath the water table. The experiment has shown that, even in saturated soil with low exchange capacity, the method of disposal was safe and the escape of hazardous radionuclides was within acceptably low limits. (auth)

716 EMBEDDING HIGHLY RADIOACTIVE WASTES IN CONCRETE. E. Lazzarini. Ind. Ital. Cemento, 33: 45-52(Jan. 1963). (In Italian)

Fission product leaching from different mortar types was studied in normal water and brine in relation in the immersion time and curing period. Attempts were made to establish the diffusion rate of fission products through layers of cement mortars. The fission product absorption by cement mortars was also investigated. (OTS)

717 PROBLEMS RELATING TO THE TREATMENT OF RADIOACTIVE EFFLUENTS. Dejonghe, P. J. Belge Radiol., 48: 249-67(1965). (In French).

The insolubilization of concentrates in hot asphalt was studied. As a result a system, which has already been carried out on a large scale, was set-up. Also a study of radioelement dispersion into the soil was made. It is thus possible to foresee exactly the movement pattern of a radioactive contamination in the soil; this allows to establish the measures to be taken in case of accidental release of radioactivity and also to calculate the receptive capacity of a given environment. When treating high level radioactive liquid waste, a selective separation of the most toxical radioelements by means of synthetic mineral ion-exchangers is aimed at. Finally, all problems of treatment or storage were studied particularly with regard to the cost of the proceeding. (P.C.H.)

718 TEMPERATURE DISTRIBUTION IN GAMMA-ACTIVE SPHERES. Sundaram, V. K.; Gopinath, D. V.; Ganguly, A. K. (Atomic Energy Establishment, Trombay, India). Nucl. Struct. Eng., 2: 151-8(July 1965).

One of the ultimate methods proposed for high-level radioactive waste disposal is to contain the activity in ceramic bodies and bury them underground or in deep sea. The limit of the activity that can be incorporated in such bodies is decided by the temperature rise in the body due to the absorption of nuclear radiation. The temperature distribution in a solid sphere containing long-lived gamma emitters and buried in a medium having the same thermal properties is discussed. Analytical expressions are given for heat source distribution and for calculations of temperature distribution in and out of a solid sphere containing gamma activity. Numerical values obtained with energies in the range of 0.5 to 3 Mev and in spheres of radii 10, 20, and 30 cm are presented. The results were compared with those of the beta case. (auth)

Foam Separation

See also the following references:

12, 19, 20, 24, 25, 34, 36, 106, 135, 210, 213, 220, 226, 233, 235, 242, 243, 244, 247, 249, 776, 961

719 FOAM COLUMN (Engineering Materials). (Oak Ridge National Lab., Tenn.). (CAPE-1100)

This package contains 3 drawings. For a listing of individual titles, see the Engineering Materials List (TID-4100). This 24 in. foam separation column is used in the countercurrent exchange of solutes between a liquid and the surface of a foam. This method of separation is employed in low-level waste processing. Sight glasses and light ports were fabricated from Plexiglas for the carbon steel column. (M.C.G.)

720 (NYO-10038) FISSION PRODUCT SEPARATION BY FOAM EXTRACTION. Jacques J. Weinstock, Sarah Mook, Ernest Schonfeld, Eliezer Rubin, and Robert Sanford (Radiation Applications Inc., Long Island City, N. Y.). June 24, 1963. Contracts AT(30-1)-2384 and AT(30-1)-2093, Subcontract 2024. 123p.

A study was made of the applicability of the principles of foam separation to the enrichment and recovery of specific fission product nuclides from typical nuclear waste solutions. A review of foam separation principles is given and the experimental methods and equipment used are described. Investigation of equilibrium included studies of the characteristics of the experimental apparatus (e.g., time required to reach equilibrium), the effects of metal ion concentration and foaming agent concentration on interfacial adsorption equilibrium, and the effect of the choice of foaming agent on the potential degree of separation attainable. Screening of commercial foaming agents established specific foaming agents and types of agents which are effective in the removal of cesium, strontium, and cerium ions from aqueous solution. Each surfactant was tested for solubility, foamability, and enrichment at three different pH values, and for the critical micelle concentration (CMC). Out of 107 surfactants tested, 73 gave a reasonably good foam at one or more pH values and were tested for metal ion enrichment. Twenty were found promising in cesium removal, 33 in strontium removal, and 10 in cerium removal. Generally, only the anionic type surfactants were found effective, and each one of the three metal ions was found to be best separated within a specific pH range. Also included in this evaluation of specific reagents was the study of several combinations of materials which resulted in promising separations of cesium from solution. Studies were made of multistage foam separation column variables. Among these variables are the effects of: reflux, foam column height, expanded column head for drainage, splitting foaming agent feed between column top and bottom, and countercurrent foam washing. A detailed discussion and evaluation of the experimental results relative to the chemical separation of radioisotopes is given. This discussion of foam separation applicability includes an account of the current status of foam separation technology and a summary of the effects of pH, of the concentrations of various ionic species and foaming agents, and of physical operating variables. A wide range of approximate process operating costs is given ($0.40 to $3.00 per 1000 gallons of feed solution), but it is concluded that current foam separation technology does not allow the preparation of reliable flowsheets or cost estimates for a foam separation isotope recovery plant. (auth)

721 (ORNL-3527) ENGINEERING DEVELOPMENT OF A FOAM COLUMN FOR COUNTERCURRENT SURFACE-LIQUID EXTRACTION OF SURFACE-ACTIVE SOLUTES (thesis). Haas, P. A. (Oak Ridge National Lab., Tenn.). June 1965. Contract W-7405-eng-26. 245p. Dep.(mn); $6.00(cy), 5(mn) CFSTI.

Submitted to Univ. of Tennessee, Knoxville.

The foam separation process uses differences in surface activity to effect the separation of components of a solution. Engineering problems were solved and methods developed for the design and operation of countercurrent liquid-foam columns. Six- and 24-in.-ID columns were operated to measure contacting efficiency, foam densities, foam drainage, and foam condensation. The experimental results agree with a model for countercurrent liquid-foam flow based on capillary flow through the Plateau borders between foam bubbles. Data for the drainage of the liquid from the foam flowing vertically or horizontally also agree with the same model. The heights of transfer units based on the liquid phase (HTU_x) were about 1 cm for the best conditions of uniform foams of 0.56- to 1.0-mm bubble diameter, liquid flows of 100 gal/ft^2-hr or less, and uniform liquid feed distribution with low inlet velocities. The abil-

ity of gas spargers to give uniform foam bubbles was important; spinnerets with 50- or 80-μ-dia holes as used by the rayon industry were superior to porous metal or fritted glass. There was little variation in HTU_x values for countercurrent column lengths of 10 to 28 cm. The major effects of liquid and foam flow rates and of liquid feed distributors on HTU_x values were due to their influence on the extent of channeling. The foams were condensed by means of orifices, air-operated sonic whistles, screen-lined centrifugal foam breakers, or cyclones. The removal of radioactive strontium to decontaminate ORNL low-level waste was shown to be practical if preceded by a precipitation step (head-end step) to decrease the calcium and magnesium concentrations. (auth)

722 (RAI-116) THE REMOVAL OF METALLIC IONS BY FOAMING AGENTS AND SUSPENSIONS: LABORATORY AND ENGINEERING STUDIES. Annual Report, July 1, 1962 to June 30, 1963. Eliezer Rubin, Ernest Schonfeld, and Jacques Weinstock (Radiation Applications Inc., Long Island City, N. Y.). Dec. 16, 1963. For Oak Ridge National Lab., Tenn. Contract W-7405-eng-26, Subcontract 2024. 47p.

It was shown that two processes involving foam separation may be used to remove ^{90}Sr from low-activity waste (LAW) solutions: pretreatment with NaOH and Na_2CO_3, followed by foam separation and simultaneous phosphate-iron coagulation, froth flotation, and foam separation. Foam-separation processing of ORNL LAW without pretreatment of any kind resulted in very low decontamination factors (DF's) for strontium. Pretreatment of ORNL LAW by pH adjustment followed by foam separation required very long waiting periods, prior to filtration and foaming, to achieve high DF's. Further evaluation and development of the $NaOH-Na_2CO_3$ pretreatment process, using a 9-inch-diameter suspended-bed scavenger and actual ORNL LAW at flow rates of 10 to 60 gal-hr^{-1}-ft^{-2} resulted in decontamination factors of 10 to 15 for ^{90}Sr and effluent total hardness of 2.0 to 4.5 ppm. This may be compared with decontamination factors of 3 to 4 for ^{90}Sr and total effluent hardness of 35 to 60 for the present ORNL lime-soda process with the normally used feed flow rate of 21 gal-hr^{-1}-ft^{-2}. Pretreatment of synthetic LAW by this process followed by foam separation resulted in overall Sr^{2+} decontamination factors of over 15,000. Similar experiments with actual ORNL LAW gave overall Sr^{2+} DF's of approximately 300. The combination of phosphate-iron coagulation and foam separation into a one-step continuous process resulted in run-average decontamination factors up to 2000 for strontium. This highest average DF was obtained for a 7-hour experiment conducted in a continuous apparatus and using synthetic ORNL LAW at a flow rate of 25 gal-hr^{-1}-ft^{-2}, air to feed flow rate ratio of 12.4, 67 mg/l PO_4^{3-} and 23.4 mg/l Fe^{2+} as coagulating agents, 90 mg/l dodecylbenzenesulfonate (DBS) as a foam separation agent, and 15 mg/l Armeen Z as a floating agent. An incomplete evaluation of this process indicated that: a 7-minute residence time in the mixing beaker, prior to foaming, results in lower Sr^{2+} DF's than residence times in the range of 17 to 23 minutes; higher DBS concentration may be required for treating actual ORNL-LAW than those required for synthetic LAW; intermittent addition of grundite clay into the mixing beaker, in addition to PO_4^{3+}, Fe^{2+}, Armeen Z, and DBS, prior to foaming produced only very low decontamination factors for Cs^+; and the use of split surfactant feed produced average Sr^{2+} DF's of up to 580 with actual ORNL LAW solutions as feed. (auth)

723 (RAI-350) THE REMOVAL OF METALLIC IONS BY FOAMING AGENTS AND SUSPENSIONS: LABORATORY AND ENGINEERING STUDIES. Annual Report, July 1, 1963–June 30, 1964. Weinstock, Jacques J. (Radiation Applications Inc., Long Island City, N. Y.). Mar. 23, 1965. Contract W-7405-eng-26, Subcontract 2024. 41p. Dep. mn; CFSTI $2.00 cy, $0.50 mn. For Oak Ridge National Lab., Tenn.

A research program was conducted on the application of foam separation to the decontamination of the low-activity waste (LAW) stream at ORNL. The continued study of the combined foam separation and phosphate-iron coagulation process established the following optimum concentrations of reactants: Armeen Z (flotation agent), None; UCAR C-149 (coagulation agent), 3 mg/l; WSR-35 (coagulation agent), 5 mg/l; Na_3PO_4, 60-70 mg/l; $FeSO_4$ (freshly prepared), 64 mg/l; DBS in feed, ~133 mg/l; and DBS to bottom of column, ~17 mg/l. All these concentrations are given in terms of milligrams per liter of LAW. The study also established the fact that a relatively long mixing time (at least 15-20 minutes) is necessary to make the process effective. The screening of biodegradable surfactants included experimental determinations of solubilities, critical micelle concentrations, foamabilities, and single-stage equilibrium distribution factors for strontium. The two materials which gave the best performance in these tests were Alipal LO 436 and Nacconol 40 FX. These materials were then further tested in a 2-inch-diameter, multistage, foam-separation column, and both gave decontamination factors at least equal to the best previously obtained with DBS (Siponate DS-10). The examination of pressure drops in foam-carrying tubes produced data that can be correlated in terms of a friction factor as a function of Reynolds Number. This correlation is similar to that used with pipes carrying more conventional fluids. (D.L.C.)

724 FOAM COLUMNS FOR COUNTERCURRENT SURFACE—LIQUID EXTRACTION OF SURFACE-ACTIVE SOLUTES. Paul A. Haas and H. F. Johnson (Oak Ridge National Lab., Tenn.). A. I. Ch. E. (Am. Inst. Chem. Engrs.) J., 11: 319-24(Mar. 1965).

Foam separation processes use differences in surface activity to separate components of a solution. Stable foams can flow upward countercurrent to liquid to give countercurrent differential separations with stripping, enrichment, or scrubbing similar to solvent extraction. Six- and 24-in. ID foam columns were operated with sodium dodecylbenzenesulfonate solutions to measure height of transfer unit (HTU) values and foam densities and to develop satisfactory gas spargers, liquid feed distributors, foam drainage conditions, and foam condensation equipment. The HTU values for stripping of ^{89}Sr were about 1 cm for the best conditions of uniform foams, liquid flows of 100 gal/ft^2 hr or less, and uniform liquid feed distribution with low inlet velocities. Variations of countercurrent column lengths within 10 to 28 cm caused little variation in HTU values. The foams were condensed, with four types of pneumatic or mechanical foam breakers used. A pilot plant was designed for decontamination of a low level radioactive waste. (auth)

725 STUDIES ON THE TREATMENT OF RADIOACTIVE LIQUID WASTES BY FLOTATION METHOD. [PART] I. Koyanaka, Yoshio (Kyoto Univ.). Nippon Genshiryoku Gakkaishi, 7: 621-6(Nov. 1965). (In Japanese).

The removal of ^{141}Ce, ^{95}Zr, ^{95}Nb, ^{106}Ru, and ^{106}Rh from dilute aqueous solutions was investigated using the technique of soap flotation. Cerium nitrate, zirconium oxychloride, and ruthenium chloride were used as inactive carrier materials. From alkaline dilute solutions, 99.0% of the ^{141}Ce, 99.5% of the ^{95}Zr and ^{95}Nb, and 99.5% of the ^{106}Ru and ^{106}Rh were removed by soap flotation method using sodium oleate as flotation collector and ferric chloride as coprecipitating reagent.

726 STUDIES ON SEPARATION OF FISSION PRODUCTS BY FLOTATION METHOD. SEPARATION OF ^{137}Cs. Koyanaka, Yoshio (Kyoto Univ.). Radioisotopes, 15: 77-82(1966). (In Japanese).

Flotation proved to be effective in removing ^{137}Cs from its aqueous dilute solution, with the aid of suitable coprecipitant such as cupric or nickel ferrocyanide and pertinent collector such as sodium oleate and octadecylamine acetate. It was found in the experiments on single stage flotation that 90 ~ 99% of ^{137}Cs can be removed by optimum choice of the concentrations of coprecipitant and collector, as well as of pH of flotation medium. Multiple flotation for four successive stages gave a decontamination factor of 10^4 to 10^6 with a volume reduction ratio of higher than 50.

727 CONCENTRATION OF RADIOACTIVE ISOTOPES WITH COLLOIDAL MANGANESE DIOXIDE BY FOAMING. Pushkarev, V. V. pp 124-9 of Radiokhimicheskie Metody Opredeleniya Mikroelementov. Moscow-Leningrad, Izdatel'stvo Nauka, 1965. (In Russian).

The radioactive isotopes ^{89}Sr, ^{90}Sr, ^{90}Y, ^{91}Y, ^{95}Zr, ^{95}Nb, ^{106}Ru, ^{106}Rh, ^{137}Cs, ^{144}Ce, and ^{144}Pr were separated and concentrated from aqueous solutions by foam chromatography, colloidal manganese dioxide being used as the carrier and gelatin as the foaming agent. The optimum amounts of gelatin at various pH values were determined. An attempt was made to obtain a precipitate with the lowest possible moisture content (1.5–2.0% of the volume of the initial solution). It was found that this content is substantially affected by the pH of the solution, amount of gelatin, rate of blowing air through the column of liquid, and difference in the levels of the liquid surface and height at which the foam fraction is poured off. It was found that from 97.0 to 99.0% yttrium at pH ≥ 6.0, zirconium with niobium at pH ≥ 2.0, and cerium at pH ≥ 3.0 are recovered from the solution with the foam product. When the foaming is repeated, these isotopes are recovered from the solution to the extent of over 99.9%. The maximum recovery of ruthenium (pH = 4.0–5.0) and cesium (pH > 6.0) is 55.0–65.0%, and that of strontium (pH = 10.0), 86.0%.

728 ENGINEERING DEVELOPMENT OF A FOAM COLUMN FOR COUNTERCURRENT SURFACE-LIQUID EXTRACTION OF A SURFACE-ACTIVE SOLUTES [Thesis]. Haas, Paul Arnold. Knoxville, Tenn., Univ. of Tennessee, 1965. 241p.

The foam separation process uses differences in surface activity to effect the separation of components of a solution. Engineering problems were solved and methods developed for the design and operation of countercurrent liquid-foam columns. Six- and 24-in.-ID columns were operated to measure contacting efficiency, foam densities, foam drainage, and foam condensation. The experimental results agree with a model for countercurrent liquid-foam flow based on capillary flow through the Plateau borders between foam bubbles. Data for the drainage of the liquid from the foam flowing vertically or horizontally also agree with the same model. The heights of transfer units based on the liquid phase (HTU_x) were about 1 cm for the best conditions of uniform foams of 0.56- to 1.0-mm bubble diameter, liquid flows of 100 gal/sq ft-hr or less, and uniform liquid feed distribution with low inlet velocities. The ability of gas spargers to give uniform foam bubbles was important; spinnerets with 50- or 80-μ-diam holes as used by the rayon industry were superior to porous metal or fritted glass. There was little variation in HTU_x values for countercurrent column lengths of 10 to 28 cm. The major effects of liquid and foam flow rates and of liquid feed distributors on HTU_x values were due to their influence on the extent of channeling. The foams were condensed by means of orifices, air-operated sonic whistles, screen-lined centrifugal foam breakers, or cyclones. The removal of radioactive strontium to decontaminate Oak Ridge National Laboratory low-level waste was shown to be practical if preceded by a precipitation step (head-end step) to decrease the calcium and magnesium concentrations. (Dissertation Abstr.)

Incineration

729 (DP-942) BURNING OF RADIOACTIVE PROCESS SOLVENT. D. Whitney Tharin, Jr. (Du Pont de Nemours (E. I.) & Co., Aiken, S. C. Savannah River Lab.). Feb. 1965. Contract AT(07-2)-1. 16p. Dep.; $1.00(cy), 1(mn) CFSTI.

Spent solvent from radiochemical separations processes is accumulated and burned in burial ground facilities. Between mid-1955 and February 1964, 290,000 gallons were stored or burned with negligible release of radionuclides. (auth)

730 (DPSPU-64-30-8B) A METHOD FOR DISPOSAL OF LARGE QUANTITIES OF DEGRADED AND LOW-LEVEL CONTAMINATED SOLVENT. J. A. Harper and L. Jolly, Jr. (Du Pont de Nemours (E.I.) Co., Aiken, S. C. Savannah River Plant). May 1964. Contract AT-(07-2)-1. 12p. (CONF-642-31). Dep.(mn); $1.00(cy), 1(mn) CFSTI.

From Health Physics Society, Cincinnati, June 1964.

Solvent used in radiochemical separations processes is degraded during exposure to radiation, and retains certain fission products. Disposal of large quantities of degraded solvent is a waste disposal problem of considerable magnitude. Of several disposal methods considered, controlled open pan burning was selected as the safest and most economical. An extensive air sampling program showed that more than 99% of all the solvent radioactivity, primarily fission products and plutonium, remained in a gummy residue. Over a period of two years, 80,000 gallons of solvent was burned which contained about 8 curies of gamma emitting isotopes and about 60 millicuries of alpha emitting isotopes. No adverse effect on the environment adjacent to the burning operations was detected. (auth)

731 (NLCO-894) INCINERATION OF WASTE-CONTAMINATED OIL. K. E. Brandner, R. L. Bipes, and L. Williams (National Lead Co. of Ohio, Cincinnati). Sept. 18, 1963. Contract AT(30-1)-1156. 7p. (CONF-255-7)

From 8th Air Cleaning Seminar Oak Ridge, Tenn., Oct. 1963.

Of the various disposal methods investigated for organic liquid wastes contaminated with uranium, incineration proved to be the most successful. An oil burner is now operating on site capable of disposing of 400 to 700 fifty-five gallon drums of mineral oils and emulsions monthly. Uranium recovery from the process is virtually complete, and air contamination is held to acceptable levels. (P.C.H.)

Incorporation in Glasses

See also the following references:

15, 19, 24, 31, 36, 46, 47, 51, 77, 79, 85, 106, 122, 123, 178, 180, 181, 187, 188, 190, 191, 193, 194, 195, 201, 203, 204, 209, 210, 213, 216, 220, 223, 224, 225, 226, 227, 233, 235, 245, 246, 247, 248, 249, 547, 555, 561, 568, 573, 575, 698, 699, 702, 703, 704, 708, 709

732 (A/CONF.28/P/188) WORK IN THE UNITED KINGDOM ON FIXATION OF HIGHLY RADIOACTIVE WASTES IN GLASS. K. D. B. Johnson, J. R. Grover, and W. H. Hardwick (United Kingdom Atomic Energy Authority. Research Group. Atomic Energy Research Establishment, Harwell, Berks, England). May 1964. 15p.

Experimental and theoretical work was carried out on the fixation in glass of the wastes from power reactors using U metal fuel of near-natural isotopic composition. The operating experience of the FINGAL pilot plant is described, and its scale-up is discussed. The storage conditions are considered. It is concluded that the costs of fixation and storage of wastes are small compared with the electric power produced. (D.L.C.)

733 (A/CONF.28/P/587) THE THERMIC METHOD OF VITRIFICATION OF THE RADIOACTIVE PULPS AND THE SAFE DISPOSAL PROBLEM OF VITREOUS PREPARATIONS. N. I. Bogdanov, V. G. Vereskunov, F. S. Dukhovich, P. V. Zimakov, B. S. Kolychev, V. S. Kuzucheva, V. V. Kulichenko, V. P. Martynov, and E. T. Rastorguev (U.S.S.R.). May 1964. 20p.

A continuous thermal treatment process for converting radioactive pulps containing metal hydroxide mixture into vitreous preparations is discussed, together with the conditions for safe disposal of these preparations. The process physico-chemical principles and bases for using a fluidized bed of quartz sand for dehydration of the pulps and calcination of their solid residue are outlined. The principal plant flow diagram is presented, and the main results of its testing in a pilot plant are given. A study of the radiation stability of vitreous preparations showed that surface destruction occurred during the irradiation, leading to increased solubility of the preparations and to the formation of the mechanically undurable and radiation instable products on the surface. It was found that the destruction of the surface was caused by the radiation chemical reactions of the components of fused preparations with the air components. The process rate decreases with increasing temperature as the gas adsorption on the surface of the preparations decreases and the annealing of radiation defects occurs. Simpler storage constructions for the highly active vitreous preparations as compared to the storages for the liquid and the other types of solid wastes are proposed on the basis of complex investigations of properties of vitreous preparations. On optimum temperature for ground disposal of about 700°K during the first period of storage is recommended on the basis of experiments with real vitreous preparations and electrical modelling with simulated preparations, as well as on the basis of thermophysical calculations. The storage dimensions and arrangements are defined. (auth)

734 (AERE-R-4685) THERMAL CONDUCTIVITY OF A BOROSILICATE GLASS SUITABLE FOR FISSION PRODUCT WASTES. D. B. Auty and J. R. Grover (United Kingdom Atomic Energy Authority. Research Group. Atomic Energy Research Establishment, Harwell, Berks, England). Statistical Appendix by B. E. Cooper. Dec. 1964. 24p. Dep.; $0.70(BIS); 3s. 6d.(HMSO).

The thermal conductivity of a borosilicate glass incorporating mock fission product wastes has been measured by a radial heat flow method and found to vary from 0.00247 cal/cm-sec-°C at 200°C to 0.00281 cal/cm-sec-°C at 550°C. The temperature drop across the glass/steel interface is small and hardly significant when calculating the axial temperature in cylinders of glass in storage.

735 (BNL-10109) BROOKHAVEN NATIONAL LABORATORY PROCESS FOR THE CONTINUOUS CONVERSION OF HIGH-LEVEL RADIOACTIVE WASTE TO PHOSPHATE GLASS. Tuthill, E. J.; Weth, G. G.; Emma, L. C.; Strickland, G.; Hatch, L. P. (Brookhaven National Lab., Upton, N. Y.). [1965]. Contract AT(30-2)-Gen-16. 36p. (CONF-660208-16). Dep. mn. CFSTI $2.00 cy, $0.50 mn.

From Symposium on Solidification and Long-Term Storage of Highly Radioactive Wastes, Richland, Wash.

The results of an investigation at Brookhaven on the development of the phosphate glass process are reported with emphasis on the performance characteristics of the individual process units. The process was tested on a pilot plant scale for some 3000 hr using simulated Purex waste most of the time. In the Brookhaven process, phosphoric acid is added to the aqueous waste and the solution converted to phosphate glass in a continuous operation upon raising the temperature in two steps to 1100–1200°C.

736 (BNL-10132) PLATINUM FOR HIGH TEMPERATURE CRUCIBLES USED IN PROCESSING RADIOACTIVE WASTE. Weth, G. G.; Strickland, G.; Tuthill, E. J. (Brookhaven National Lab., Upton, N. Y.). [nd]. Contract AT-30-2-GEN-16. 14p. (CONF-660208-17). Dep. mn. CFSTI $1.00 cy, $0.50 mn.

From Symposium on Solidification and Long-Term Storage of Highly Radioactive Wastes, Richland, Wash.

In the development of the phosphate glass process for conversion of highly radioactive wastes to stable solids, platinum was used successfully for the high temperature crucible. In this application a platinum crucible was used at temperatures of 800–1200°C for a total of 8000 hours with no signs of deterioration. It was demonstrated that chemical reducing conditions must be avoided; otherwise rapid deterioration of platinum may take place in the presence of molten phosphate. Tests to determine the useful life of platinum included long-term creep measurements at 1250°C. The latest results of these tests, which are still in progress, are presented. Economic factors associated with the use of platinum in high level waste processing are discussed briefly.

737 (BNWL-76) RESEARCH AND DEVELOPMENT ACTIVITIES–FIXATION OF RADIOACTIVE RESIDUES. Quarterly Progress Report, January–March 1965. Platt, A. M. (ed.) (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). Apr. 1965. Contract AT(45-1)-1830. 43p. Dep.(mn); $3.00(cy), 1(mn) CFSTI.

Leaching test results on high-level phosphate glass show that dry storage is essential for maintaining low leachability, moist air being more deleterious to the glass surface than immersion in water. Ground water at pH 7.8 was found to yield leach rates about twice those obtained with double-distilled water. Solubility measurements on the phosphate mixtures of synthetic fission product and process oxides containing 3, 10, and 50% fission products were completed. Comparative measurements for a sulfate-free waste show that over the desirable phosphate range there is little effect produced by the presence of sulfate. Examination of the hot spray calciner revealed extensive corrosion of the Inconel wall, with some apparent stress cracking. Ru does not seem to be preferentially absorbed on the calciner off-gas line. Evolution of SO_3 from melts composed of Purex 1WW waste and phosphate (M/P = 2.3) was greater than from Purex 1WW-B_2O_3 melts. The percentage of sulfate retained in the 1WW-phosphate melts when held at 950°C for 3 hr decreased as the concentration of sulfate was increased. The fusion point of melts containing added phosphate increased from 940°C at an M/P of 1.7 to 1040°C at an M/P of 3.1. Cooled melts changed from glassy to rock-like and nonhomogenous as the M/P was increased. A Pt-Inconel 600-Pt sandwich suffered catastrophic attack when exposed in a Purex 1WW melt at 1100°C but did not at lower temperatures. Addition of solid fluxing agents to the spray calciner-melter as a means of glass formation was found to be feasible in a run with liquid 1WW and a separate addition of borate-containing glass balls. Fabrication of the phosphate glass system for waste solidification is underway. Fabrication of the spray calciner-melter system is nearing completion. The waste solidification evaporator and acid fractionator has undergone testing in pot calcination runs. Results of sealing and weld joint tests on stovepipe waste container heads indicate that such heads can be

used to simplify fabrication and reduce costs. Studies are continuing on crystallization of sulfates from acid condensates generated in glass making processes. A scheme which employs scavenging, ion exchange, and adsorption was investigated for the treatment of intermediate level laboratory-type waste. A pilot scale run with a horizontal plate filter and anion and cation exchange columns removed all radioisotopes except ^{106}Ru from a steam stripped condensate waste stream to below MPC_w values. (J.R.D.)

738 (BNWL-80) SOLIDIFICATION OF HIGH LEVEL WASTES. PART IV. PHOSPHATE MELTS FOR FIXATION OF RADIOACTIVE RESIDUES FROM PUREX TYPE WASTES—THREE TO FIFTY PER CENT FISSION PRODUCT OXIDES. Barton, G. B. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). July 1965. Contract AT(45-1)-1830. 39p. Dep. mn; CFSTI, $2.00 cy, $0.50 mn.

The possible composition range for the high-level radioactive wastes generated from various power reactor fuels processed through a Purex-type process is very broad. A partial survey of this wide range was conducted in an effort to establish the composition limits that bound the operable range of the phosphate melt system. Two screening criteria were used: (a) drip temperature, an empirical measure of flow temperature at which a container could be filled, and (b) solubility. Measurements indicated the occurrence of two (or possibly more) optimum phosphate additions near M_2O/P_2O_5 ratios of 2.1 and 1.0. It was found that fission products equivalent to at least 50% of the weight of oxides in the plant waste can be successfully incorporated in a phosphate melt that can be transferred as a fluid into containers at less than 900°C. (auth)

739 (BNWL-139) RESEARCH AND DEVELOPMENT ACTIVITIES, FIXATION OF RADIOACTIVE RESIDUES. Quarterly Progress Report, April–June 1965. Platt, A. M. (ed.) (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). Sept. 1965. Contract AT(45-1)-1830. 36p. Dep. mn; CFSTI $2.00 cy, $0.50 mn.

Samples of phosphate glass produced from Purex waste in the Hot Cell Glass Experiment did not show appreciable change in leachability over a seven-month period since they were made. However, they are showing evidence of separation of a minute second phase. Laboratory studies of glass formulations indicated that the properties of the glass are affected by the precious metal and fission product contents of the glasses. Flowsheet development with simulated NFS 1st and 2nd cycle waste and added phosphoric acid was completed with a successful 40-hr spray calciner run. Metallic cation equivalent to phosphorous atom ratio (M^+/P) equal to 3.1 was the most successful composition. A feed composed of a sulfate-bearing condensate adjusted to melt with caustic was not operable in the spray calciner-melter. Other tests included successful thermal shock cleaning of the calciner wall, an acceptable air vibrator to replace an air hammer, and wire filters. Design verification testing of the pot calciner prototype and its related WSEP hardware was successfully completed. Preliminary fission product release tests were made with 15 g samples of simulated waste material spiked with ^{137}Cs. Release fractions of up to one percent were observed during one hour at temperatures up to 1200°C. The reaction between the molten waste material and the stainless steel containers was briefly studied. Temperatures at Purex waste tanks in infinite arrays with 120-foot center-to-center tank spacings were compared with those for single tanks with no neighbors. A recycle flowsheet to improve cesium product purity was demonstrated. Nominal Na/Cs ratios of 8/1 were reduced to 1/3.4 by use of recycle and increased scrub volumes.

740 (BNWL-SA-564) FIXATION OF HIGH-LEVEL RADIOACTIVE WASTES IN PHOSPHATE GLASS. Upson, U. L.; Weth, G. C. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). Feb. 1, 1965. Contract AT (45-1)-1830. 22p. (CONF-660208-14). Dep. mn. CFSTI $1.00 cy, $0.50 mn.

From Symposium on Solidification and Long-Term Storage of Highly Radioactive Wastes, Richland, Wash.

The Brookhaven process for converting high-level aqueous fuel reprocessing waste directly to a phosphate glass, without an intermediate drying step, was demonstrated with actual Purex 1WW waste. A small-scale pilot plant was remotely installed in a hot cell containing the equipment from a preceding study on waste calcination. Technical feasibility of the process was tested, operating characteristics were observed, and the off-gas decontamination problem evaluated. Samples of glass were produced for observation and evaluation of stability and chemical durability.

741 (ERI-6) RESULTS OF PRELIMINARY SURVEY AND MONITORING OF THE FIRST FIELD TEST OF THE DISPOSAL OF FISSION PRODUCTS INCORPORATED INTO GLASS. W. F. Merritt and I. L. Ophel (Atomic Energy of Canada Ltd., Chalk River, Ont.). Feb. 1960. Reprinted Mar. 1964. 18p. (AECL-1918)

Twenty-five glass blocks containing about three hundred curies of aged mixed fission products have been buried in a selected area of the Chalk River disposal area. Before the burial, the direction and velocity of ground water movement were measured using tritiated water, sulphur-35, and fluorescein dye as tracers. After the burial the ground water was monitored to detect leaching of fission products and soil samples have been obtained from the vicinity of the blocks. To date, no radionuclides attributable to leaching of the blocks have been detected in the ground water. A small quantity of strontium-90 has been found in some of the soil samples. (auth)

742 (IDO-14672) LEACHABILITY OF GLASS PREPARED FROM HIGHLY RADIOACTIVE CALCINED ALUMINA WASTE. Paige, B. E. (Phillips Petroleum Co., Idaho Falls, Idaho. Atomic Energy Div.). Feb. 1966. Contract AT(10-1)-205. 32p. Dep. mn. CFSTI $2.00 cy, $0.50 mn.

Alumina-phosphate glass was prepared from full-level radioactive calcine produced in the Waste Calcining Facility at the Idaho Chemical Processing Plant. The glass was prepared from 25 weight percent alumina calcine, 40.5 percent P_2O_5 and 18.6 percent Na_2O added as $NaH_2PO_4 \cdot H_2O$, and 15.9 percent PbO. The details of preparation and the effect of radiation and heat treatment on the glass are reported. Leaching data obtained for fission products and the glass components are compared for the glass in distilled water at 25 and 94°C. A specially designed apparatus which circulated the water continuously was used. The high level of activity made it feasible to use a single molded piece permitting the surface area and other experimental conditions to be defined to a far greater degree than is feasible with the crushed glass techniques required for low levels of activity. Devitrification by heating for 30 days at 450°C was found to have no effect on the leaching rate.

743 (ORNL-3612) LABORATORY DEVELOPMENT OF PROCESSES FOR FIXATION OF HIGH-LEVEL RADIOACTIVE WASTES IN GLASSY SOLIDS: WASTES CONTAINING (1) ALUMINUM NITRATE AND (2) THE NITRATES OF THE CONSTITUENTS OF STAINLESS STEEL. W. E. Clark and H. W. Godbee (Oak Ridge National Lab., Tenn.). July 1964. Contract W-7405-eng-26. 59p.

Glassy solids satisfactory for the long-term storage ("ultimate disposal") of radioactive wastes were prepared from solutions simulating those obtained or expected from aqueous reprocessing of aluminum- and stainless steel-

bearing fuel elements. Only tracer levels of radioactivity were tested. Vapor-liquid equilibria and solution densities were determined for both waste types over wide ranges of composition. Phosphate was the principal fluxing agent in all products reported. Borate and sodium were employed as other additives with both waste types. Aluminum generally improved the glassiness of the products obtained with stainless steel waste, while lead was helpful in lowering the softening temperature of the melts obtained with the aluminum waste. Volume reductions (volume of waste solution/volume of glass) varied from 7.2 to 9.3 for the aluminum waste and from 2.5 to 6.6 for the stainless steel waste. Leaching tests on small specimens of the more satisfactory solids prepared from both waste types gave dissolution rates of about 10^{-6} g/cm^2 day based on the fraction of ^{137}Cs spike leached from the product. Semicontinuous fixations of both waste types were run on a semiengineering scale. These were highly successful for the aluminum waste. The melt used for fixation of the stainless steel waste was excessively corrosive and produced a solid containing excessive voids. Use of stainless steel fixation pots appears feasible with both waste types, but titanium is preferred for the construction of an evaporator for concentration of the waste solution. Thermal conductivity of one aluminum-type glass varied from 1.05 Btu hr^{-1} ft^{-1} °F at 300°F to 1.60 Btu hr^{-1} ft^{-1} °F^{-1} at 1050°F. Fixation of both waste types in phosphate "glasses" is feasible on the basis of current technology. Further work needs to be done to define the compositions of products having optimum properties. (auth)

744 (ORNL-3629) CONVERSION OF SIMULATED HIGH-LEVEL RADIOACTIVE WASTE SOLUTIONS TO GLASSY SOLIDS IN A POT BY A RISING LIQUID LEVEL METHOD: THE RL-POTGLASS PROCESS. H. W. Godbee and W. E. Clark (Oak Ridge National Lab., Tenn.). Aug. 1964. Contract W-7405-eng-26. 25p.

The permanent disposal of reactor wastes very probably will require their burial as solids. If so, glassy solids may be more desirable than porous solids, such as calcined oxides. Twelve experiments were completed using a rising-liquid-level (RL-Potglass) process for the conversion of high-level radioactive waste solutions to glassy solids in a 4-in.-dia and 24-in.-long pot at average feed rates of 32 to 45 ml/min. The procedure, the results, and the operating difficulties encountered during these experiments are described. Fixation of TBP-25 (aluminum), Purex, and Darex (stainless steel) simulated high-level wastes was demonstrated; phosphate was used as the principal fluxing agent. Other added ions include aluminum, lithium, sodium, calcium, boron, and lead. The glassy solid obtained with TBP-25 waste had a density of 2.86 g/cm^3, contained 26 wt % oxides from waste, and represented a volume reduction (volumes of waste per volume of solid product) of 8. The "noncorrosive" product obtained with Purex (FTW-65) waste had a density of 2.41 g/cm^3, contained 44.5 wt % oxides from the waste, and represented a volume reduction of 36. The solid product from a single experiment with Darex waste contained 25 wt % oxides from the waste and represented a volume reduction of 5. It was concluded that glassy solids could also be formed in pots as an alternative to forming calcined solids in a waste processing plant designed for the conversion of wastes to solids by the pot process. Pressure surges in the pot were successfully held to 2 psi or less, and the levels of boiling solution, calcine, and melt were detected satisfactorily with thermocouples or electrical conductivity probes. Corrosion was satisfactorily low for the Purex and TBP-25 systems. Corrosion was excessive in an initial test with the Darex system when an early melt composition was used. However, the corrosion rates are expected to be satisfactorily low when improved melt compositions are used. (auth)

745 (ORNL-3640) LABORATORY DEVELOPMENT OF PROCESSES FOR FIXATION OF HIGH-LEVEL RADIOACTIVE WASTE IN GLASSY SOLIDS. 3. WASTES FROM THE PUREX SOLVENT EXTRACTION PROCESS. Clark, W. E.; Godbee, H. W.; Fitzgerald, C. L. (Oak Ridge National Lab., Tenn.). June 1965. Contract W-7405-eng-26. 61p. Dep.(mn); $3.00(cy), 2(mn) CFSTI.

Glassy solids suitable for the long-term storage ("ultimate disposal") of radioactive wastes were prepared from solutions simulating those obtained or expected from the processing of nuclear fuel by the Purex solvent extraction process. The studies were applicable to both rising level (RL-Potglass) and continuous glass (Con-Potglass) processes. The simulated waste solutions studied varied from those in which sodium sulfate was the principal constituent to those which contained no sulfate. Simulated fission products varied from zero to amounts expected from a burn-up of 35,000 Mwd/metric ton. Actual radioactivity was limited to tracer amounts used in determining leaching rates and volatilities. Minimum volumes of "glass" obtained (at 0.2% simulated fission products) corresponded to about 2.7 gallons of solid per ton of uranium processed. This solid contained 46% (by weight) waste oxides (process oxides plus simulated fission products), 31.5% P_2O_5, 11.8% Al_2O_3, 7.4% Li_2O, and 4.9% CaO. Melts incorporating higher concentrations of fission products required higher percentages of fluxing agents to keep melting temperatures and viscosities low enough for practical operation in stainless steel equipment. Maximum temperatures allowable are about 1050 and 900°C for the rising level (RL-Potglass) and the continuous melting glass (Con-Potglass) processes, respectively; lower operating temperatures are desirable in both cases. Phosphate, aluminum, sodium, lithium, and calcium were the principal fluxing agents used for sulfate-containing wastes. The most glassy non-sulfate melts were obtained with phosphate and lead as fluxes. Increasing the content of aluminum or of fission products generally increased softening and melting temperatures but resulted in a more uniform product. Addition of lithium to give a final ratio, Na/Li $\cong 1$ usually resulted in less soluble, more homogeneous, and lower melting solids than when only sodium was present. Volatility of sulfate was minimized by keeping the ratio [chemical equivalents (alkali + alkaline earths)/chemical equivalents (PO_3^{3-} + SO_4^{2-} + other nonvolatile anions)] $\geqq 1.1$ to form a "neutral" glass. Leaching rates of ^{137}Cs from two sulfate-containing ceramics corresponded to penetration rates of about 10^{-4} g cm^{-2} day^{-1}. The thermal conductivity of a true "glass" product was 1.3 to 1.7 Btu hr^{-1} ft^{-1} °F^{-1} vs 0.2 to 0.7 Btu hr^{-1} ft^{-1} °F^{-1}, for a calcined product. Vapor-liquid equilibria and densities were measured for sulfate-containing waste solutions. Titanium was the most resistant material tested for use in the evaporator, condenser, and off-gas lines. The maximum rate observed was 3.3 mils/month; expected rates are $\ll 1$ mil/month. Stainless steel and nickel alloy suffered severe intergranular attack under the most aggresive conditions. Wastes of low acidity (i.e. "acid killed") wastes were relatively noncorrosive ($\leqq 0.3$ mil/month) toward stainless steel and nickel alloys and even less corrosive toward titanium. Stainless steel is sufficiently resistant to serve as the fixation-storage vessel for the RL-Potglass process in spite of relatively high corrosion rates. Stainless steel is also suitable to contain waste for final storage in salt. (auth)

746 (ORNL-TM-882) FORMATION OF LOW LEACHABLE GLASSES FROM RADIOACTIVE WASTES USING THE RISING LEVEL TECHNIQUE IN STANDARD

POT CALCINATION EQUIPMENT. C. W. Hancher and J. C. Suddath (Oak Ridge National Lab., Tenn.). Sept. 1964. Contract W-7405-eng-26. 39p.

There is a safety advantage in handling radioactive waste as a massive glass-like solid of low leachability rather than a calcined solid which tends to crumble easily and has a higher solubility in ground water. Tests were made to see if a glass-like product could be made in standard pot calcination equipment. These tests were successful. Using the rising level technique, the feed rate to the calciner vessel is controlled so that only 1 ft of aqueous liquid is held up at any one time. The liquid holdup is limited because of the layered nature of the system. If the calcined solid which acts as a separation barrier between the molten glass on the bottom and the aqueous liquid should fracture allowing the aqueous liquid and melt to contact each other, there would be a vigorous steam reaction which could cause an instantaneous high pressure surge. Three tests using this rising level technique were completed. Two tests were made using TBP-25 waste to which monosodium phosphate and lead nitrate were added to form a glass. One test was made using FTW-65 X 3 Purex waste to which lithium hydroxide and aluminum nitrate were added to form a meltable mixture and calcium nitrate was added to reduce sulfate volatility. In an 8 in. dia × 80 in. long calciner pot the average feed rates were 5 to 8 liter/hr for the two types of waste fed. There was some difficulty with solids collecting in the off-gas lines. The solids were easily removed by frequent flushing of the off-gas line with dilute nitric acid or water. The rising level technique in standard pot calcination equipment is a feasible procedure for solidification of radioactive wastes into low leachable glass-like product. (auth)

747 (ORNL-TM-1094) CHEMICAL TECHNOLOGY DIVISION, UNIT OPERATIONS SECTION MONTHLY PROGRESS REPORT, FEBRUARY 1965. M. E. Whatley, P. A. Haas, R. W. Horton, A. D. Ryon, J. C. Suddath, and C. D. Watson (Oak Ridge National Lab., Tenn.). May 1965. Contract W-7405-eng-26. 30p. Dep.(mn); $2.00(cy), 1(mn) CFSTI.

748 (TID-7699(p.24-35)) STATUS OF PHOSPHATE GLASS PROCESS FOR FIXATION OF HIGH-LEVEL RADIOACTIVE WASTE. L. C. Emma, E. J. Tuthill, and G. G Weth (Brookhaven National Lab., Upton, N. Y.).

Several extended runs of a pilot plant for the phosphate glass fixation of wastes were carried out. Operating data and chemical data from the runs are shown. The break in the freeze valve tube which occurred during one run is discussed. Instrumentation of the pilot plant is discussed. Bench-scale equipment to be used in small hot-cell experiments on Purex waste at Hanford, and the experiments themselves, are described. Results are given of studies on the corrosion of Pt, Pt–Rh, Pt–Au, and Pt–Ir by phosphate glass for 3000 hr at 1200°C. (D.L.C.)

749 (TID-21882) INVESTIGATION OF MATERIALS AND METHODS FOR FIXATION OF LOW AND MEDIUM LEVEL RADIOACTIVE WASTE IN STABLE SOLID MEDIA. Progress Report, August 1, 1964–January 31, 1965. R. C. Rastogi, J. D. Sehgal, and K. T. Thomas (Atomic Energy Establishment, Trombay (India)). Mar. 29, 1965. Contracts AT(30-1)-3431 and IAEA US/261. 34p. Dep. (mn); $2.00(cy), 1(mn) CFSTI.

A proposal is described, for fixation of spent vermiculite, incinerator ashes, and chemical sludges in vitreous matrices. Results on the fusion studies of the mineral vermiculite, the leaching characteristics of the fused mass, and the volume reduction ratios are given. The apparatus for studying volatilization loss is described. (auth)

750 (TID–22531) INVESTIGATION OF MATERIALS AND METHODS FOR FIXATION OF LOW AND MEDIUM LEVEL RADIOACTIVE WASTE IN STABLE SOLID MEDIA. Progress Report, February 1–October 31, 1965. Rastogi, R. C.; Sehgal, J. D.; Thomas, K. T. (Atomic Energy Establishment, Trombay (India)). Dec. 4, 1965. Contract No. 261/US. 48p. Dep. mn. CFSTI $2.00 cy, $0.50 mn.

The results of fusion studies on incinerator ash from the pilot incinerator and synthetic chemical treatment plant sludge obtained as secondary wastes in the treatment of radioactive liquid effluents are described. The leaching characteristic of the fused mass and the volume reduction obtainable were studied. A method for evaluating loss on volatilization during firing, using stable cesium salt and flame photometry for the estimation of volatilized cesium, is described. Results are presented. Details of a pilot furnace under fabrication, proposed to be used for preparing 20–50 kg batches of glass using inactive wastes, and suggestions for future work are also included.

751 ON THE RADIATION STABILITY OF VITREOUS RADIOACTIVE PREPARATIONS. F. S. Dukhovich and V. V. Kulichenko. At. Energ. (USSR), 18: 361-7(Apr. 1965). (In Russian)

Data are given on the influence of ionizing radiation on the chemical stability of vitreous materials used for radioactive waste fixation. It is shown that irradiation in air caused chemical changes on the surface of specimens, due to the heterogeneous radiochemical reaction of solids with air components. The products of reactions formed carbonates, hydroxides, and nitrates. (tr-auth)

752 THE ENCAPSULATION OF HIGH-LEVEL NUCLEAR WASTES. R. N. Rickles (Dorr-Oliver, Inc., Stamford, Conn.). Atomics, 17: 50-2(Mar.-Apr. 1964).

A variety of liquid-to-solid conversion techniques presently used in waste disposal are reviewed. Encapsulation by formation of glass directly from the liquid product and by preparing a metal matrix containing solid waste are discussed. 17 references. (R.E.U.)

753 WASTE SOLIDIFICATION PROTOTYPE PROGRAM. Platt, A. M.; Cooley, C. R. (Battelle-Northwest, Richland, Wash.). Atomics, 18: 22-5(July-Aug. 1965).

A spray solidification technique is described, which employs a spray calciner coupled to a continuous melter to convert high-level radioactive waste to a chemically inactive solid for storage. A pot calcination technique is described, in which a batch of liquid waste is added to a heated stainless steel pot until the waste becomes a thick sludge or salt cake. At this time the feed is stopped and heating continued until the waste is converted to a solid calcine at about 900°C. In the continuous phosphate glass process the liquid waste is mixed with phosphoric acid and concentrated to a thick slurry in an evaporator. The slurry is fed into a platinum melter where glass is formed. The molten glass flows into a receiver where it is solidified for storage. (M.C.G.)

754 FINGAL'S CAVE FOR FIXATION OF RADIOACTIVE WASTES IN GLASS. Brit. Chem. Eng., 8: 555(Aug. 1963).

Disposal of fission product wastes by fixation in glass is discussed as an alternative to the practice of evaporating the fission products followed by underground storage in stainless-steel tanks. A large experimental apparatus was set up for this purpose. The project was named FINGAL (Fixation in Glass of Active Liquors) and the hot cell, in

which the experimental work will be carried out, was dubbed "Fingal's Cave." In the process a stream of the fission product waste solution in nitric acid is mixed with a stream of slurried borax and silica and the slurry is pumped into the top of a steel cylinder 5 ft long and 6 in. diameter. The cylinder is situated in a segmented furnace, which enables the input of heat to be regulated along the length of the cylinder. The oxides fuse to glass. Thus, glass is continuously formed in the vessel and when the level has risen to about 40 in. the operation is stopped and the glass is allowed to cool. The stream and nitric acid vapors produced leave the vessel and are conducted through a second cylindrical vessel identical with the first (except that this vessel contains a filter and absorber for dust and volatile ruthenium) to a condenser and scrubber system. When glass making is complete in the first vessel, it is cooled and then removed from the furnace to storage. The second vessel is now moved into the furnace and a clean vessel with filter installed takes its place. A third vessel, containing an extra filter and absorber, is placed in series with the first two vessels, but since its capacity is large it is not moved through the process positions.

755 BEHAVIOR OF STRONTIUM AND CESIUM DURING FIXATION OF RADIOACTIVE WASTES IN BASALT. Saidl, J.; Ralkova, J. (Inst. for Nuclear Research, Rez, Czechoslovakia). Collect. Czech. Chem. Commun., 31: 871-80(Feb. 1966). (In Czech).

Molten basalt possesses very good properties as a suitable material for the fixation of highly active wastes. A special advantage with respect to the usual glasses lies in the fact that its properties, especially its mechanical properties and chemical resistivity, can be improved by recrystallization. The extractability of ^{90}Sr and ^{137}Cs fused in basalt was followed. The values for the apparent diffusion coefficients were found in the region from 10^{-15} to 10^{-17} cm^2/sec at temperatures from 30 to 70°C.

756 TREATMENT OF FISSION PRODUCT SOLUTIONS BY VITRIFICATION. [R.] Bonniaud, Brodsky, Burko, and [P.] Cohen. Energie Nucl., 5: 427-30(Sept.-Oct. 1963). (In French)

The treatment of irradiated atomic reactor fuels produces residues of very high specific activity, which makes their disposal into the environment impossible. Prolonged storage in liquid form is inconvenient. Of the different possible solidification methods, one of the most promising is the incorporation, after burning, into a homogeneous vitreous mass. Laboratory research on active solutions leads to the building of a continuous pilot plant, which is described. (auth)

757 RADIOISOTOPE INCORPORATION INTO FUSED SILICATE MELTS. (THE INVESTIGATION OF Sr AND Cs ELUTION AND DIFFUSION. Jarmila Rálková and Jaroslav Saidl (Inst. of Nuclear Research, Czechoslovak Academy of Sciences, Rez). Jaderna Energie, 10: 14-18(Jan. 1964). (In Czech.)

The applicability of various materials for the fixation of high level wastes was investigated. Fused basalts or usual glasses were found to be the most convenient. Radioisotope elution and diffusion in these materials was investigated. The elution mechanism of silicate glasses is discussed and precautions for safe high-level waste storage are suggested. (auth)

758 DISPOSAL OF HIGH-LEVEL RADIOACTIVE WASTES. Rolf Eliassen (Stanford Univ., Calif.). J. Water Pollution Control Federation, 36: 201-16(Feb. 1964).

The present temporary and unsatisfactory disposal methods for nuclear wastes, and the need for a more reliable method at a reasonable cost are discussed. The failing points and inefficiencies of current methods of disposal are cited. Results are given of a recent study to incorporate the wastes into glass at high temperatures. An analysis of the problems that were confronted in the study and in the solutions is presented. As a result of the study, glass can be produced that will provide the necessary degree of security needed in high-level radioactive waste disposal by a method that can cope with the anticipated growth of the nuclear power industry. (auth)

759 "FIXING" RADIOACTIVE WASTE IN GLASS. K. D. B. Johnson (Atomic Energy Research Establishment, Harwell, Berks, Eng.). New Scientist, 23: 632-4(Sept. 10, 1964).

Experiments on the incorporation of radioactive waste in glass are reported. Radioactive waste in solution, and a slurry of glass-forming materials in water, flow into a vessel made of high alloy steel. The vessel is heated by a furnace at a temperature of 1050°C. Evaporation, conversion of nitrates to oxides, sintering of oxides and glass formation occur steadily during a period in which the vessel becomes filled with glass. The design of the pilot plant called FINGAL (Fixation In Glass of Active Liquor) is given. (P.C.H.)

760 FIXATION OF HIGH-LEVEL RADIOACTIVE WASTE IN PHOSPHATE GLASS. Upson, U. L. (Battelle-Northwest, Richland, Washington); Weth, George. Trans. Amer. Nucl. Soc., 8: 336-7(Nov. 1965).

761 GLASS IN NUCLEAR TECHNOLOGY. Robredo, J. Verres Refractaires, 18: 308-18(July-Aug. 1964).

The contribution of the glass industry to nuclear technology is considered. Two questions are discussed in detail: elimination of radioactive wastes by incorporating them in vitreous materials and manufacture of fuel elements by incorporating fissile matter in vitreous systems. (French Technical Press, No. 1, 1965)

In-Tank Solidification

See also the following references:

19, 24, 78, 133, 188, 191, 195, 246, 249, 703, 708, 875

762 (BNWL-101) TESTING OF IMMERSION HEATERS FOR IN-TANK SOLIDIFICATION CIRCULATORS. Dunn, J. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). July 1965. Contract AT(45-1)-1830. 15p. Dep. mn; CFSTI $1.00 cy, $0.50 mn.

The use of direct immersion electric heaters is proposed for concentrating intermediate level wastes in underground tanks. A series of tests was completed to investigate the major technical concern with this concept (that of scale deposition on the heat transfer surfaces). Tests were performed with simulated coating wastes, concentrated up to four times the original volume, and containing either 0 or 1% added sodium sulfate or phosphate. Power density of the heaters ranged from 17 to 500 w/in.^{2}. No scale deposition on heated surfaces occurred in either bench- or pilot-scale tests during any of the tests; maximum test duration was approximately 2600 hr. Some solids buildup occurred on heater supports in the vapor space above the heaters during early stages of the test program. This problem was overcome by relocating the reflux return line to provide a washing action, and the deposit proved readily soluble in

water. Periodic washes would be expected to minimize solids formation, which otherwise might occur with either the immersion heater or hot air system. Based on the heater performance and on the absence of scale on the heated surfaces, an electric heater prototype is recommended for testing under plant conditions. (auth)

763 (HW-83218) FIRST IN-TANK WASTE SOLIDIFICATION UNIT INFORMATION MANUAL. J. J. Shefcik, comp. (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Aug. 1, 1964. Contract AT(45-1)-1350. 47p. Dep.(mn); $2.00(cy), 1(mn) OTS.

A description is given of the immobilization of high-level waste solutions containing low fission product concentrations by solidification within the underground storage tanks. The process operation, safety, and control are discussed. The following equipment is described: tank, air compressor, heater, circulator, deentrainer, filter, condenser, exhauster, pressure-vacuum relief, crib, instrumentation, sampling station, and piping. (D.L.C.)

764 (ISO-SA-1) OPERATING EXPERIENCE ON IN-TANK SOLIDIFICATION OF RADIOCHEMICAL WASTES. Shefcik, J. J. (Isochem Inc., Richland, Wash.). [nd]. Contract AT(45-1)-1851. 19p. (CONF-660208-8). Dep. mn. CFSTI $1.00 cy, $0.50 mn.

From Symposium on Solidification and Long-Term Storage of Highly Radioactive Wastes. Richland, Wash.

Operation of the First In-Tank Solidification Unit is described. Radiochemical waste solutions are evaporated within a 750,000-gallon underground tank by sparging with air electrically heated to 1200°F. The air sparger is also an air-lift circulator used for agitation. To minimize entrainment, most of the heat is transferred to the solution by passing the air down the hollow wall of the circulator-sparger. Stud fins improve heat transfer. The off-gas is filtered and dehumidified, producing effluent streams low in residual radioactivity. Decontamination factors of $>10^4$ are obtained. Radiochemical wastes are added as evaporation occurs. After concentration by a factor of 4 to 6, the concentrate will solidify on cooling.

765 (RL-SA-41) OPERATING EXPERIENCE ON IN-TANK SOLIDIFICATION OF RADIOCHEMICAL WASTES. Shefcik, J. J. (General Electric Co., Richland, Wash. Hanford Atomic Products Operation). Aug. 17, 1965. Contract AT(45-1)-1350. 18p. (CONF-650902-10). Dep. mn; CFSTI $1.00 cy, $0.50 mn.

From American Chemical Society, 150th National Meeting, Atlantic City.

The initial operation of the First In-Tank Solidification Unit is described. Alkaline radiochemical waste solutions are evaporated within a 750,000-gallon underground tank by sparging with compressed air electrically heated to 1200°F. The air sparger also serves as an air-lift circulator to agitate the solution and as a heat exchanger. To avoid the high entrainment normally associated with hot air sparging, more than half of the available heat is transferred to the solution by passing the hot air down the hollow wall of the 40-inch diameter, 12-foot long, circulator-sparger. Stud fins are used to improve the heat transfer coefficient. The off-gas vapors are filtered and dehumidified, producing condensate and effluent air streams very low in residual radioactivity. Solution decontamination factors of greater than 10^5 are obtained. As evaporation occurs, additional radiochemical waste solutions are added. After wastes have been boiled down by a factor of 4 to 6, the concentrate will be allowed to cool and solidify. (auth)

766 (RL-SEP-65) CHEMICAL PROCESSING DEPARTMENT HAZARDS EVALUATION. IN-TANK WASTE SOLIDIFICATION PROJECT CAC-965. G. E. Backman, E. M. Johnston, H. C. Rathvon, L. V. Roddy, and M. F. Wiitala (Hanford Atomic Products Operation, Richland, Wash.). Nov. 24, 1964. Contract AT(45-1)-1350. 27p. Dep.(mn); $2.00(cy), 1(mn) CFSTI.

The in-tank solidification unit, installed in an underground storage tank, is designed to concentrate high-salt wastes by hot-air sparges. A study was made of safety considerations in the process, design, and operating plans It is concluded that the unit is basically adequate for its intended use. Some additional measures for the control o possible hazards are recommended. (D.L.C.)

Lime-Soda Softening Process

See also the following references:

19, 24, 31, 34, 87, 135, 511

767 SOFTENING AND DECONTAMINATING WASTE WATER BY CAUSTIC-CARBONATE PRECIPITANTS IN A SLOWLY-STIRRED SLUDGE-BLANKET COLUMN. Schonfeld, E.; Davis, W. Jr. (Oak Ridge National Lab., Tenn.). Health Phys., 12: 407-15(Mar. 1966).

In order to reduce the concentration of calcium in low-radioactivity-level process waste water and thereby make the water more suitable as a feed to a countercurrent foam column in which strontium and other radioactive nuclides can be removed, a modified lime-soda precipitation process was developed. The process consists in adding sodium carbonate and sodium hydroxide to the water to a concentration of 0.005 M each (calculated before chemical reaction) and using a slowly stirred, suspended-bed sludge column to remove (by a filtering action) the resulting calcium carbonate and magnesium hydroxide. With this procedure, and with Fe^{3+} (a few parts per million) as a coagulant, the residual hardness of the water was reduced at 5 ppm (as $CaCO_3$) or less. This low value was achieved with sludge columns of diameter up to 9 in., at water flow rates up to 60 gal/ft^2/hr, at sludge-height to column-diameter values of 4 to 0.5, and at stirring-paddle peripheral rates of 5 to 10 in./sec. With ORNL low-radioactivity-level process waste water as feed, the radionuclide decontamination factors across the sludge column were as follows: 10 to 15 for ^{90}Sr, 30 to 40 for ^{144}Ce, 10 to 50 for ^{60}Co, 1.7 to 2.2 for ^{106}Ru and 1.1 for ^{137}Cs. The calcium concentration can be reduced to a point such that it will not significantly interfere with subsequent removal of strontium in a foam separation column. The degrees of softening and decontamination are significantly better than can be achieved in the ORNL lime-soda plant. The process is being evaluated on a 15,000 gal/day basis.

Melt-Freeze Process

See also the following references:

34, 197, 198, 199, 200, 201, 203, 247, 691, 938

768 DEWATERING PROPERTY OF A RADIOACTIVE SLUDGE CONCENTRATOR. Tetsuo Mamuro, Toshio Ishiyama, and Takashi Matsumura. Ann. Rept. Radiation Center Osaka Prefect., 4: 36-41(1963).

The dewatering property of a small radioactive sludge concentrator was investigated. The plant adopts the freezing treatment for concentrating the radioactive chemical sludges that are produced in the treatment of radioactive waste waters through flocculation. The plant can handle 125 l of sludge/day and can reduce the final volume of the con-

centrated sludge to about $1/5000$, $1/2000$, and $1/4000$ of the volume of the original waste water to be decontaminated, for bentonite-Konan Floc sludge, calcium phosphate sludge, and ferric hydroxide sludge, respectively, in the case that separation by settling is applied for the treatment after thawing. In previous beaker experiments it was found that the addition of some polymer as agent is effective in increasing the volume reduction ratio or in shortening the settling time after thawing. This finding was applied to the large-scale treatment, and good results were obtained. (P.C.H.)

769 VOLUME REDUCTION OF RADIOACTIVE SLUDGES BY FREEZING AND THAWING TREATMENT. Tetsuo Mamuro, Toshio Ishiyama, and Takashi Matsumura (Osaka Radiation Center). Nippon Genshiryoku Gakkaishi, 5: 925-32(Nov. 1963). (In Japanese)

A small plant was constructed that adopts the freezing treatment for concentrating radioactive sludges. The volume reduction characteristic of the plant was investigated. The plant can handle 125 l of sludge per day and can reduce the final volume of the concentrated sludge to about 1/5000, 1/2000 and 1/4000 of the volume of the original waste water to be decontaminated, for bentonite-Konan Floc, calcium phosphate, and ferric hydroxide sludges, respectively, in the case that the separation by settling is applied as the treatment after thawing. Appreciably higher volume reduction ratio can be obtained when the vacuum filtration is applied after thawing. In the beaker-experiments that were carried out, it was found that the addition of some polymer as agent is effective in improving the volume reduction or in shortening the settling time after thawing. This was applied for the large scale treatment by the plant and good results were obtained. (P.C.H.)

Precipitation, Coprecipitation, and Flocculation

See also the following references:

12, 14, 19, 20, 24, 25, 34, 43, 72, 76, 79, 81, 82, 83, 85, 87, 89, 94, 99, 102, 103, 106, 108, 121, 141, 146, 149, 154, 162, 182, 205, 206, 209, 220, 233, 235, 246, 247, 263, 510, 511, 512, 515, 942, 945, 946, 947, 953, 964, 965

770 (AERE-R-4493) THE EXTRACTION AND PURIFICATION OF STRONTIUM FROM FISSION-PRODUCT SOLUTIONS. H. J. de Nordwall and G. J. Gittens (United Kingdom Atomic Energy Authority. Research Group. Atomic Energy Research Establishment, Harwell, Berks, England). Jan. 1964. 26p.

An appraisal of the conditions required for the precipitation of alkaline earth nitrates from fission product solutions was made. The theoretical predictions were experimentally confirmed. A scheme involving precipitation at 55°C from 16N nitric acid and leaching of the strontium from the precipitate with 5N nitric acid at 35° is suggested. The product is a mixture of strontium and barium nitrates essentially free from other elements. (auth)

771 (CEA-tr-A-1462) DONNEES FONDAMENTALES RECUEILLIES AU LABORATOIRE EN VUE DE LA CONSTRUCTION DE L'INSTALLATION DE TRAITEMENT DES DECHETS AU LABORATOIRE DE HAUTE ENERGIE DE L'E.I.R. A WURENLINGEN. I. TRAITEMENT DES DECHETS DE FAIBLE ACTIVITE. (Fundamental Data Collected at the Laboratory with the Object of Construction of an Installation for Treatment of Wastes at the E.I.R. High-Energy Laboratory at Würenlingen. I. Treatment of Wastes of Low Activity). Gunten. Translated into French from Neue Tech., 4: 345-403(1962). 31p.

The decontamination of slightly radioactive effluents was studied. Flocculation by $Na_3PO_4-FeCl_2$ at a pH of 11.5, followed by ion exchange on vermiculite, gave decontamination coefficients of 300 to 500. The vermiculite in its natural state, however, appears unsuitable for use in the installation under construction. Vermiculite substituted with sodium gave the best results. (D.C.W.)

772 (DP-877) RECOVERY OF FISSION PRODUCTS FROM PUREX WASTE. William E. Prout and E. Kenneth Dukes (Du Pont de Nemours (E. I.) & Co. Savannah River Lab., Aiken, S. C.). Feb. 1964. Contract AT(07-2)-1. 12p.

Laboratory studies were made to determine the feasibility of simultaneously recovering cesium, strontium, and rare earth fission products from Purex wastes by precipitation techniques. In the primary separation, rare earth-sodium sulfates, strontium sulfate (with lead carrier), and cesium molybdophosphate were precipitated. Cesium was separated satisfactorily as a crude fraction from the primary product; however, suitable fractions of strontium and rare earths were not obtained by the procedures that were tested. (auth)

773 (EUR-2535.f) DECONTAMINATION EN ^{137}Cs DES SOLUTIONS CONCENTREES DE PRODUITS DE FISSION. (^{137}Cs Decontamination of Concentrated Fission Product Solutions). Aubertin, Ch.; Lefevre, I.; Raggenbass, A. (Commissariat a l'Energie Atomique, Saclay (France). Centre d'Etudes Nucleaires). 1965. 52p. Dep. mn.

In order to obtain a ^{137}Cs decontamination factor of at least 10^4 for concentrated fission product solutions, two separation techniques were examined in detail. These are: phosphotungstate precipitation by addition of phosphotungstic acid; and Cs fixation by exchange on a preformed insoluble ammonium phosphotungstate. For the conditions studied, the most promising solution was found to be double precipitation by phosphotungstic acid using as carrier the ammonium already present in the concentrated fission product solutions.

774 (HW-81092) PUREX PLANT TEST OF THE CESIUM PHOSPHOTUNGSTATE PROCESS. S. J. Beard and W. C. Schmidt (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Mar. 16, 1964. Contract AT(45-1)-1350. 6p. Dep.(mn); $1.00(cy), 1(mn) OTS.

Approximately 98 percent of the cesium in Purex IWW was removed by a single phosphotungstate precipitation. Decontamination factors from the other isotopes could not be determined due to cross-contamination between feed and product streams. Chemical costs for this process were approximately $22.00 per ton of uranium processed to produce the IWW. (auth)

775 (IDO-12036) REMOVAL OF COBALT AND CHROMIUM BY PRECIPITATION AND ION EXCHANGE ON SOIL, LIGNITE AND CLINOPTILOLITE FROM CITRATE-CONTAINING RADIOACTIVE LIQUID WASTE.

D. B. Hawkins (Idaho Operations Office. Health and Safety Div., AEC). June 1964. 24p.

Laboratory studies indicate that ^{60}Co and ^{51}Cr can be removed from separate waste streams containing ammonium citrate and alkaline permanganate by a combined process of precipitation and ion exchange. The general procedure used was to neutralize the alkaline permanganate with sulfuric acid and to combine this solution with the ammonium-citrate solution causing the oxidation of the citrate. The Co and Cr originally present as complex anions were converted to cationic form by this process and scavenged from solution by the manganese-dioxide precipitate formed by reduction of the permanganate. The Co and Cr remaining in solution were then removed by ion exchange on lignite, soil, and clinoptilolite. Distribution coefficients for Co and Cr were 56 and 1.2, 800 and 36, and 24 and 0.1 for soil, lignite, and clinoptilolite respectively. Lignite used in combination with the oxidation-precipitation step resulted in the removal of 99.7% of the ^{60}Co and 95.1% of the ^{51}Cr from the waste streams. (auth)

776 (ORNL-3811) LABORATORY DEMONSTRATION OF THE TWO-STEP PROCESS FOR DECONTAMINATING LOW-RADIOACTIVITY-LEVEL PROCESS WASTE WATER BY SCAVENGING-PRECIPITATION AND FOAM SEPARATION. Davis, W. Jr.; Kibbey, A. H.; Schonfeld, E. (Oak Ridge National Lab., Tenn.). Oct. 1965. Contract W-7405-eng-26. 32p. Dep. mn; CFSTI $2.00 cy, $0.50 mn.

The two-step process for decontaminating low-level radioactive waste water was demonstrated in two continuous runs on a laboratory scale. The process consists in first precipitating and eliminating most of the suspended solids and soluble hardness in a suspended-bed sludge column and then producing further decontamination in a countercurrent foam column. Overall decontamination factors for Sr, Co, Ru, and Ce were $>3.7 \times 10^3$, 2.3 to 4, 2 to 5, and 50 to 180, respectively. The cesium decontamination factor was 20 when 60 ppm of grundite clay (baked at 600°C for 20 min) was added to the low-level waste (LLW) during the precipitation step. The ratio of surface to liquid flow rate is the crucial factor governing satisfactory strontium decontamination in the foam column. The metal ion decontamination factor in the foam-separation step is proportional to the metal ion distribution coefficient and inversely proportional to the volume reduction and liquid throughput. The cost of chemicals for this process is 26.6¢ per 1000 gal of LLW. Development of the process flowsheet required the study of several problems or variables. These included: (1) the effects of pH on precipitation of hardness ions; (2) the prevention of inhibitory effects of phosphates present in the waste water on the precipitation of calcium carbonate; (3) study of the sludge density as a function of liquid flow rate and bed depth; (4) the degree of biodegradability and fast quantitative analysis of surfactants; (5) the effect of splitting the feed of surfactant into the foam column; (6) volume reduction; and (7) recovery of surfactant.

777 (ORNL-3863) PILOT PLANT DEMONSTRATION OF THE DECONTAMINATION OF LOW-LEVEL PROCESS WASTES BY A RECYCLE SCAVENGING-PRECIPITATION ION-EXCHANGE PROCESS. King, L. J.; Ichikawa, M. (Oak Ridge National Lab., Tenn.). Dec. 1965. Contract W-7405-eng-26. 39p. Dep. mn. CFSTI $2.00 cy, $0.50 mn.

It was demonstrated that low-radioactivity-level waste (LLW) can be treated by an improved scavenging-precipitation ion-exchange process to reduce strontium and cesium activity levels, by factors of about 1000 at a cost of 74.6¢ per 1000 gal in a 750,000-gal-per-day plant. The unit operations in the improved process are: passage through an alumina column; scavenging-precipitation; and ion exchange using CS-100 phenolic-carboxylic cation exchange resin. Process improvements included: an alumina column to remove phosphate; a high-capacity, 4-ft-dia agitated clarifier with a processing rate of 47 gal $hr^{-1}ft^{-2}$, contrasted to 27 for the older unit; and a recycle system. In this process, the strontium, cesium (sorbed on the clay), and other activities are removed from the system in a single stream, the clarifier sludge. The major part of the reduction in the cost of treatment is due to this modification. Overall decontamination factors for the modified system averaged 17, 11, 1200, 1400, 2.6, and 580, for gross beta, gross gamma, cesium, strontium, ruthenium, and rare earths, respectively, which reduced the concentrations of these radiation emitters to less than 2% of the 168-hr MPC_w for occupational exposure. Final treatment with an anion exchange column increased the ruthenium and cobalt decontamination factors to about 14 and 11, respectively.

778 (ORNL-TM-827) PROPOSED METHODS FOR THE REMOVAL OF MERCURY FROM RADIOACTIVE WASTE SOLUTIONS. W. E. Clark and C. L. Fitzgerald (Oak Ridge National Lab., Tenn.). Mar. 31, 1964. Contract W-7405-eng-26. 22p.

A method was developed for the removal of mercury from radioactive waste solutions prior to calcination-fixation. In this process a more active metal, such as copper or aluminum, is used to displace the mercury. A batch operation is possible in which the column is simply discarded upon breakthrough, or a continuous process in which the metal is added through the top of the column as it is needed. The amalgam formed would be removed at the bottom of the column. A vibrated column filled with copper shot appeared best for FTW-65 Purex waste, and, in principle, for all waste solutions with an acidity lower than about 1.0M. Aluminum may be used for column packing with Purex waste solutions containing up to 1.5M free acid but lower acidities are required for TBP-25 (aluminum) wastes. Anodization of the copper prolonged the effective life of the column, but it was considered questionable whether the increase warrants the added complexity of design. The number of equivalents of copper dissolved per equivalent of mercury removed ranged from about 16 to about 25.5. The presence of copper should cause no difficulty in the calcination-fixation of the treated waste solution. The composition of the copper amalgam formed was not determined, but the product was a thixotropic liquid, which flowed through a glass stopcock at the bottom of the column. This indicates a probable copper content of 6% or less by weight. (auth)

779 (ORNL-tr-916) TREATMENT OF LIQUID WASTE CONTAINING ^{51}Cr. Hauang, E. J.; Yang, T. S. Translated for Oak Ridge National Lab., Tenn., from Ho Tsu K'o Hsueh, 4: 135-7(Feb. 1964). 5p. Dep. mn; JCL $1.10 fs, $0.80 mf.

A simple precipitation method of treating chromium-51 liquid waste is described. The waste is adjusted to pH 8 with 1N HCl, and is precipitated by adding $Pb(NO_3)_2$ aqueous solution. After these treatments the radioactivity of the filtrate is about background.

780 (UJV-850/63) ISOLATION OF LONG-LIVING FISSION PRODUCTS FROM RADIOACTIVE WASTES. I. GROUP SEPARATION OF MIXTURE OF FISSION PRODUCTS. J. Krtil, Z. Kolařík, and V. Kouřím (Československá Akademie Věd. Ústav Jaderného Výzhumu, Rež). [1962]. 28p.

An investigation was made of sorption on ferric hydroxide as a method of separating fission product mixtures containing Fe, Al, and U. Precipitation of hydroxide was carried out at either medium or low pH, and Sr, Y, and/or Ce were then desorbed from the precipitate. The separations were satisfactory. (D.L.C.)

781 (UJV-1113/64) SUL'FAT BARIYA S ISKUSSTVENNO VVEDENNYMI DEFEKTAMI SUL'FATA KAL'TSIYA-SORBENT DLYA MIKROKOLICHESTV STRONTSIYA I RADIYA. (Barium Sulfate with Calcium Sulfate Impurities Acting as a Srobent for Microquantities of Strontium and Radium). L. Berak and Yan Myunikh (Ceskoslovenská Akademie Ved. Ustav Jaderného Výzkumu, Rez). [1961]. 19p.

A phase diagram was constructed for barium and calcium sulfates. It was found that strontium capture by barium sulfate can be increased by introducing calcium sulfate impurities. The structure of the barium sulfate is deformed, and during recrystallization the active sorption locations trap isomorphous cations forming mixed crystals. Such sorption is effective in solutions containing large quantities of calcium ions and can be applied for separating strontium from drinking water containing calcium. (R.V.J.)

782 (UJV-1489) ISOLATION OF TRACE AMOUNTS OF STRONTIUM BY THE COPRECIPITATION WITH BARIUM SULFATE. Kyrs, M.; Konecny, C.; Kadlecova, L. (Ustav Jaderneho Vyzkumu, Ceskoslovenska Akademie Ved, Rez). 1966. 11p. Dep. mn.

The coprecipitation of strontium with barium sulfate was investigated in dependence on the order of reagent addition, rate of precipitation, and amount of barium sulfate precipitate. The removal of strontium was found to be optimum (99%) when the total amount of sulfate ions was first added to the raw water and afterwards, the barium nitrate solution was added in small portions. Based on the results obtained, the mechanism of strontium coprecipitation is discussed.

783 ON THE DECONTAMINATION OF RADIOACTIVE WASTE WATER BY FERRIC HYDROXIDE FLOCCULATION. [PART] II. Takashi Matsumura, Toshio Ishiyama, and Tetsuo Mamuro. Ann. Rept. Radiation Center Osaka Prefect., 4: 24-7(1963).

The decontamination properties of ferric hydroxide flocculation at high concentration of foreign matter were investigated. It was found that the removal of Sr was effectively raised by the addition of equivalent or more amount of carbonate ion when the waste water to be decontaminated contained alkaline earth metal ions. No noticeable difference was found between the flocculations in a beaker and in a flocculator under the same flocculation conditions. The main role of a slurry circulation-type flocculator in the decontamination of radioactive waste water is to clarify the effluent by dynamical separation of solid from liquid. (P.C.H.)

784 HUMIC ACID AS COAGULATION AGENT FOR CALCIUM PHOSPHATE FLOCCULATION. Takashi Matsumura, Toshio Ishiyama, and Tesuo Mamuro. Ann. Rept. Radiation Center Osaka Prefect., 4: 28-30(1963).

The effectiveness of humic acid in supplementing phosphate flocculation was investigated. The removal of ^{137}Cs and ^{106}Ru-^{106}Rh was increased by the addition of humic acid; ^{137}Cs by about 25 to 40%. The removal of ^{89}Sr increased with decrease in the ratio $Na_3PO_4 \cdot 12H_2O/Ca(OH)_2$, reaching 95.1% at the ratio 1 when 200 ppm of humic acid was added. The removal of ^{95}Zr-^{95}Nb was increased by about 10% by the addition of humic acid, and ^{144}Ce-^{144}Pr was completely removed. However, the removals of various radioisotopes were somewhat lower in calcium humate flocculation than humic acid-calcium phosphate flocculation. (P.C.H.)

785 DECONTAMINATION OF RADIOACTIVE RUTHENIUM BY CELLULOSE DERIVATIVE-BENTONITE FLOCCULATION. [PART] I. Toshio Ishiyama, Takashi Matsumura, and Tetsuo Mamuro. Ann. Rept. Radiation Center Osaka Prefect., 4: 31-5(1963).

The possibility of using cellulose ion exchange as a flocculation agent was investigated. Two kinds of cellulose ion exchangers were synthesized from powdered cotton cellulose. They were used as flocculation agent for bentonite-polyelectrolyte flocculation and their effect on the decontamination of trivalent Ru was investigated. It was found that weak basic cellulose anion exchanger has a remarkable effect on the removal of trivalent Ru. The experimental method is outlined. (P.C.H.)

786 DECONTAMINATION OF RADIOACTIVE RUTHENIUM BY CELLULOSE DERIVATIVE BENTONITE FLOCCULATION. [PART] II. Ishiyama, Toshio; Matsumura, Takashi (Radiation Center of Osaka Prefecture). Ann. Rept. Radiation Center Osaka Prefect., 5: 37-42(Mar. 1965).

The decontamination properties of cellulose anion exchanger-bentonite flocculation were studied with respect to removal of fission product Ru. An efficient decontamination of fission product Ru through DEAE cellulose-bentonite flocculation can be achieved by treating the fission product solution with hydrochloric acid and sodium nitrite and changing nonionic nitrato complex of $^{106}RuNO^{3+}$ ion into $[^{106}RuNO(NO_2)_4OH]^{--}$. The formation of "organo-clay complex" through combination of DEAE cellulose and bentonite plays an important role in raising the removal of Ru, possibly because of the high adsorption capacity of "organo-clay complex" for $[^{106}RuNO(NO_2)_4OH]^{--}$. Tertiary amine cellulose is superior as the agent for bentonite flocculation to quaternary amine cellulose. (auth)

787 CLARIFYING AND PURIFYING OF RADIOACTIVE WASTE LIQUIDS BY THE FLOTATION METHOD. V. V. Pushkarev, Yu. V. Egorov, E. V. Tkachenko, and V. L. Zolotavin. At. Energ. (USSR), 16: 48-51(Jan. 1964). (In Russian)

Use of Al or Fe hydroxides for the clarification of radioactive waste effluents results in the formation of bulky deposits. Tests were carried out on a method for decreasing the volume of these deposits by carrying out the purification process with the addition of a flotation agent, which renders the surface of the solid particles hydrophobic. The system studied consisted of the hydroxides and surface-active and complexing agents, added to the radioactive waste solution. The moisture content of the flotated hydroxides and the efficiency of the removal of the radioactive components, such as ^{90}Sr, ^{90}Y, and ^{95}Nb and the hardness of the solution were determined as a function of the pH, the amount of the flotation agent, and the amount of the added hydroxide. The best rate was found to be 0.002 to 0.004 ml of sulfate soap flotation agent per mg of the hydroxide. The lowest moisture content of the flotated deposit (92.0 to 95.0%) was obtained at a pH between 6.0 and 7.0; the moisture content was not affected by the amount of the flotation agent. Comparison with settling tests revealed

that the flotation method yields deposits with lower moisture content and lower volume; the time needed for clarification was also considerably shorter. (TTT)

788 RECENT INVESTIGATIONS INTO TREATMENT OF RADIOACTIVE EFFLUENT. J. H. Clarke and T. D. Wright (Atomic Energy Research Establishment, Harwell, Berks, Eng.). Effluent Water Treat. J., 3: 276-9(May 1963).

The treatment of radioactive effluent has been studied intensively at Harwell, and the most suitable single stage chemical treatment consists of adding 50 mg/l Ca^{2+}, 80 mg/l PO_4^{3-}, and 40 mg/l Fe^{2+} at a pH value of 11.5. This is for an effluent containing a wide range of radioactive isotopes with no excessive concentration of any one of them. The treatment removes approximately 90% of the total β-γ emitters, the principle isotope not effectively removed being Cs^{137}. It is usual, where high decontamination is required, to remove the Cs^{137} by a final ion exchange stage. The ferrocyanides of Zn, Ni, Fe, and Cu, however, have been used in chemical processes for the extraction of Cs from concentrated atomic wastes. In this work the aim has been the recovery of Cs free from other fission products and precipitation has been under acidic conditions. This particular use of ferrocyanide for Cs precipitation has been applied in the treatment of radioactive effluent where Cs concentrations are far less than in the concentrated atomic waste. Its application, however, has been limited to neutral or low pH conditions which are not suitable for the removal of other radioactive species. This disadvantage has been overcome in the past by following a ferrocyanide precipitation with a phosphate or other treatments at a high pH value to remove the residual active ions. It was considered to be advantageous if the ferrocyanide and phosphate precipitation could be combined into a single stage treatment without losing the benefits of either. The development of this scheme and further increase in efficiency which can be obtained by incorporating ion exchange on vermiculite when necessary are discussed.

789 TREATMENT OF RADIOACTIVE WASTE. Hwang, Yin-Si; Huang, An-I (National Tsing Hua Univ., Hsinchu, Taiwan). Ho Tsu K'o Hsueh, 4: No. 3-4, 85-90 (Aug. 1965).

Attempts were made to investigate the feasibility of disposing radioactive liquid wastes from ^{131}I and ^{60}Co produced in the radioisotope laboratory of the National Tsing Hua University. In particular much effort was devoted to finding a simple method of disposal of the liquid waste resulting from the production of ^{131}I and the recovery of telluric acid. It was found that the pH adjustment of liquid waste as well as the addition of precipitant (sodium phosphate : calcium hydroxide = 1 : 0.6) is effective to decontaminate the radioactivity of the liquid waste. The nitric acid precipitation method and the alcohol precipitation method were tried for the recovery of telluric acid from the liquid waste. For the ^{60}Co liquid waste mixed reagent, sodium carbonate : calcium hydroxide = 1 : 1 was found to be effective.

790 TREATMENT OF LIQUID WASTE CONTAINING CHROMIUM-51. E. J. Huang and T. S. Yang (National Tsing Hua Univ., China). Ho Tsu K'o Hsueh, 4: 135-7 (Feb. 1964). (In Chinese)

A simple precipitation method for treating liquid wastes containing ^{51}Cr is described. The pH of the waste solution is adjusted to 8 with 1-N HCl, after which the Cr is precipitated by adding a $Pb(NO_3)_2$ solution. After treatment, the filtrate activity is near the background activity. (D.C.W.)

791 EFFECT OF A SYNTHETIC DETERGENT ON THE WATER TREATMENT OF RADIOACTIVE WASTES BY THE COAGULATION PRECIPITATION PROCESS. Ito, Fumio (National Inst. Radiol. Scis., Chiba, Japan). J. Water Waste (Tokyo), 6: 921-6(Dec. 1964). (In Japanese).

The effect of ABS-Na (sodium alkylbenzenesulfonate) on the water treatment of radioactive wastes was studied. Water (1000 cmp/ml) containing ^{89}Sr and 1% ABS-Na was used as the test solution. Two coagulants, $FeCl_3$ (300 ppm) and Na_3PO_4 (300 ppm + auxiliary coagulant $Ca(OH)_2$ 150 ppm), were used. The pH of the solution was varied between 10, 11, and 12. The synthetic detergent in the radioactive waste affected the water treatment by coagulation precipitation. When the ABS-Na concentration exceeded 240 ppm and pH was higher than 10, the decontamination efficiency increased considerably. The effect of ABS-Na on the coagulation precipitation became larger as the pH was lowered. In the case of Na_3PO_4, the decontamination efficiency was not varied considerably if the pH exceeded 9. (NSA of Japan)

792 THE PURIFICATION OF RADIOACTIVE WASTE WATERS BY CONTACT PRECIPITATION. PART I. THE PRINCIPLE OF CONTACT PRECIPITATION. I. Mittag and G. Sachse (Zentralinstitut für Kernforschung Rossendorf, Ger.). Kernenergie, 7: 339-44(May 1964). (In German)

In the purification method described calcium carbonate is precipitated onto a contact mass, thereby eliminating the need for filtration and dehydration of the precipitate. The operation is continuous; however, removal of the precipitates (approximately 5 millimeter particle diameter) must be accomplished batchwise. The purified water has a German degree of hardness of 1°; further purification can be accomplished by means of ion-exchange columns. The techniques and apparatus are described in detail and typical results are presented. (OTS)

793 ON THE PURIFICATION OF RADIOACTIVE WASTE WATERS BY CONTACT PRECIPITATION. PART 2. SEMITECHNICAL-SCALE EXPERIMENTS FOR PURIFYING LABORATORY RADIOACTIVE WASTE WATERS. G. Sachse and I. Mittag (Zentralinstitut für Kernforschung, Rossendorf, Ger.). Kernenergie, 7: 614-20(Aug. 1964). (In German)

Two procedures were proved to be suitable for radioactive waste water precipitation—a single-stage process and a two-stage one. For these processes, the effects of impurities of the radiochemical waste waters as complexing agents, detergents, and others on contact precipitation were studied. By means of a preliminary semitechnical-scale plant built in the Zentralinstitut für Kernforschung Rossendorf more than 100 m^3 of waste waters produced in the institute were purified without any prejudicial affecting of the contact mass adsorptive power. For carrying out the experiments waste water of about 10^{-7} $\mu C/ml$ specific activity was contaminated by fission products. Owing to the softening of waste water as a result of the contact precipitation, values of less than 1°dH of residual hardness were reached. As an experiment, further purification of the waste waters was carried out by means of ionites. (auth)

794 DECONTAMINATION OF RADIOACTIVE WASTE WATERS BY CONTACT PRECIPITATION. PART III. THE CALCULATION OF THE INVESTMENT AND OPERATING COSTS OF A PLANT USING CONTACT PRECIPITATION COMBINED WITH SUBSEQUENT ION EXCHANGE PURIFICATION. Sachse, G.; Haentzschel, H. (Zentralinstitut fuer Kernforschung, Rossendorf, Ger.). Kernenergie, 8: 697-704(Dec. 1965). (In German).

A calculation of the cost of construction and operation of a plant with a capacity of 10 m^3/h was carried out on the basis of semitechnical experiments of decontaminating ra-

dioactive waste waters by contact precipitation and ion exchange. The investment cost of a plant of the capacity mentioned above for the two variants in account amounts to about 290,000 and 320,000 M, respectively. The operating cost of contact precipitation amounts to 0.99 M/m^3 for the one-stage plant, and to 0.94 M/m^3 for the two-stage one. Two variants were taken into account for decontaminating the radioactive waste waters by ion exchange: using the ionite in the Na–Cl form or in the H–OH form. Using the ionite in the Na–Cl form leads to a total operating cost of 2.45 M/m^3, using it in the H–OH form to that of 2.86 M/m^3.

795 AN INVESTIGATION OF THE USABILITY OF SOME FLOCCULANTS FOR TREATMENT OF RADIOACTIVE WASTE WATER BY MINERAL SORBENTS. Oglaza, Jan; Siemaszko, Aleksander. Nukleonika, 10: 519-22(1965). (In Polish).

Investigations were carried out with the water of the Vistula River, having a hardness of 13° north. The sorbents used were raw bentonite of the "Jawar" diggings, bentonite loam from the Chmielnik region, and silicious earth from the Lechowek pit, and as flocculants, Separan 2610 (polyacrylamide), and swelling-type shavings NTL. The water was contaminated with ^{137}Cs and ^{144}Ce with carriers of CsCl and $Ce(NO_3)_3 \cdot 6H_2O$ in quantities of 0.5 mg/liter and an activity of 10 to 20 μC/liter. In the first stage of the investigation, the rate of growth of the clear column was measured, as observed visually, for the 3 sorbents with and without flocculants. Also, the activity was measured with a collimated radiation detector viewing only the center of the column. This stage showed that the shavings gave a large increase in settling out of the radioactivity, and the Separan gave a considerably larger increase. In the second stage, the amount of radioactivity found in the sediment was first measured as a function of the mass of sorbent added, with 2 mg/liter of Separan. Next, without flocculants, the rate of decontamination of the water was measured as a function of time after the mixing ceased. The addition of flocculants increased the rate of sedimentation and the rate of decontamination, and resulted in a great improvement in producing a very distinct boundary of the clear column of the fluid, making decanting much more feasible.

796 DECONTAMINATION OF RADIOACTIVELY POLLUTED LABORATORY WASTES. H. Koch (Institut für Angewandte Radioaktivitat, Leipzig). Symp. Tech. Zagadnien Ochrony Promieniowaniem, Warsaw, 456-67 (1962). (In German)

The decontamination of radioactive wastes is still in such a state of flux that the selection of the different decontamination methods depends on the given situation. Results obtained from coprecipitation reactions of a number of radioisotopes with various precipitation agents are discussed. The laboratory wastes with constantly changing radioisotope composition can be decontaminated by a single precipitation with phosphate or mixed phosphate-ferrocyanide. As research with waste processing installations have shown, in almost all cases a single precipitation will reduce the radioactivity of wastes below the permissible limits. The waste processing installations consist of numerous storage containers with 2.5 m^3 capacity that are connected with precipitation, filtering, and control devices. (tr-auth)

797 EXPERIENCE IN CONSTRUCTION AND MANAGEMENT OF A RESEARCH INSTALLATION FOR SEPARATION OF RADIOACTIVE WASTES AT THE INSTITUTE FOR APPLIED ISOTOPE RESEARCH, BERLIN-BUCH, DDR. D. Schoenberg (Institut für Angewandte Isotopenforschung, Berlin). Symp. Tech. Zagadnien Ochrony Promieniowaniem, Warsaw, 476-88(1962). (In German)

The constructional and technical aspects of the Institute for Applied Isotope Research are described. The flow scheme is explained. The accumulating quantities of wastes are discussed in connection with the function and dimensions of the installation. The precipitation and measurement methods used in waste processing are described, and the results are reported. The limits of applicability of such coprecipitation methods are discussed. (tr-auth)

798 POSSIBILITIES FOR IMPROVEMENT OF WATER-FREE PRECIPITATES IN PURIFICATION OF RADIOACTIVE WASTES BY THE WIRBOS PROCESS. G. Sachse (Zentralinstitut für Kernphysik, Rossendorf, Ger.). Symp. Tech. Zagadnien Ochrony Promieniowaniem, Warsaw, 609-23(1962). (In German)

In the purification of radioactive wastes by precipitation methods, great difficulties arise in the separation and further processing of radioactive precipitates containing large concentrations of water. In the Central Institute for Nuclear Physics at Rossendorf, the Wirbos method was investigated and further developed to make possible the precipitation of the hardening components of water as solid, water-free precipitates. In this method the bicarbonate hardening components of water were precipitated with $Ca(OH)_2$ as lime water. The precipitates were then separated on contact grains and held in the material by the flow-through water in the conical vessel in which the reaction was carried out. The contact mass adhered to contact spheres with a diameter of 5 to 7 mm and could easily be separated by draining of the liquid phase from the reaction vessel. On the basis of the investigations, it could be shown that not only the bicarbonate hardeners were removed, but it was possible to remove part of the permanent water hardeners. Preliminary results obtained with a pilot-plant installation working on the Wirbos principle are reported. (tr-auth)

799 THE DECONTAMINATION OF RADIOACTIVE WATER WITH CHEMICAL PRECIPITATION. H. Keese. Thesis, Brunswick, Technische Hochschule, 1963. 66p. $2.00(Gmelin, AED-DISS. 63-306). (In German)

Various precipitation methods, which can be used for effective decontamination of large quantities of weakly contaminated water, are studied. Decontamination factors for ^{144}Ce, ^{91}Y, ^{89}Sr, ^{90}Sr, ^{137}Cs, and ^{60}Co were determined. In some cases the adsorption isotherms are given. Results indicate that adsorption is linear up to activities of 90 μC/l. (Gmelin Inst.)

800 THE REMOVAL OF RADIONUCLIDES FROM WATER AND WASTE-WATER BY MANGANESE DIOXIDE [Thesis]. Touhill, Charles Joseph Jr. Troy, N. Y., Rensselaer Polytechnic Inst., 1964. 238p.

The extent and mechanism of removal of radionuclides from aqueous solution by hydrous manganese dioxide for a variety of process conditions were investigated. The extent of removal of radionuclides was correlated with pH and with one of the electrokinetic properties of solutions containing colloidal hydrous manganese dioxide. The property measured was the electrophoretic mobility of the colloidal particles. Cesium-137, iodine-131, cerium-144 (with the praseodymium-144 daughter), strontium-90, and yttrium-90 were the radionuclides whose adsorption was correlated with electrophoretic mobility and pH. Three chemical reactions for precipitating manganese dioxide were investigated to determine which would be suitable for use in a water or waste-water treatment process. The well-

known Guyard reaction and the reaction in which manganous ion is oxidized by oxygen in the presence of alkali produced satisfactory slurries of manganese dioxide. Reduction of permanganate with hydrogen peroxide was not satisfactory. The dependence of electrophoretic mobility on pH provided fundamental information about the nature of the manganese dioxide micelle and the extent of adsorption of anions, especially hydroxyl ions. A micelle model was developed from the correlation of electrophoretic mobility with pH. This model was extended using the results of radionuclide adsorption and was used to interpret the observed adsorption phenomena. Use of hydrous manganese dioxide in a treatment plant for removing radionuclides from water and waste-water would seem to be feasible.

Solvent Extraction

See also the following references:

14, 24, 34, 65, 69, 96, 133, 188, 218, 219, 228, 242, 244, 247, 483, 510, 944, 948, 951, 952, 953, 958, 960, 965, 966, 967

801 (BNWL-68) SOLVENT EXTRACTION OF CESIUM FROM HANFORD ALKALINE SUPERNATE BY 4-SEC-BUTYL-2(α-METHYLBENZYL)PHENOL. LABORATORY STUDIES. Bray, L. A. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). Aug. 1965. Contract AT(45-1)-1830. 18p. Dep. mn; CESTI $1.00 cy, $0.50 mn.

A cesium solvent extractant, 4-sec-butyl-2(α-methylbenzyl)phenol, was investigated for possible use at Hanford. The substituted phenol, diluted with a kerosene, was found to be a very selective extractant for cesium, both in the laboratory and, subsequently, by G. L. Richardson in pilot-plant-scale pulse column demonstrations. The extractant provides an efficient method for cesium recovery from the very large volumes of Alkaline Supernate now being stored at the separations facilities. The extraction of cesium depends upon the concentration of phenol, the pH of the aqueous waste solution, the temperature of the equilibration, and the aqueous Na^+ concentration. Approximately 0.3 of a mole of additional OH^- was required per liter of synthetic Alkaline Supernate (9M Na^+, pH 10.3) for maximum cesium extraction. Cesium distribution values are enhanced by an equilibration temperature decrease. The heat of extraction, $-\Delta H$, was found to be approximately 19 kcal. The cesium distribution ratio was also inversely dependent on the sodium concentration with a slope of slightly less than –2 over the range of 1 to 8M sodium. (auth)

802 (BNWL-187) PURIFICATION OF RARE EARTHS FROM HANFORD WASTES BY SOLVENT EXTRACTION. Bray, L. A.; Roberts, F. P. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). Feb. 1966. Contract AT(45-1)-1830. 31p. Dep. mn. CFSTI $2.00 cy, $0.50 mn.

Laboratory and hot-cell experiments showed that a good separation of Mn and the rare earths is possible by controlled pH solvent extraction. Because pH control is very difficult in the plant, unreliable results and high product losses occurred in the full-scale tests; the D2EHPA solvent extraction scheme for Mn removal proved impractical. An alternate scheme, based on sulfate precipitation was developed. This process, although complicated by the lack of a centrifuge in the plant, does not require pH control. Preliminary results indicated 85–95% rare earth recovery and an adequate Mn decontamination factor of 50–100. The separation of cerium from trivalent rare earths with D2EHPA was improved by the use of a straight chain hydrocarbon diluent. The effects of radiolysis were markedly reduced and high cerium distribution ratios were maintained.

803 (HW-72666) HOT SEMIWORKS STRONTIUM-90 RECOVERY PROGRAM. (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). July 17, 1963. Contract AT(45-1)-1350. 179p.

The laboratory studies, radioactively cold development studies, and radioactively hot development-recovery activities involved in the Hot Semiworks operations leading to the recovery and purification of nearly a megacurie of Sr^{90} are described. The D2EHPA extraction process was used for separation of Sr^{90} from Purex plant crude Sr^{90} solution. Four runs, each processing 210 to 480 kC of Sr^{90}, were made, with each lasting about 2 weeks. A total of 744 kC of purified Sr^{90} was isolated for shipment out of a total input of 1180 kC. (M.C.G.)

804 (HW-79762(Pt.1)) SOLVENT EXTRACTION OF STRONTIUM, CERIUM AND RARE EARTHS WITH D2EHPA. PART 1. LABORATORY STUDIES. W. W. Schulz (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Feb. 1964. Contract AT(45-1)-1350. 56p.

A solvent extraction process for separating strontium and rare earths from Purex process acid waste has been developed. The extractant is di-(2-ethylhexyl) phosphoric acid (D2EHPA) dissolved in a tributyl phosphate and hydrocarbon diluent. There are two product streams; one contains the strontium and the other contains the rare earths. (auth)

805 (HW-79762(Pt.2)) SOLVENT EXTRACTION OF STRONTIUM, CERIUM, AND RARE EARTHS WITH D2EHPA. PART 2. PILOT PLANT STUDIES. G. L. Richardson (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Feb. 1964. Contract AT (45-1)-1350. 48p.

A four-column D2EHPA process was developed for the simultaneous extraction of Sr and rare earths from Purex acid waste (PAW). The clarified PAW, with added citric or tartaric acid, is adjusted to pH 4.7 and contacted with 0.2 M D2EHPA–0.2 M TBP–hydrocarbon diluent. The solvent, containing only Sr, rare earths, Ca, and Na, has its Na stripped by formic acid, its Sr stripped by dilute HNO_3, and its rare earths and Ca stripped by more concentrated HNO_3. The spent solvent is then washed with a basic solution to remove Y and "irreversibly" extracted metals. Typical process performance is given. Temperature has a pronounced effect on Ce extraction in the first column. A total extraction concept was also tested with a stored alkaline sludge dissolved in HNO_3. The major limitation to this process was the difficulty of producing a stable feed at the desired pH without an expensive complexing agent. (D.L.C.)

806 (HW-80686) SOLVENT EXTRACTION OF CESIUM BY 4-SEC-BUTYL-2-(α-METHYLBENZYL) PHENOL (BAMBP): PILOT PLANT STUDIES. G. L. Richardson (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Feb. 3, 1964. Contract AT(45-1)-1350. 21p.

The extraction and recovery of cesium from a simulated Purex alkaline waste supernate was successfully demonstrated in the pilot plant pulse column using both a nozzle plate cartridge and Raschig ring packing. Stable volumetric capacities as high as 500 gph/ft^2, sum of flows, were obtained with the nozzle plate cartridge together with HTU's (overall raffinate-film basis) of 2 feet or less (waste

losses less than 2 percent using a 9 foot extraction section). Still higher capacities were demonstrated in the pulsed, packed column but with some loss in efficiency. The cesium product was decontaminated by factors of 300 to 1400 from sodium by the use of weak, inorganic acid scrubs (such as boric or phosphoric acid) and stripped from the solvent with dilute HNO_3. The HTU's in the stripping operation were generally greater than 2 feet (optimum waste losses of 2 to 3 percent typical). The sodium decontamination factors (DF's) and cesium stripping HTU's were considerably poorer than expected. The poor performance was eventually traced to degradation of the solvent through reaction with nitrite ion entrained with the feed into acidic portions of the purification cycle. The degradation did not appear to affect cesium extraction, but it greatly interfered with sodium and cesium stripping. Addition of sulfamate ion to the stripping solutions prevented the degradation reaction. A second series of runs was made (with fresh BAMBP) to test the ability of a BAMBP extraction process to purify a conceptual cesium crude. This crude, containing 0.05M Cs, 0.01M K, and 2M Na, was decontaminated an average of 8000-fold from sodium and 800-fold from potassium. The final product was concentrated by a factor of five. Typical HTU's for both extraction and stripping ranged from 1 to 2 feet (optimum waste losses less than 1 percent) at volume velocities as high as 900 gph/ft^2 in the 1A column and 550 gph/ft^2 in the 1C column. Water was satisfactorily used to scrub sodium from the solvent prior to cesium stripping; scrub DF's averaged five for sodium and greater than 10 for potassium. Dilute sulfamic acid was used to strip the cesium; however, nitric acid (containing a trace of sulfamic acid or hydrazine as insurance against the nitrite reaction) could be used. BAMBP was proved to be a highly satisfactory, selective solvent for cesium recovery and purification, but its sensitivity to nitrite degradation may limit its usefulness in high radiation fields where nitrite ion is continuously formed by nitrate radiolysis. Evaluation of this aspect is currently in progress. (auth)

807 (HW-SA-3211) SOLVENT EXTRACTION PROCESSES FOR RECOVERY OF RADIOACTIVE MATERIALS. G. L. Richardson (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Oct. 18, 1963. Contract AT(45-1)-1350. 15p. (CONF-196-1)

From American Institute of Chemical Engineers Northwest Regional Meeting, Portland, Oreg., Oct. 1963.

Four solvent extraction processes were developed for separating fission products from stored waste solutions: D2EHPA, DPA, BAMBP, and CSREX. The chemistry, flowsheets, and pilot plant performance of these processes are described briefly, and the pH dependency of the separations is pointed out. Either the D2EHPA or CSREX process will be used to process current and stored radioactive wastes for the Waste Management Program. (D.L.C.)

808 (ORNL-3830, pp 222-7) RECOVERY OF FISSION PRODUCTS BY SOLVENT EXTRACTION. (Oak Ridge National Lab., Tenn.).

Activities in a program to develop processes for large-scale recovery of fission products from reactor-fuel processing wastes are described. Information is included on Cs recovery, Ce and Pm separations, Ru extraction, and mixer-settler efficiencies.

809 (ORNL-P-1603) UNEXPECTED RATE EFFECTS IN THE EXTRACTION OF RARE EARTHS BY DI(2-ETHYLHEXYL)PHOSPHORIC ACID. Crouse, D. J.; Horner, D. E. (Oak Ridge National Lab., Tenn.). [1964]. Contract W-7405-eng-26. 16p. (CONF-650918-7). Dep. mn; CFSTI $1.00 cy, $0.50 mn.

From International Conference on the Chemistry of the Solvent Extraction of Metals, Harwell, Berks, Eng.

In extraction of fission product rare earths with di(2-ethylhexyl) phosphoric acid from tartrate- (or citrate-) complexed waste liquors (pH 5), the rates of extraction are decreased by the presence of certain metal ion contaminants, such as iron, aluminum, and particularly chromium. Apparently, these metal ions react with the rare earths ions and tartrate in the aqueous phase to form non-extractable complexes that are slow to dissociate. The magnitude of the effect is different with each interfering metal ion and also for the different rare earths. Increasing the tartrate concentration increases the rates of extraction. (auth)

METHODS OF DISPOSAL

Dilution

See also the following references:

10, 19, 23, 24, 30, 31, 32, 34, 35, 36, 80, 86, 87, 93, 106, 122, 137, 138, 145, 162, 163, 164, 174, 177, 182, 183, 184, 211, 212, 215, 217, 220, 222, 233, 237, 247, 253, 254, 259, 260, 262, 263, 264, 269, 270, 271, 278, 279, 280, 281, 283, 284, 288, 289, 291, 295, 296, 297, 298, 300, 302, 303, 304, 306, 307, 308, 309, 312, 313, 314, 317, 318, 319, 320, 323, 327, 329, 330, 333, 334, 335, 350, 484, 532, 615, 927

810 (A/CONF.28/P/287) NORTH AMERICAN EXPERIENCE IN THE RELEASE OF LOW-LEVEL WASTE TO RIVERS AND LAKES. Herbert M. Parker (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.), R. F. Foster, I. L. Ophel, F. L. Parker, and W. C. Reinig. 15p. (HW-SA-3421)

Prepared for the United Nations Third International Conference on the Peaceful Uses of Atomic Energy, 1964.

Experience in liquid-waste disposal is reported at the large atomic energy sites at Chalk River, Hanford, Oak Ridge, and Savannah River. Nuclear power plants are also considered. The fate of the radionuclides in the environment was studied, and it was determined that the total dose to the public from multiple environmental sources is well within limits. (D.L.C.)

811 (AE-182) THE HANDLING OF LIQUID WASTE AT THE RESEARCH STATION OF STUDSVIK, SWEDEN. S. Lindhe and P. Linder (Aktiebolaget Atomenergi, Studsvik (Sweden)). Mar. 1965. 17p. Dep.(mn).

The following quantities of radioactive waste are allowed to be released into a strait between the islands of Stora Bergö and Studsviksholme: total α-activity = 0.2 Curie/month and total β-activity = 36 Curie/month, of which cerium, yttrium, rare earths = 15 Curie/month and strontium = 2.4 Curie/month. Before the release the radioactive waste has to be collected and controlled. Quantities approaching or exceeding the disposal limits have to be removed and concentrated by evaporation. The liquid waste

is classified in several categories depending upon the level of activity: high active and medium active waste, low active waste, process water, sanitary water, surface water and reactor cooling water. The technical dimensioning of each category was based upon expected specific production figures (volume/man · month and activity/ man · month). These figures are based upon information obtained from Harwell. Actual production figures obtained during 1963 and the first half of 1964 are shown and compared with the expected ones. As a conclusion, it is stated that the actual production follows the predictions fairly well. (auth)

812 (AECL-2030) STUDIES OF DILUTION IN THE OTTAWA RIVER USING RHODAMINE B. 2. CRNL TO PEMBROKE. W. F. Merritt (Atomic Energy of Canada Ltd., Chalk River, Ont.). July 1964. 10p.

Rhodamine B, a fluorescent dye, was used to study dilution in the Ottawa river downstream of the CRNL reactor site. Dye was injected into the Process sewer and the concentration measured as it dispersed into the river and moved downstream. Addition times were varied from 20 minutes to 3 days to simulate typical radioactive releases. An initial dilution factor of 7 was observed at the river surface above the outfall. The dye dispersed into the wide deep section of the river downstream of the plant and was completely mixed throughout the river 8 km downstream. Short term discharges mixed throughout a large volume of the river, giving high dilution factors. Transit time to Pembroke (33 km) was 10 days. The tracer was detected as far downriver as Ottawa (200 km). (auth)

813 (AECL-2295) A STUDY OF DILUTION IN THE WINNIPEG RIVER NEAR WNRE. Merritt, W. F. (Atomic Energy of Canada Ltd., Chalk River (Ontario)). Aug. 1965. 8p. Dep.; AECL $0.20.

Rhodamine B, a fluorescent dye, was added to the water in the process sewer at the Whiteshell Nuclear Research Establishment, and its subsequent dilution in the Winnipeg River was measured using a fluorometer mounted in a small boat. The tracer followed the east branch of the river very closely and did not dilute throughout the whole volume of the river until a distance of 13 km was reached. Only small concentrations of tracer were detected in the water supply of Lac du Bonnet, a small village on the west bank of the river 8 km downstream from the sewer outfall.

814 (AEET/HP/RWD/3) DISPOSAL OF WASTE IN COASTAL WATERS. Working Party No. 4. P. R. Kamath and A. K. Ganguly (India. Atomic Energy Establishment, Trombay). Jan. 27, 1959. 40p.

A treatment is presented of the various factors to be considered for the ultimate disposal of radioactive wastes in sea water. The chemistry (constituents and radioactivity) and physics (circulation, sedimentation, etc.) of sea water are described. Chemical and biological interactions of radioactive wastes are considered. (D.L.C.)

815 DETERMINATION OF COLUMBIA RIVER FLOW TIMES DOWNSTREAM FROM PASCO, WASHINGTON, USING RADIOACTIVE TRACERS INTRODUCED BY THE HANFORD REACTORS. Nelson, Jack L. (Battelle Memorial Inst., Richland, Wash.); Perkins, R. W.; Haushild, W. L. Water Resour. Res., 2: 31-9(1966). (BNWL-SA-176).

To measure flow times of the Columbia River downstream from Pasco, Washington, as far as Astoria, Oregon, the use of two tracer methods was investigated. One method used the decay of a steady release of ^{24}Na (15-hr half life) to determine flow times to various downstream locations, and flow times were also determined from the time required for peak concentration of instantaneous releases of ^{131}I (8-day half life) to reach these locations. Flow times determined from the simultaneous use of the two methods agreed closely. The measured flow times for the 224 mi from Pasco to Vancouver, Washington, ranged from 14.6 to 3.6 days, respectively, for discharges of 108,000 and 630,000 ft^3/sec at Vancouver, Washington. A graphic relation for estimating flow times at discharges other than those measured and for several locations between Pasco and Vancouver was prepared from the data of tests made at four river discharges. Some limited data are also presented on the characteristics of dispersion of ^{131}I in the Columbia River. These measurements would be of great value in the event of an accident at Hanford that released undesirable amounts of activity to the river.

816 (BNWL-SA-466) DISPOSAL OF RADIOACTIVE WASTES INTO FRESH WATER. Honstead, J. F. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). Sept. 24, 1965. Contract AT(45-1)-1830. 36p. Dep. mn. CFSTI $2.00 cy, $0.50 mn.

Studies were made on the dilution, dispersion, deposition, removal, and transformation, and reconcentration of radioactive materials in fresh waters in order to estimate the movement and fate of the radionuclides in a water system, and also to estimate the degree of hazard involved. The processes of metabolic concentration by fish and wildlife are also considered. Aquatic foodstuffs were monitored for ^{32}P and ^{65}Zn. It was concluded that discharge of waste water into a fresh water system should be accompanied by an intensive monitoring program which includes special studies that detect the behavior and fate of radionuclides in the biological communities affected.

817 THE CLINCH RIVER STUDY: AN INVESTIGATION OF THE FATE OF RADIO-NUCLIDES RELEASED TO A SURFACE STREAM. R. J. Pickering, P. H. Carrigan, Jr., and F. L. Parker (Oak Ridge National Lab., Tenn.). New York, American Inst. of Chemical Engineers, 1964, Preprint 18e, 34p, $0.50. (CONF-393-3)

From American Institute of Chemical Engineers Symposium on Techniques at High Pressures, Memphis, Feb. 1964.

The Clinch River Study is a multi-agency effort to evaluate the physical, chemical, and biological effects of the release to the Clinch River of low-level radioactive wastes from the Oak Ridge National Laboratory. The major radionuclides released are ruthenium-106, cesium-137, cobalt-60, and strontium-90. Hydrologic and biologic studies indicated that radiation dosages known to result or assumed to result from direct and indirect means of exposure are below maximum permissible levels. Radionuclide concentrations in river water were measured at seven sampling stations on the Clinch and Tennessee Rivers. Mass balance calculations for 44 weeks of sampling indicate that losses of radionuclides from the water phase to river bottom sediments represent only a very small portion of the total radioactivity released to the river. A study of Clinch River bottom sediment cores collected in 1962 disclosed a recurring pattern of variation in radioactivity with depth, which may reflect past events in waste disposal operations at the laboratory. Current investigations are expected to provide information about the chemical forms in which the major radionuclides exist and the mechanisms by which they were incorporated in the sediments. (auth)

818 THE DISPOSAL OF RADIOACTIVE WASTE INTO IRRIGATION WATER. G. Michon. 9p. (CONF-642-2)

From Health Physics Society, Cincinnati, June 1964.

The disposal of radioactive waste into water streams used for irrigation was studied. Accumulation of long-lived radioistopes in the soil and plants from the irrigation water was studied. Data are presented in tables. It was concluded that the ICRP standards cannot be used as a basis for the disposal of radioactive waste in water used for irrigation since it allows man to be exposed to more radioactivity than is anticipated from the maximum permissible concentration. (M.C.G.)

819 ADVANCES IN WATER POLLUTION RESEARCH. Proceedings of the International Conference held in London, September 1962. Volumes 1, 2, and 3. Southgate, B. A. (ed. Vol.1); Eckenfelder, W. W. (ed. Vol. 2): Pearson, E. A. (ed. Vol.3). New York, Pergamon Press, 1964. 1402p. $45.00. (CONF-620911-(Vols.1, 2,3)).

Proceedings of the First International Water Pollution Research Conference are presented in three volumes. Self-purification, organic compounds in water, purification by biological means, absorption of impurities, processing radioactive wastes, marine waste discharge, and other methods of purifying wastes are discussed with special regard for efficiency and economics. The pollution effect of waste or contaminated water on natural waters, estuaries, and coastal areas is considered. Two papers were abstracted previously for NSA and appeared as 16: 32523 and 32622.

820 (CONF-640413, pp 34-9) HAZARDS OF TRITIUM OXIDES INDUCED IN HEAVY WATER REACTORS. Holmquist, C. E. (Swedish State Power Board, Stockholm).

The tritium activity induced in a heavy-water moderated and cooled power reactor presents some problems of considerable health significance. These problems increase with with the operation time of the reactor. The activity will be approximately 10 c DTO/D_2O after 10 years of operation. Unlike the fission product activity that is contained within the cladding of the fuel element, this activity is a constant hazard because of its vapor pressure from DTO. The health physics consequences of this in the operation of a full size nuclear power plant are the following: The personal protection must include a scrupulous industrial hygiene survey of the internal uptake and an extensive use of plastic suits. The layout of the station and the ventilation requirements will be greatly affected. Decontamination procedures will be complicated through the risk of internal contamination through lungs and skin. The environmental survey must include as a major item the tritium fallout around the power plant and the concentration in milk. Even minor mishaps in the operation of the plant may involve a pronounced internal risk. The discharge of the liquid effluent of low level tritium activity to a lake requires careful study of its capacity to act as a recipient for tritium since ordinary waste treatment is not successful in reducing this activity.

821 WATER POLLUTION FROM VARIOUS SOURCES INCLUDING URANIUM MINING OPERATIONS. Hilliard, T. R. (Ontario Dept. of Energy and Resources). Toronto, Canadian Nuclear Association, 1965, Preprint No. 65-CNA-219, 9p. (CONF-650515-2). ORINS

From International Conference and Exhibition of Canadian Nuclear Assn., Quebec.

The control of contamination of water supplies by wastes from various industries is discussed. A study of expenditure for pollution control and related research in United States chemical companies showed that expenses over the last several years averaged ⅓ of a percent of net plant and equipment valuation, and that pollution control expenses amounted to about 0.2 percent of net sales. The cost to prevent water shortage, to control waste, and to regain a lost position will be high. Nature performs the first act of replenishment by a perpetual recycling system and it is our responsibility to aid nature so that use and re-use of water can be achieved for our modern way of life. Water will not remain the cheap commodity that it has been in the past. Competition for its use will increase, and pollution control will be accepted by the public. (auth)

822 (HW-81746(p.3.81)) PRELIMINARY MEASUREMENTS OF THE ADSORPTION OF REACTOR EFFLUENT WATER RADIONUCLIDES BY COLUMBIA RIVER WATER PARTICLES. L. L. Humphreys and R. W. Perkins (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.).

Overall rates of absorption of reactor effluent water radioisotopes (^{56}Mn, ^{24}Na, ^{239}Np, ^{51}Cr, ^{64}Cu, ^{46}Sc, ^{65}Zn, and ^{140}La) by Columbia River water particles were measured. In the first minute of contact, absorption ranged from 0.2 to 14%. This was 25 to 40% of that occurring in 1500 min.

823 (HW-83614) PROGRESS IN STUDIES OF RADIONUCLIDES IN COLUMBIA RIVER SEDIMENTS. A Summary of Hanford Achievements in This Program Under General Electric, 1963–1964. J. L. Nelson, R. W. Perkins, and J. M. Nielsen (Hanford Atomic Products Operation, Richland, Wash.). Dec. 1964. Contract AT(45-1)-1350. 26p. Dep.; $2.00(cy), 1(mn) CFSTI.

Observations on the interaction of the Columbia River radionuclides and sediments have provided considerable information on the behavior of these materials. A limited survey of the river bottom sediment inventory and a study of its radionuclide composition suggests that the main radionuclide deposits are in the McNary Reservoir and in the estuary. A study of the radionuclide concentrations and particularly of radionuclide ratios such as ^{65}Zn to ^{51}Cr in suspended and deposited sediments has clearly shown that a large amount of the Columbia River sediments with their associated radioactivity are scoured from the river bottom during the spring freshet. The magnitude of this sediment scouring action and the location of final deposition are being studied. The tracing of the scoured material by means of its associated radioactivity will answer not only the question of its final deposition but is expected to shed light on the movement and deposition of sediments in the Columbia River shipping channels which at present require almost continuous dredging to keep them open. Fractionation of the river bottom sediments into particle size groups has shown a maximum in radioactivity associated with both the smallest and largest particle sizes. This anomaly is due to the presence of larger amounts of high specific activity organic material in the coarse fractions. A clean separation of the organic material from inorganic sediments without some leaching of the inorganic material has not been accomplished; however, the studies have shown that the organic material in the sediments contains a substantial portion of the radioactivity and must certainly be considered in studying the sorption-release mechanism of Columbia River radionuclides. The physical and chemical forms of the more abundant long-lived radionuclides were investigated during the spring freshet at sampling locations between the reactors and Vancouver. A large fraction of the observed ^{65}Zn, ^{54}Mn, and ^{46}Sc from sampling stations below McNary Dam was associated with particulate material which had presumably been scoured from the McNary Reservoir. Only a small fraction of the total amount of these radionuclides which were carried by the river was in true solution.

824 VARIATIONS IN ELEMENTAL CONCENTRATIONS IN THE COLUMBIA RIVER. W. B. Silker (General Electric Co., Richland, Wash.). Limnol. Oceanog., 9: 540-5(Oct. 1964). (HW-SA-3147)

The seasonal changes in concentration of several elements in the Columbia River at Hanford, Washington, were measured during 1962. An annual spring increase in the concentration of arsenic, manganese, copper, and lanthanum was related to the introduction of water from the Spokane River to the Columbia River system, while other elemental

concentration changes were ascribed to the relative contribution of the Pend Oreille and Spokane rivers to the Columbia River. Data are presented on the concentration of several elements at various points on the Columbia River upstream from the Hanford project.

825 (NP-13685) REPORT OF THE OCEANOGRAPHIC SURVEY CARRIED OUT IN YOKOHAMA HARBOR WITH SOME CONSIDERATIONS ON THE DISPOSAL OF RADIOACTIVE WASTES FROM NUCLEAR POWERED SHIPS INTO THE MARINE ENVIRONMENT. Gensenkyo-34. (Nuclear-Powered Ship Research Assn. of Japan, Tokyo). Feb. 19, 1963. 90p.

Some aspects of radioactive wastes from nuclear powered ships, problems of the waste disposal, and hazards of nuclear powered ships are outlined. The results of the oceanographic survey carried out in Yokohama Harbor are given with the features of the tidal currents, the distributions of chlorinity, soluble silicates, and dissolved oxygen in the harbor water. The various theories of diffusion are outlined as a basis of considerations on the safety evaluation of the waste disposal into the marine environment sound. Bases for evaluating safe discharge rates of wastes from nuclear powered ships are described, and analytical descriptions of the experiments on diffusion of dye carried out in Santa Monica Bay, California, and Yokohama Harbor are also given. In the harbor water of Yokohama, the radius of the dye patch (r_0) is well expressed by $r_0^2 = 0.7\ t^2 \ln(\tau/t)$, where t is the time after the release of dye from an instantaneous point source and τ the time when the patch disappears is known from the observation. In the above equation t and τ are expressed in minutes and r_0 in meters. τ was estimated as 420 minutes in one case when the initial amount of dye was 1 kg. A summarized translation of a referred report relating to the disposal of radioactive wastes from nuclear powered ships into the marine environment is given. It is concluded that it is essential for a systematic oceanographic survey program to be initiated as soon as possible to establish the operating doctrine for nuclear powered ships. The program should be carried out by any Government agencies and cover whole of the appropriate inshore waters including harbors, the continental shelf, and the open sea. Reliable data of ocean and tidal currents, the exchange rate of sea water, the processes of dispersion of substance in the various localities of the marine environment are scarce at present, and also the concentration by marine organisms of the various significant isotopes in the wastes from nuclear powered ships, or the route by which such introduced activity would return to man from sea, are not yet studied sufficiently to meet the expected requirements relating to the operation of nuclear powered ships. (auth)

826 (NYO-8104) HORIZONTAL DIFFUSION FROM AN INSTANTANEOUS POINT-SOURCE DUE TO OCEANIC TURBULENCE. Technical Report No. 32. Akira Okubo (Johns Hopkins Univ., Baltimore. Chesapeake Bay Inst.). Dec. 1962. Contract AT(30-1)-1477. 124p.

A discussion is presented of horizontal turbulent diffusion of a patch of a diffusive substance released from an instantaneous point-source in the sea—a situation to which a knowledge of the similarity theory of turbulence may be applied to some extent. A systematic derivation of some proposed concentration distributions for a patch is developed on the basis of a non-Markov hypothesis combined with knowledge of the rate of mean square dispersion. A new proposal for the concentration distribution is also presented. The solutions of Joseph and Sendner, Ozmidov, Obukhov, Schönfeld, Okubo and Pritchard, and Okubo are critically discussed in comparison with available information about horizontal diffusion in the sea. A re-examination is made of the relations between the variance of the distribution and diffusion time, and between the coefficient of eddy diffusion and the scale of the phenomenon, using accumulated data. In addition to this, some speculation about the two-dimensional horizontal energy spectrum of oceanic turbulence is presented. (auth)

827 (NYO-10447) OPERATIONS RESEARCH IN DISPOSAL OF LIQUID RADIOACTIVE WASTES IN STREAMS. Final Report. Thomas, Harold A. Jr. (Harvard Univ., Cambridge, Mass.). Dec. 1965. Contracts AT(30-1)-966; AT(30-1)-3068. 202p. Dep. mn. CFSTI $6.00 cy, $1.25 mn.

A review and summary of the general principles of disposal of low-level radioactive wastes in streams is presented. A classification of physical and economic factors that are important in radioactive waste disposal in streams is included. From this, a classification of rivers and other fresh water bodies into four basic types was developed. A descriptive mathematical model is presented to show the logical structure of the scheme and to indicate the wide variety of waste disposal environments that obtain in different streams. In some types of streams a potential exists for storage and sporadic release of activity from benthal deposits and biomasses that may constitute a public health hazard. Various mechanisms of release are discussed. Three different models are also presented for economic and engineering analysis of stream disposal systems for radioactive wastes. Test results of model laboratory streams that were operated to supply information about the interaction of stream and radiocontaminant compounds and processes that could not readily be obtained from field studies on prototype streams are described.

828 (ORNL-3697(p.32-42)) CLINCH RIVER STUDY. F. L. Parker, K. E. Cowser, W. P. Bonner, P. H. Carrigan, Jr., R. J. Pickering, and B. J. Frederick (Oak Ridge National Lab., Tenn.).

Analyses of stable chemical constituents in discharge-weighted water samples from the Clinch and Tennessee Rivers show that variations in constituent concentrations are not great; bicarbonate is the predominant anion and calcium the predominant cation. Seasonal variations in concentration of several of the constituents were detected. Calculation of correlation coefficients for constituent pairs indicated that turbidity, apparent color, centrifuged color, suspended solids, iron, and manganese concentrations varied in a similar manner. Measurement of gross gamma radioactivity in 146 cores of Clinch River bottom sediment has shown that radioactive sediment up to 8.7 ft thick is present in the river. Tracer studies were conducted to study the effect of releases from Melton Hill Reservoir on dispersion of radioactive releases from White Oak Dam. Analyses of fish taken from the Clinch River show that about 2.5% of the maximum permissible intake (MPI) would be received from eating 37 lb of fish per year. Radiation levels near filter beds in water plants upstream and downstream from White Oak Creek are at background. Bottom sediments from the Clinch River were selectively leached. Salts between a pH of 6 to 8 removed less than 10% of the ^{60}Co, ^{106}Ru, and ^{137}Cs, though up to 80% of the ^{90}Sr. At lower pH's up to 80% of the ^{90}Sr and ^{60}Co were leached. At higher pH's up to 50% of the ^{106}Ru and up to 20% of the ^{60}Co are released.

829 (ORNL-3697(p.95-104)) CLINCH RIVER AND RELATED AQUATIC STUDIES. D. J. Nelson, S. I. Auerbach, B. G. Blaylock, et al. (Oak Ridge National Lab., Tenn.).

The series of analyses of stable strontium in white crappie flesh and bone was completed, and it was possible to

test the relation of the distribution of stable strontium and ^{90}Sr between the environment and the fish. Uptake and accumulation of strontium and calcium by algae and snails representing contrasting types of biogeochemical systems are reported. There is an increase of newly occurring chromosomal aberrations in the natural population of Chironomus tentans from White Oak Creek. These newly occurring aberrations are eliminated by selection or genetic drift and are not maintained in the gene pool of the population. The increased mutation rate produced by the chronic irradiation in the White Oak Creek population has not reduced the amount of chromosomal polymorphism in this natural population.

830 (ORNL-P-605) MOVEMENT OF RADIONUCLIDES IN THE CLINCH RIVER. F. L. Parker, B. J. Frederick, and P. H. Carrigan (Oak Ridge National Lab., Tenn.). Oct. 6, 1964. Contract W-7405-eng-26. 14p. (CONF-772-1). Dep.(mn); $1.00(cy), 1(mn) OTS.

From Proceedings of the Third Sanitary Engineering Conference, Nashville, May 1964.

The effects of temperature on the movement of radionuclides in the Clinch River, a highly regulated river with stratified flow, was studied using Rhodamine B dye as a tracer. The dispersion of the dye was followed by fluorometric measurements at various sampling stations. The effects of water released from Melton Hill Dam on water flow and mixing of radioactive waste effluents from White Oak Creek, that discharges waste from Oak Ridge National Laboratory, were determined for various temperature profiles. (C.H.)

831 (ORNL-tr-965) PROPOSAL FOR THE CONSTRUCTION OF A SEWER FOR THE DISCHARGE OF INDUSTRIAL WASTE WATER CONTAINING RADIOACTIVE SUBSTANCES INTO THE ESCAUT RIVER. (Ministere de la Sante Publique et de la Famille, Brussels (Belgium). Administration de l'Hygiene Publique). Translated for Oak Ridge National Lab., Tenn., from manuscript dated Aug. 1964. 30p. Dep. mn. JCL $2.60 fs, $0.83 mf.

The construction of a sewer along the Albert Canel is proposed to move industrial waste water to the Escaut River downstream from Antwerp from already established industries or from those to be established in the region traversed by this waterway as well as the construction of a side-sewer along the canal from Herentals to Bocholt, discharging into the former at Herentals and intended to collect the industrial waste from plants established in Mol, Dessel, and Olen. Control of radioactivity of Escaut water before and after disposal is discussed, and data on various radioisotopes present in the water are tabulated.

832 (PG-Report-500) THE BIOLOGY OF THE SOLWAY FIRTH IN RELATION TO THE MOVEMENT AND ACCUMULATION OF RADIOACTIVE MATERIALS. I. GENERAL INTRODUCTION. E. J. Perkins and B. R. H. Williams (United Kingdom Atomic Energy Authority. Production Group. Chapelcross Works, Annon, Dumfriesshire, Scotland). 1963. 16p.

An investigation was made of the Solway Firth, an inlet of the North East Irish Sea, designed to determine its capacity for the receipt of radioactive effluent. The general features of the area and its fisheries are described. The nature of the ebb-flood channel system is shown to be dependent upon the geology of the area. (auth)

833 (PG-Report-605) THE BIOLOGY OF THE SOLWAY FIRTH IN RELATION TO THE MOVEMENT AND ACCUMULATION OF RADIOACTIVE MATERIALS. IX. SEA-BED AND SEA-SURFACE DRIFTER RELEASES. C. N. E. IRISH SEA AND SOLWAY FIRTH 1963. E. J. Perkins, M. Bailey, and B. R. H. Williams (United Kingdom Atomic Energy Authority. Production Group. Chapelcross Works, Annon, Dumfriesshire, Scotland). 1964. 41p. Dep.(mn); $2.00(BIS); 10s. (HMSO).

Results are reported from studies of the ebb and flow of tidal waters at the surface and sea bed levels of the Solway Firth and adjacent Irish Sea during 1962 and 1963. The movements of residual silts were also studied. Sea-bed and sea-surface drifters were released in batches of 25 or 50 and the distribution of recoveries analyzed. It was estimated that about 45% of the drifters released were recovered. The effects of rainfall, river flow, wind, tides, and other factors on distribution of the drifters are discussed. The relation of the movements of water and silt to the dispersal of radioactive wastes is implied. (C.H.)

834 (PG-Report-647) HEALTH AND SAFETY RESEARCH AT CHAPELCROSS. Annual Report, 1964. J. H. Martin (Production Group, United Kingdom Atomic Energy Atuhority, Annon (Scotland)). 1965. 8p. Dep.

Progress is reported in investigations of biological indicators of radiation injury with emphasis on radioinduced changes in thymine catabolism in rats and man. The metabolism of ^{85}Sr and ^{239}Pu in rats and the removal of ^{85}Sr by treatment with citrate and carrier Sr were studied. Data are included on investigations of the effects of waste disposal on radioactivity of the Solway Firth and Dounreay pond waters; neutron dosimetry, and the development of low-level counting methods for use in environmental monitoring programs. (C.H.)

835 (PHS-999-RH-13) BEHAVIOR OF CERTAIN RADIONUCLIDES RELEASED INTO FRESH-WATER ENVIRONMENTS. Annual Report, 1959-1960. Friend, Albert G.; Story, Albert H.; Henderson, Croswell R.; Busch, Kenneth A. (Public Health Service, Washington, D. C. Div. of Radiological Health). June 1965. Contract AT(49-5)-1288. 89p. Public Health Service Free.

Radionuclides from various sources, including facilities that produce, process, and handle nuclear material, are entering the fresh-water environment. As part of its assigned responsibilities, the Public Health Service is concerned with the radiological conditions in the various components of the aquatic environment. The Service's Division of Radiological Health, in cooperation with the Atomic Energy Commission, has conducted a number of investigations of streams below atomic energy installations. Two such studies are presented, one on the Mohawk River near Knolls Atomic Power Laboratory and the other on the Clinch and Tennessee Rivers near the Oak Ridge National Laboratory. In general, the results show that all segments of the aquatic environment related to these rivers contain measurable quantities of manmade radionuclides. The results of some uptake studies, carried out in a DRH laboratory, are included to supplement the river studies and to aid in interpreting the results obtained in the field. (auth)

836 (RLO-1725-41) DISTRIBUTION AT SEA OF COLUMBIA RIVER WATER AND ITS LOAD OF RADIONUCLIDES. Barnes, Clifford A.; Gross, M. Grant (Washington Univ., Seattle. Dept. of Oceanography). [1966]. Contract AT(45-1)-1725. 12p. (CONF-660507-1). Dep. mn. CFSTI $1.00 cy, $0.50 mn.

For Presentation at Symposium for the Disposal of Radioactive Wastes into Sea, Oceans, and Surface Waters, Vienna.

Properties, distribution, and movement of Columbia River water in the Northeast Pacific Ocean and of the sediments near the river mouth were investigated. Distribution of salinity, nutrients, chlorophyll a, fission products from atmospheric fallout, and neutron-activation products from the Hanford reactor are reported. Surface salinities near the mouth of the river were generally under 35‰. Studies of river discharge yielded two periods of peak flow, the highest being from mid-May until mid-June. During late spring and summer the Columbia River

contributes over 90% of the runoff reaching the Northeast Pacific Ocean between 38° and 48°N. Distinctive features of the discharge plume for late summer 1961 are plotted. Distribution of salinity, nitrate, phosphate, and chlorophyll a in surface waters near the Columbia River were determined during late summer 1961 and in winter 1962. Radionuclide contamination by ^{51}Cr, ^{95}Zr and ^{95}Nb was detected in surface waters, and ^{95}Zr, ^{95}Nb, ^{160}Ru, and ^{144}Ce were found in sediments. The vertical distribution and abundance of fission products in sediment cores provide possible means of determining the rate of sediment accumulation in the region and of evaluating the effects of the vertical mixing of sediment due to the activities of marine burrowing organisms. Preliminary data suggest that rapid vertical mixing extends to a depth of at least 2 cm in the cores. Radionuclides from the Hanford effluent provided the opportunity to study the distribution of Columbia River sediment on the Continental Shelf and its movement in the area. Zinc-65 proved to be the most valuable of the neutron-activated products in this respect.

837 ENVIRONMENTAL MONITORING AROUND A URANIUM MILL IN THE UNITED STATES OF AMERICA. Tsivoglou, E. C. (Robert A. Taft Sanitary Engineering Center, [Cincinnati,] Ohio). pp 63-70 of STI-PUB-98.

An environmental monitoring program is described that illustrates the combined effects upon a river environment of waste-disposal operations from an operating uranium mill, and the residual effects from an abandoned uranium mill located approximately 15 miles upstream. The area described is located in the southwestern United States in the State of Colorado. Water samples are analyzed primarily for dissolved ^{226}Ra and U.

838 (TID-19608) SPECIES COMPOSITION AND DISTRIBUTION OF MARINE NEKTON IN THE PACIFIC OCEAN OFF OREGON. Final Progress Report, September 1, 1961 through September 1, 1963. (Oregon State Univ., Corvallis. State of Science). Contract AT(45-1)-1726. 156p.

Studies were made over a 2-yr period of the distribution, ecology, and radioactivity of oceanic nekton and macroplankton in the Pacific Ocean off the Oregon coast. The role of fishes, squids, shrimps, euphausiids, and other swimming animals in the accumulation and biological transport of nuclides in ocean waters was investigated. Data are included on the species composition of oceanic marine nekton and macroplakton; the distribution of animals in regard to spatio-temporal differences related to depth and diurnal vertical migration, distance from shore, latitude, and seasons of the year; and the collection and identification of oceanic animals for analysis for radioisotopes formed by neutron activation at the Hanford reactors and passed to the ocean by waters of the Columbia River. The content of $Zr^{95}-Nb^{95}$, Zn^{65}, Cr^{51}, Ce^{141}, Ru^{103}, Co^{60}, K^{40}, Mn^{54}, Cs^{137} in marine animals from various ocean depths and in relation to the Columbia River plume and seasons was determined. Sampling and analytical methods are described. (C.H.)

839 DISPOSAL OF RADIOACTIVE WASTES IN THE SEA. Alfredo Spena. Ann. Med. Navale (Rome), 67: 463-84(July-Aug. 1962). (In Italian)

The disposal of highly radioactive industrial wastes is contrasted with that of medium and low activity wastes, which may be treated either by concentration and containment or dilution and dispersion. The U. S. Atomic Energy Commission's standards for alpha, beta, and gamma emitters are reviewed; applications to disposal at sea are noted, both in AEC tests and those of others. The following subjects are considered; the three zones of the oceans: coastal regions, continental shelf, and the open sea, the last of which is divided into superficial and deep strata; the natural radioactivity of the sea; the fate of radioactive wastes immersed in the sea (dispersion, sedimentation, biological assumption by marine bacteria, phytoplankton, zooplankton, benthos, fish); factors affecting the maximum quantity of radioelements disposable in the sea. Maximum permissible doses for various segments of the population are reviewed, and also maximum permissible concentrations relative to the present problem. Disposal of radioactive wastes by nuclear propelled vessels is also discussed. Conclusions are drawn for disposal of small amounts of radioisotopes near the coast, and greater amounts ten miles from the coast at a depth of 1800 m; sampling of algae and phytoplankton every 15 days, fish and crustaceans every month, and sediment every two months; types of controlling bodies; $1/25$ of maximum permissible dose in territorial waters as the upper level, (i.e., 0.02 R/yr), and 0.2 R maximum in 30 yr in the open sea, as safety limits. (BBB)

840 BEHAVIORS IN MARINE WATER OF RADIOACTIVE NUCLIDES CONTAINED IN COOLING WATER OF THE ATOMIC REACTOR JRR-1. Ojima, Tsutomu; Toratani, Hirokazu (Radiation Center of Osaka Prefecture). Ann. Rept. Radiation Center Osaka Prefect., 5: 96-8(Mar. 1965).

In a fundamental survey of the behavior of radioactive nuclides in marine water, the cooling water of the atomic reactor JRR-1 and a $^{60}CoCl_2$ solution were used as radioactive substances. It was recognized that the radioactive nuclides contained in primary cooling water in the ionic state formed radiocolloids partly because of the higher pH of marine water. No precipitates were formed as a result of direct reaction with marine salts. It was supposed that radioactive nuclides would be adsorbed by planktons and mud floating in marine water at comparatively low ratio. It was considered, consequently, that most radioactive nuclides would be diluted with marine water and moved away with a current, but, as they had a chance of contact with mud, most of them would be fixed by the mud. In this case the fixation ratio was larger for ^{60}Co than for ^{54}Mn and these radioactive nuclides were fixed firmly in the non-exchangeable state. (auth)

841 CONCERNING NUCLEAR PROPULSION OF SHIPS AND CHANCE OR UNFORESEEABLE ACCIDENTS: PRESENT STATUS. F. Le Chuiton and L. Tanon. Bull. Acad. Natl. Med. (Paris), 146: 634-8(Nov. 20-7, 1962). (In French)

Problems and possible control measures relating to the use of atomic power for submarines are reviewed. Important considerations included precise determination of the ratio of the mass of the reactor to vessel's tonnage; adaptability of the reactor required by changes in speed of the ship; detailed study of reactor's structural stability, because of constant motion of the ship in stormy weather; study of eight variations due to such motion and of vibrations produced by propellers. Additional problems considered are possibilities of collision, grounding, icebergs, shipwreck, accidental fire due to reactor breakdown, and waste disposal. Findings are reviewed from a congress that met in London to pool research on safety measures and a Euratom conference in Brussels in 1960 on responsibility in the event of accidents of atomic powered vessels. Data are included from a study of radioactive wastes and their effect on marine flora and fauna. In 1962 a group of experts, meeting in Monaco under the auspices of the International Atomic Energy Agency, reached two opposing conclusions: disposal of all waste radioactive material is potentially dangerous and should be prohibited; and disposal of certain wastes is not dangerous and can be permitted.

In the design of ocean-going reactor-powered vessels, it was considered important to adopt a policy of least risk to human life, since the walls retaining radioactive material undergo changes with time and are not entirely leak-proof, and also since unpredictable ocean currents exist. It is concluded that the hazard of marine pollution is limited with improved vessel and reactor construction design, and because collision of submarines is less likely than that of surface ships. (BBB)

842 THE ELIMINATION OF RADIOACTIVE EFFLUENTS. P. Bovard and M.-E. Gahinet. Eau, L, 50: 301-6(May 1963). (In French)

The sources of radioactivity in water are discussed, together with the treatment and release of radioactive effluents. Controls on release and activity levels are also indicated. (D.C.W.)

843 DILUTION OF RADIOACTIVE EFFLUENTS IN A SLOW-MOVING RECIPIENT. S. Bergstroem (Aktiebolaget Atomenergi, Stockholm). Euro Nucl., 2: 129-33(Mar. 1965).

The active waste release situation at the Ågesta Heat and Power Station is described. The use of high-velocity jet release of the low-level waste in Lake Magelungen as a result of water court verdicts and the recipient characteristics is discussed. Mass transfer and concentration reduction in round symmetrical jets are considered against the background of theoretical and experimental knowledge. Results from experiments made at Studsvik are quoted and equipment described. (auth)

844 RADIOACTIVE CONTAMINATION OF THE SEA. Masamich Saeki (National Inst. of Radiological Sciences, Chiba, Japan) and Katsuko Saruhashi. Genshiryoku Kogyo, 10: No. 7, 7-29(July 1964). (In Japanese)

Several articles presented under the title heading include the following: The disposal of radioactive wastes into the sea (by Saeki): the present state and future of radioactive wastes in Japan, treatment and disposal of wastes, disposal into the sea and coastal waters, disposal of solid waste packages into the deep sea, and disposal into river waters. Radioactive contamination of the sea (by Saruhashi): contamination in the northern Pacific, ^{137}Cs and ^{90}Sr in surface waters, vertical distribution of these isotopes, disposal into the deep sea, and problems. Radioactive contamination of marine products (by Hiyama): effect of sea contamination on marine products, uptake of radioactive isotopes (RI) by marine organisms, concentrations of RI in marine organisms, disposal by dilution and isolation, protection against the radiations, and necessity of the law that controls sea contamination. Nuclear-powered ships and contamination of the sea (by Hattori): reactors used in nuclear-powered ships, radioactive wastes from the ships, disposal of wastes into the sea, and accidents. Radioactive contamination of the sea and international law (by Oda): the progress of the law panel of IAEA started in January 1961. (NSA of Japan)

845 RADIOACTIVE CONTAMINATION OF THE SEA. Katsuko Saruhashi (Meteorological Research Inst., Japan). Genshiryoku Kogyo, 10: No. 7, 12-17(July 1964). (In Japanese)

The migration of the center of contamination in the northern Pacific during the period June 1954–August 1955 is described. The concentrations of ^{137}Cs and ^{90}Sr in the northwestern Pacific decreased and became uniform during 1957–1962, while the ^{137}Cs concentration in the northeastern Pacific increased during 1959–1962. The vertical distribution of ^{137}Cs and ^{90}Sr is discussed, especially for the diffusion process. The rates of leakage or diffusion into the surface waters for a period of 100 years are calculated for ^{90}Sr packaged or not packaged and then disposed into the deep sea (4000 m deep). Factors influencing the behaviors of radioactive substances in the sea are also described. (NSA of Japan)

846 DISPOSAL OF EFFLUENTS CONTAMINATED WITH RADIOACTIVE SUBSTANCES. A. T. Avdonin. Gigiena i Sanit., 28: No. 8, 14-19(1963).

Problems connected with the disposal of effluents from various enterprises dealing with open radioactive substances are discussed. Data are presented on the quantity and degree of effluent contamination and on the method of determining the extent of effluent decontamination required before its discharge into the sewerage system or a water basin. The choice of methods of effluent decontamination are discussed, and a short description and assessment of various decontamination methods is given. (Public Health Eng. Abstr., 43: Dec. 1963)

847 A STUDY OF DILUTION IN THE OTTAWA RIVER USING RHODAMINE B. I. NPD TO DEEP RIVER. W. F. Merritt (Atomic Energy of Canada Ltd., Chalk River, Ont.). Health Phys., 10: 195-201(Mar. 1964).

Rhodamine B, a fluorescent dye, was used to study dilution in the Ottawa River downstream of the NPD reactor site. Dye was injected into the NPD process sewer for a 3-hr period and the concentration measured as it dispersed in the river. An initial dilution of 100 was observed near the outfall. A wide, deep section of the Ottawa River downstream of the reactor site acted as a detention and mixing basin. Dye concentrations at the town of Deep River water intake, 16 km downriver, indicated a minimum overall dilution of the effluent of 3×10^4. The water velocity in this section of the Ottawa River was about 9 km/day. (auth)

848 ENVIRONMENTAL MONITORING ASSOCIATED WITH THE DISCHARGE OF LOW ACTIVITY RADIOACTIVE WASTE FROM WINDSCALE WORKS TO THE IRISH SEA. H. J. Dunster (United Kingdom Atomic Energy Authority, Harwell, Berks, Eng.), R. J. Garner, H. Howells, and L. F. U. Wix. Health Phys., 10: 353-62(May 1964).

Since 1952, low activity radioactive waste has been discharged by pipeline from Windscale Works to the Irish Sea off Sellafield. Environmental monitoring studies have been carried out throughout this period in order to assess the resultant radiation dose to man, and hence to estimate the discharge rate which would correspond to the maximum permissible radiation doses recommended by the International Commission on Radiological Protection. These studies have been concerned principally with those materials most likely to contribute substantially to the dose to man, and they have been supplemented by more widespread marine surveys and by laboratory studies aimed at gaining information about the movement and behavior of radioactive nuclides in the marine environment. It has been concluded from these studies that the rate of discharge from Windscale Works to the Irish Sea cannot be permitted to exceed about 100,000 C/month of mixed fission products, and that specific restrictions are needed on a number of individual nuclides, namely, ^{90}Sr (about 2000 C/month), ^{106}Ru (about 5000 C/month), and ^{144}Ce (about 20,000 C/month). The present discharge rates are about 4000 C/month of mixed fission products, of which about half is ^{106}Ru and about 2% ^{90}Sr. (auth)

849 SORPTION PHENOMENA OF RADIONUCLIDES TO CLAY PARTICLES IN RIVER WATER. Karen Garder (Inst. for Atomic Energy, Kjeller, Norway) and Olav Skulberg. Intern. J. Air Water Pollution, 8: 229-41(Mar.-Apr. 1964).

The sorption of radionuclides to river sediments was investigated. Both laboratory experiments and experiments in a model recipient were performed. In the laboratory experiments, post-glacial marine clay from the River Nitelva area, Norway, was investigated. In the model recipient the investigation was performed on bottom sediments of the same type as in Nitelva. The sorption of cesium, cerium, ruthenium, zirconium-niobium, yttrium, calcium, and strontium to sediments were measured. The results showed that appreciable amounts of cesium and cerium were sorbed to the sediments. The sorption of ruthenium, zirconium-niobium and yttrium was intermediate, and the sorption of strontium was insignificant. The results also indicate that in Nitelva the removal of radionuclides from solution was mainly due to sorption of suspended clay minerals. This work is part of an extensive investigation of problems related to radioactive pollution of fresh-water environments. (auth)

850 RADIOACTIVITY AND ITS RELATIONSHIP TO OCEANIC FOOD CHAINS. Charles Osterberg (Oregon State Univ., Corvallis), William G. Pearcy, and Herbert Curl, Jr. J. Marine Res. (Sears Found. Marine Res.), 22: 2-12(Jan. 15, 1964).

Gamma-ray spectra of some primary producers (single-cell plants), filter-feeding herbivores, and carnivores, assigned to trophic levels I, II, and III–V, respectively, were prepared from marine samples taken in the Pacific off Oregon during 1961-1962. The organisms were exposed in their natural environment to both fission products from fallout and neutron-induced radionuclides from reactors on the Columbia River. Comparisons of spectra of organisms from different trophic levels, determined from stomach contents and the literature, show that $^{95}Zr-^{95}Nb$ and ^{141}Ce were concentrated by primary producers and herbivores but not by carnivores. ^{51}Cr was abundant in only filtered samples (primary producers). ^{54}Mn, ^{60}Co, and ^{137}Cs were found in only herbivores and carnivores. ^{65}Zn was found in every marine organism examined. The authors conclude that the abundance of $^{95}Zr-^{95}Nb$ and ^{141}Ce, in particular, may be useful in marine trophic-level studies. Peaks due to these fission products are greatly reduced in spectra of predaceous animals as compared with spectra of herbivores. (auth)

851 RESEARCH FOR THE CONTROL OF RADIOACTIVE POLLUTANTS. E. C. Tsivoglou (Robert A. Taft Sanitary Engineering Center, Cincinnati). J. Water Pollution Control Federation, 35: 242-59(Feb. 1963).

Laboratory and field research studies on the α radioactivity in streams due to natural radioactivity or contamination by the uranium milling industry are discussed. The use of special photographic emulsions for the identification and counting of α-emitting radioisotopes in various kinds of samples, the use of gross α radioactivity analysis of environmental samples of all types as an indicator of pollution from uranium mill wastes, and the leaching of Ra^{226} from spent uranium are solids deposited in river beds, and results of in-plant surveys of uranium mills for an evaluation of the waste potential of the different processes of uranium extraction are discussed. Results are summarized from field studies on radioactive pollution from uranium mills in streams of the Colorado River Basin and a survey on the fate of P^{32} and Zn^{65} in the Columbia River as a result of radioactive waste disposal practices at the Hanford works. (C.H.)

852 Zn^{65} CONTENT OF SALPS AND EUPHAUSIIDS. Charles Osterberg (Oregon State Univ., Corvallis). Limnol. Oceanog., 7: 478-9(Oct. 1962).

Comparisons were made of the Zn^{65} content in plankton collected during July, August, and November 1961 in the Pacific Ocean near the mouth of the Columbia River, 200 mi south, and 50 mi offshore. A gradual decrease with distance from the mouth of the Columbia River indicated that the river is the major source of this isotope. (C.H.)

853 ^{90}Sr AND ^{137}Cs IN COLUMBIA RIVER PLUME, JULY 1964. Park, Kilho (Oregon State Univ., Corvallis); George, Marilyn J.; Miyake, Yasuo; Saruhashi, Katsuko; Katsauragi, Yukio; Kanazawa, Teruko. Nature, 208: 1084-5(Dec. 11, 1965).

The extension of Columbia River outflow into the Pacific Ocean, or plume, was analyzed for ^{137}Cs and ^{90}Sr content in a study of ocean mixing processes. Samples of water were collected from the surface of the river, the river plume, and its vicinity during July 1964. The river water contains radionuclides from world-wide fallout and from Hanford atomic reactor waste. Both ^{137}Cs and ^{90}Sr contents in the Columbia River plume waters were found to be proportional to the contribution of the river waters as determined from salinity measurements.

854 THE EVALUATION AND CONTROL OF THE RADIOACTIVE WASTE DISPOSAL TO SEA. Yoshio Hiyama (Tokyo Univ.). Nippon Genshiryoku Gakkaishi, 6: 406-11(July 1964). (In Japanese)

Types of hazards, groups of people to be considered, and the basic dose effect relation to evaluate the size of hazards are discussed. As the methods of disposal permissible at present, two ways are mentioned. One is the high radioactive material kept in unbreakable container can be disposed to the bottom of some depth. The other is the diluted low level waste that can be continuously poured into some area where dilution is going on continuously. Any other means other than these two, for example a bay where insufficient exchange of water occurs and where the surface water is usually productive, is not suitable for disposal. The concentration factor obtained both by the experiment using tracers and by chemical analysis of the stable element in sea water and organisms differs greatly according to the radio-nuclide and the taxonomical position of the organisms. The rate of uptake differs depending on the chemical form, radio-nuclide, temperature of sea water, growth stage, and the kind of organism. Ways to evaluate Maximum Permissible Concentration of sea water are described. Three steps of food chain, radioisotope concentration in sea water, index organisms, and marine food products on the market should be monitored constantly. For this purpose proposal of International Index Organisms have been made. A table of concentration factors determined is included. (auth)

855 REVIEW OF REACTOR FUEL REPROCESSING PLANT SAFETY IN FOREIGN COUNTRIES. Maruyama, Masatomo (Atomic Fuel Corp., Tokyo). Nippon Genshiryoku Gakkaishi, 8: 81-90(1966). (In Japanese).

Preliminary site surveys and safety control measures associated with reactor fuel reprocessing plants as practiced in foreign countries are reviewed. Representative items selected for description in some detail include the Windscale plant in Britain regarding disposal to sea of liquid radioactive wastes, the Idaho Chemical Processing Plant in the U. S. regarding the release to the atmo-

sphere of gaseous wastes, the Hanford plant in the U. S. regarding the movement of liquid wastes dispersed into the ground, and the Marcoule plant in France regarding safety control.

856 DISPOSAL OF RADIOACTIVE WASTES IN FRESH WATER. F. L. Parker. Nucl. Safety, 6: No. 1, 89-97(1964).

The discharge of radioactive waste into fresh-water systems is discussed. The Clinch River studies, IAEA recommendations, studies of river sediment, system analyses of river pollution, and foreign studies of waste balances are discussed. (M.C.G.)

857 THE POLLUTION OF THE WATERS OF RIVERS AND OCEANS BY RADIOACTIVE WASTES. Pierre Pizon. Presse Med., 70: 1901-4(Sept. 29, 1962). (In French)

The recommended max level for nonidentifiable isotopes for the general population is given by the International Commission on Radiological Protection as 10^{-6} μC/ml water; levels for important isotopes are tabulated. Discussion is given for the methods used for disposal of low activity wastes in rivers and oceans, and the rest as concentrates or solids in the oceans. The disposal of materials in rivers at Mol, Harwell, Chalk River, Marcoule, Fontenay-aux-Roses, and Saclay is discussed. One case is discussed for seas, that of Windscale and Calder Hall on the Irish Sea; the data indicate that more study is needed on waste disposal in such bodies of water. The disposal of radioactive wastes in the oceans requires first that samples be maintained at great depth for avoidance of transport by ocean currents; 19 such troughs deeper than 7000 m are known. Oceanographic studies indicate that at least some of these deep waters are not stagnant, but contain little salt and are highly oxygenated. It appears likely that long-lived samples can rise to upper layers of the oceans and contaminate life. Resistance of radioactive contents to currents and corrosion appears doubtful. The absorption of radioactive isotopes by living organisms is discussed, particularly by plankton from the Marshall Island tests and from the Mediterranean, and by fish and algae in the Irish Sea near Windscale. The strict regulations of the National Bureau of Standards Manual 58 appear quite justified from these studies. (BBB)

858 ^{51}Cr AS A RADIOACTVE TRACER OF COLUMBIA RIVER WATER AT SEA. Osterberg, Charles; Cutshall, Norman; Cronin, John (Oregon State Univ., Corvallis). Science, 150: 1585-7(Dec. 17, 1965).

The plume of the Columbia River was followed 350 kilometers to sea by measurement of its ^{51}Cr content. This radioactive tag, introduced into the river by nuclear reactors at Hanford, Washington, promises to provide a useful oceanographic tool for determining rates of transport and mixing, and for identifying plume waters in the presence of other sources of fresh water.

859 RADIOACTIVE CONTAMINATION OF THE SEA. PROBLEMS IN THE DISPOSAL OF RADIOACTIVE WASTES IN THE SEA. Yasuo Miyake (Tokyo Univ. of Education). Shizen, 18: No. 8, 22-7(Aug. 1963). (In Japanese)

The present state of radioactive contamination of the sea around Japan is described. The concentrations of Sr^{90} and Cs^{137} in the surface waters in 1957 to 1961 are given. These data are compared with those obtained for coastal waters off California, U. S. A. The vertical distribution of these nuclides in the sea is discussed. The disposal of radioactive wastes in the sea requires studies on an economical and strong package of wastes and ecology of marine organisms. Wastes from nuclear-powered ships pose new problems. Behaviors of radioactive substances in the sea are described. The behaviors depend on the following factors: layers in the sea, transport by ocean currents and tidal currents, horizontal diffusion, vertical diffusion, stagnation of coastal water, upwelling, discontinuity among water masses, accumulation of radioactive substances in organisms, transport by organisms, chemical and physical separation, different chemical forms of radioactive substances, ion exchange and adsorption, and changes after sedimentation. If disposal in the sea is necessary, it is important that the disposal must be effected without impairing marine resources. (Nucl. Sci. Abstr. Japan, Vol. 2, No. 4)

860 LIQUID RADIOACTIVE WASTE DISPOSAL AT THE PRETORIA GENERAL HOSPITAL. D. J. Savage, J. Karbendam, P. N. Swanepoel, and T. Fichardt (Pretoria General Hospital). S. African J. Radiol., 2: 11-15(Mar. 21, 1964).

An automatic system is described for the safe disposal of liquid radioactive waste from a radiotherapy institution that is entitled to discharge as much as 5 C of radioactive waste per month. The system operates on the principle of dilution and storage, in three 25,000-gal. retention tanks operating in rotation, until the radioactivity has reached a concentration in the tank ready for discharge that is low enough for discharge into a municipal sewer. (auth)

861 RADIOACTIVE CONTAMINATION OF THE OCEANS. Suschny, O. WMO (World Meteorol. Organ.), Tech. Note No. 68, 123-46(1965).

Average levels of radioactivity and sedimentation, transportation, and accumulation of radioisotopes in the oceans are reviewed. Three main sources of radioactivity in the oceans are primordial radioisotopes, cosmic-ray produced radioisotopes, and artificial radioisotopes (from nuclear explosions and waste disposal). Chemical factors, water movement, and biological processes affect the distribution of radioisotopes. It is concluded that most of the radioactivity in the oceans is of natural origin. Of that part contributed by man, the predominant share comes from nuclear weapon tests. The contribution from waste disposal is insignificant on a world-wide scale, but more research on mixing is needed to assure that no local problems result. (151 references)

862 SOME FACTORS OF IMPORTANCE IN EVALUATING SITES FOR NUCLEAR INDUSTRY AS DETERMINED BY A LIMNOLOGICAL STUDY OF THE UPPER OHIO RIVER. M. A. Shapiro, J. F. Ficke, P. V. Morgan, R. D. Spear, and C. C. Kisiel (Univ. of Pittsburgh). Verhandl. Intern. Ver. Theoret. Angew. Limnol., 15: 299-306(1964).

Factors important in the design and siting of a nuclear industry to keep the levels of radioactivity in the flowing streams below prescribed limits are discussed. At the same same time some of the natural factors that might be utilized by nuclear industry to reduce costly overdesign of waste-treatment facilities are pointed out, and the problems involved in the disposal of radioactive wastes to fresh flowing streams and their watersheds are considered.

863 GUIDING RADIOACTIVE SEWAGE INTO SURFACE WATERS. [PART] II. Wasser, Luft Betrieb, 8: 469-73[1964].

Information presented during a specialized conference on the subject of radioactive contamination of surface waters and treatment and monitoring of liquid radioactive wastes

from atomic plants and radioisotope laboratories is summarized. (Excerpta Med., Sect. XXIII, 2: No. 3, (Mar. 1965))

864 DEFENCE REQUIREMENTS FOR PREDICTING DISPERSAL OF RADIOACTIVE CONTAMINANTS IN TIDAL INLETS. S. H. Small (Norwegian Defence Research Establishment, Kjeller, Norway). p.167-76 of "Nuclear Detonations and Marine Radioactivity." Kjeller, Norway, Norwegian Defence Research Establishment, 1963.

In view of the limitations of dose/effect relationships, simple mathematical models should suffice for predicting exposure dose-rates from contaminants released into tidal inlets. Models already developed for predicting mean concentrations of dissolved contaminants in estuaries can be of great assistance, but attention must also be paid to the hazards of contaminants not in solution.

865 RADIOAKTIVNAYA ZAGRYAZNENNOST' MOREI I OKEANOV. (Radioactive Contamination of Seas and Oceans). N. I. Baranov and L. M. Khutrov, eds. Moscow, Publishing House "Science," 1964. 224p.

Twenty-five articles are presented on radioactive contamination of seas and oceans by waste disposal fallout, and natural radioactivity. The results are given of measurements taken in the Indian and Pacific Oceans and the Black Sea during 1959–1960. Radiobiological effects on marine organisms are discussed. Methods for water sampling at various depths and radioactivity measurement are described. (R.V.J.)

866 REMOVAL OF LIQUID RADIOACTIVE WASTES FROM THE LABORATORY. E. Gashpar, and P. Shandru (Inst. for Atomic Physics, Bucharest). Symp. Tech. Zagadnien Ochrony Promieniowaniem, Warsaw, 435-45(1962). (In Russian)

The radioactive wastes from a laboratory can be conducted in a diluted state in the public sewers under the assumption that their specific concentration and their general daily activity will not exceed a given value. If these values are exceeded, the wastes must be stored in special vats until the activity has reached a permissible value. When $T_{1/2}$ > 15 days, the radioactive wastes are filtered through a series of filters before they are released to the public sewers. (tr-auth)

867 THE DISPERSION OF RADIONUCLIDES IN OPEN CHANNEL FLOW. Calvin Clyde Patterson. Thesis, Austin, Tex., Univ. of Texas, 1963. 107p.

The Colorado River and the Pierce Ranch Canal were selected as test reaches. The fluorescent organic dyes, Rhodamine B and Fluorescein, were chosen as tracers and detection was accomplished with a Turner model 111 fluorometer. Tracer releases were made instantaneously at midstream. Longitudinal dispersion was measured in both the Colorado River and the Pierce Canal, and transverse dispersion was measured in the Colorado River. Successful sampling was accomplished with portable continuous flow equipment operated in a small aluminum boat. Instantaneous injection was approximated with an injector operated with compressed nitrogen. Equations were derived, which made possible the calculation of the true longitudinal (D_x) and transverse (D_y) dispersion coefficients with concentration-time data obtained from two-dimensional tracer plumes. A comparison of D_x calculated with Thomas' dispersion equation was made with measured values of D_x. The calculated coefficient was within the scatter of the measured coefficients. An empirical dispersion coefficient formula, $D_x = 0.8e^{0.034 U\sqrt{A}}$ was derived from data obtained in four different channels, where A is the cross-sectional area of the channel. Dispersion coefficients measured were: D_x = 14.4 ft^2/sec, Colorado River; D_x = 1.14 ft^2/sec, Pierce Canal; and D_y = 0.55 ft^2/sec, Colorado River. The data collected in seven longitudinal tests and two transverse tests are presented in summary tables and recorder strip charts. (Dissertation Abstr., 24: No. 12, Pt. I (June 1964))

Storage

See also the following references:

13, 18, 19, 24, 26, 31, 34, 35, 49, 79, 83, 87, 92, 94, 95, 98, 104, 122, 123, 130, 133, 135, 140, 141, 147, 155, 162, 171, 193, 195, 213, 220, 233, 234, 238, 244, 249, 253, 257, 321, 517, 528, 536, 683, 983

868 CHEMICAL PROCESSING PLANT, WASTE STORAGE TANKS (Engineering Materials). (Idaho Operations Office, AEC). (CAPE-1147)

Thirty-one drawings are included for 300,000 gal tanks used to store liquid wastes. The tanks are 50 ft ID, 21 ft high, and are constructed of 304L and 347 stainless steel. Domes 8 ft 6 in. in height and constructed of 304L stainless steel are located on top of each tank. The tanks are cooled with heavy wall pipe coils, which hold the waste temperature between 50 and 20°C to limit corrosion and precipitation. The tanks are equipped with secondary containment sumps, emergency vent condensers, alarms, and transfer jets in case of leaks. The tanks have level, specific gravity, temperature, and pressure monitors with appropriate alarms. Activity levels in the storaged wastes range from 10 to 200 β curies and 3 to 60 γ curies per gal. No leaks from storage tanks or piping systems have been detected. (P.C.H.)

869 (CEA-R-2894) DISPOSITIF DE RECUPERATION D'EFFLUENTS RADIOACTIFS. (Device for Recovering Radioactive Waste). Mas, Roland (Commissariat a l'Energie Atomique, Saclay (France). Centre d'Etudes Nucleaires). Nov. 1965. 9p. Dep. mn.

A new system is presented, which consists essentially of a standard reservoir of stainless steel fixed in a concrete well lined with stainless steel. Each reservoir has a standard lid on which are placed all the accessory devices, which are standardized. The new reservoirs offer a great degree of security, and are interchangeable; the wells are easy to decontaminate and it is possible to work in a well without any danger of the personnel being irradiated by neighboring reservoirs.

870 (DP-966) TAFI-II, A COMPUTER CODE FOR MANAGEMENT OF RADIOACTIVE WASTE. Tharin, Catherine R. (Du Pont de Nemours (E. I.) and Co., Aiken, S. C. Savannah River Lab.). Aug. 1965. Contract AT(07-2)-1. 42p. Dep. mn; CFSTI $2.00 cy, $0.50 mn.

A computer program, TAFI-II, was written in FORTRAN II to assist in the regulation and inventory control of radioactive fission products that are stored in waste tanks at the Savannah River Plant. The inventories and heat generation rates are computed for thirteen isotopes in each of twenty-four waste tanks. The program prepares a report that summarizes the data and provides an up-to-date status of the conditions in the waste tanks. (auth)

871 (DP-990) LEAKS IN RADIOACTIVE-WASTE TANKS. Girdler, R. M. (Du Pont de Nemours (E. I.) and Co., Aiken, S. C. Savannah River Lab.). Dec. 1965. Contract AT(07-2)-1. 30p. Dep. mn. CFSTI $2.00 cy, $0.50 mn.

A number of leaks developed in four of sixteen underground storage tanks used for storage of alkaline radioactive waste from the separations area at the Savannah River Plant. The tanks are 75 to 85 feet in diameter and were field erected by welding; the welds were not stress relieved. The investigation of these leaks revealed no correlation with the history of the construction or usage of the tanks. Stress corrosion cracking is the most likely explanation of the leaks. Similar stress corrosion cracking in tanks of this type can be prevented by full stress relieving after welding has been completed.

872 (EIR-56) DIE LAGERUNG VON RADIOAKTIVEN ABFAELLEN IM EIDGENOESSISCHEN INSTITUT FUER REAKTORFORSCHUNG IN WUERENLINGEN. (Storage of Radioactive Wastes at the Federal Institute for Reactor Research in Wurenlingen). K. F. Eschle (Switzerland. Eidgenössisches Institut für Reaktorforschung, Würenlingen). Aug. 1963. 16p.

The storage of radioactive wastes is discussed. Storage buildings and tanks were built on the basis of an estimated total volume of 3000 to 4000 m^3, corresponding to an activity of about 20,000 C. (M.C.G.)

873 (HW-79522) REMOTE WELDING CLOSURE FOR RADIOACTIVE WASTE STORAGE CONTAINERS. Final Report. V. L. Hammond (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Dec. 13, 1963. Contract AT(45-1)-1350. 26p.

Studies were conducted on welding techniques for remote closure of high-level radioactive waste storage containers. The effects of welding variables were studied. The tungsten inert gas welding process was found to be satisfactory for producing high-integrity closures on stainless steel containers. (D.L.C.)

874 (HW-80848) THERMAL CONSEQUENCES OF LEAKS IN RADIOACTIVE WASTE STORAGE TANKS. G. Jansen, Jr. (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Mar. 2, 1964. Contract AT(45-1)-1350. 12p.

An analytical solution for determining the consequences of the leak of radioactive heat-generating solutions from storage tanks into soil was developed. As an example problem the case of a leak of Purex 241-A supernate containing $5 \times 10^{-4}\underline{M}$ cesium (0.18 Btu/hr/gal) was considered. The calculations indicated that for a leak of 5000 gallons the maximum steady-state rises encountered are 14 and 62°F at the concrete-soil interface and in the soil at a point 4.2 feet from the concrete liner, respectively. For a 50,000 gallon leak comparable temperature rises are 29 and 301°F, the latter at a point 10 feet from the concrete. It was concluded that even relatively large supernate leaks from the Purex waste storage tanks do not constitute a thermal hazard. A correlation is presented, which allows the calculation of maximum steady state temperatures for hemispherical leaks of solutions containing fission products of varying activity and adsorbability on the soil. The method of solving this problem is presented and it may be useful in determining temperature distributions near tanks, pipes, and other radioactive waste solution sources in the ground. (auth)

875 (HW-84603(p.37-49)) WASTE STORAGE AND TREATMENT. G. J. Alkire, R. E. Burns, and W. H. Swift (General Electric Co., Richland, Wash. Hanford Atomic Products Operation).

Studies were initiated on the precipitation of sulfate from a simulated waste. A 50-day run was made using an electrodeionization unit with an alkaline condensate. It was found that fission product cesium loading in shipping casks may be increased by using AW-500 zeolite in place of the Decalso. The relationship between zeolite SiO_2/Al_2O_3 ratio and Cs and Sr selectivity was determined for several sodium-based zeolites. Vapor pressure vs temperature and concentration data were obtained for three waste compositions which will be subjected to in-tank concentration. A number of special cases in the removal of heat from underground storage vessels were studied. An approximate model was used to calculate maximum tank temperatures in Purex Waste Tanks containing solidified wastes. (D.L.C.)

876 (ORNL-2873) THE COSTS OF PERMANENT DISPOSAL OF POWER-REACTOR FUEL-PROCESSING WASTES IN TANKS. Blomeke, J. O.; Frederick, E. J.; Salmon, R.; Arnold, E. D. (Oak Ridge National Lab., Tenn.). Sept. 1965. Contract W-7405-eng-26. 118p. Dep. mn; CFSTI $4.00 cy, $0.75 mn.

The costs of permanent storage of reactor fuel processing wastes are estimated for a 22,400-Mw(e) nuclear power economy. The wastes arise from chemical dissolution of the fuel cladding and solvent extraction purification of a combination of uranium and thorium converter fuels. The economics are examined for three representative types of financing: government ownership, private ownership, and a combination of government and private ownership. Total costs for storing raffinate waste, in terms of the cost of electricity generated, were least for storage of acid wastes in tanks of 75-yr life with government ownership, and greatest for storage of alkaline wastes in tanks of 25-yr life with private ownership. For acid wastes, the optimum tank capacity is about a million gallons, and minimum total costs range from 0.0165 to 0.0272 mill/kwh. For alkaline wastes, the optimum tank capacity is 2.5 million gallons, and minimum total costs range from 0.0177 to 0.0294 mill/kwh. Doubling the concentration of fission products in the wastes, as stored, reduced the total costs between 15 and 30%. Total costs for storing cladding wastes ranged from 0.0215 to 0.0398 mill/kwh. These costs were greater than those for raffinate storage because of the greater volumes of cladding wastes. (P.C.H.)

877 (ORNL-3578(p.78-95)) PROCESS INSTRUMENTATION DEVELOPMENT. (Oak Ridge National Lab., Tenn.).

Tank-level telemetering from 24 intermediate level waste tanks to Building 3105 for the Operations Division was updated to provide greater accuracy with less maintenance and to utilize a minimum of leased telephone lines for the system.

878 (ORNL-P-1926) ESTIMATED COSTS OF HIGH-LEVEL WASTE MANAGEMENT. Blomeke, J. O.; Salmon, R.; Roberts, J. T.; Bradshaw, R. L.; Perona, J. J. (Oak Ridge National Lab., Tenn.). [nd]. Contract W-7405-eng-26. 21p. (CONF-660208-1). Dep. mn. CFSTI $1.00 cy, $0.50 mn.

From Symposium on Solidification and Long-Term Storage of Highly Radioactive Wastes, Richland, Wash.

The costs of perpetual tank storage of both acid and alkaline wastes were estimated as a function of tank size, tank life, and fission product concentration in the waste for three representative types of financing. For acid wastes, the optimum tank capacity is about a million gallons, and minimum total costs range from 0.0165 to 0.0272 mill/kwhr. For alkaline wastes, the

optimum tank capacity is 2.5 million gallons, and minimum total costs range from 0.0177 to 0.0294 mill/kwhr. Reducing the volume of wastes, as stored, by doubling the concentration of fission products reduces the total costs between 15 and 30%. The minimum total costs for management by the series of operations consisting of interim liquid storage, pot calcination, interim solid storage, shipment, and disposal in salt mines, are estimated to range from 0.017 to 0.020 mill/kwhr when carried out over a 30-year period.

879 (RL-SA-62) STORAGE FACILITIES FOR HANFORD HIGH-LEVEL FISSION WASTE CONTAINERS. Barnes, J. W. (Isochem Inc., Richland, Wash.). [1966]. Contract AT(45-1)-1851. 22p. (CONF-660208-2). Dep. mn. CFSTI $1.00 cy, $0.50 mn.

From Symposium on Solidification and Long-Term Storage of Highly Radioactive Wastes, Richland, Wash.

A facility designed for the surveillance and long term storage of high level waste containers produced under the Hanford Waste Management Program is described. This facility is located in an existing canyon type structure to assure maximum containment. A closed loop high purity water system with dual heat exchangers and circulating pumps is provided. Equipment and container handling are designed for remote operation and maintenance. Instrumentation is provided for monitoring and surveillance of fission waste containers. Flexibility exists for safely handling containers and waste streams in the event of a container failure. Alternative storage vaults and cooling systems are also considered.

880 COMPARISON OF THE ECONOMIC ASPECTS OF THE TREATMENT AND STORAGE OF FISSION PRODUCTS FROM INSTALLATIONS PROCESSING IRRADIATED NATURAL URANIUM. Thiriet, L. (Commissariat a l'Energie Atomique, Paris); Lesur, P.; Giraud, R.; Wanlin, J. pp 571-83 of STI-PUB-110. (In French).

Policies adopted for the management of fission product effluents are summarized, and the essential technical features of the stainless steel storage installations (in acid medium) used in France are described, with breakdown of the cost involved. The dependence of these costs on the activity of the solutions stored and on their heat release is shown. The cost of storage in terms of feasible tank size is also discussed. The optimum size for such tanks and the rate at which they are placed in service were determined in relation to the characteristics of the fission products to be stored and to the respective capacities of the installations for processing irradiated natural uranium. It is shown that although storage costs depend on the assumptions made regarding the useful life of installations, rates of interest, and operating costs, optimum policy decisions (as to size and rate of entry into service) are independent of these assumptions, being determined solely by the variation in the cost of tanks, according to size. These are the factors that enter into the optimized cost of storing these effluents indefinitely. This method of indefinite storage was compared with other possibilities of fission product management, e.g., vitrification and separation of Cs and Sr. Economics of the various methods are discussed, and some possible long-term solutions are summarized.

881 STORAGE OF RADIOACTIVE FISSION PRODUCT SAMPLES. William J. Maeck and James E. Rein (Phillips Petroleum Co., Idaho Falls, Idaho). Anal. Chem., 36: 252-3(Jan. 1964).

To evaluate the storage of dilute fission product solutions, the degree of absorption of 1-day cooled and 1-year cooled fission products was measured for two different media in four container materials over periods of time up to 1 month. The selected media were nitric and hydrochloric acids. Acid concentration was varied over the range of 1 to 10 M. The containers were borosilicate glass, wax-coated glass, Vycor glass, and polyethylene. Borosilicate glass was found to be superior for 1-day cooled fission product solutions. The adsorption of 1-year cooled gamma-emitting fission products was much less than that of young fission products. Polyethylene is the preferred storage material and hydrochloric acid-hydrogen peroxide is a better medium than nitric acid. (P.C.H.)

882 THE STORAGE OF RADIOACTIVE WASTE IN THE JULICH NUCLEAR RESEARCH CENTRE. K. H. Rattay and R. Schulze-Rettmer (Kernforschungsanlage, Julich, Ger.). Atomwirtschaft, 9: 617-19(Dec. 1964). (In German)

With the coming into operation of the first Institute of the Jülich Nuclear Research Center the necessity arose for a temporary, small storage place for weakly radioactive waste. For this purpose a small waste store with 60 m^2 surface was built and put into operation. Screening measures against gamma rays were not involved. Small quantities of waste of higher activity could be stored in safety either by screening with lead bricks or by the use of screened waste containers. To extend this small store a final store was constructed whose capacity should suffice for the next 5 to 6 years. This store, primarily intended for the Jülich Nuclear Research Center, was simultaneously to serve as a collecting place for radioactive waste for North-Rhine Westphalia. The new store is of a hangar construction of pre-cast, reinforced concrete. Ten cells of standard concrete open at the top and each with a base area of 100 m^2 serve as storage areas. To provide safety in case of earthquakes each of the cells is an individual building unit. The inner walls have a total thickness of 60 cm, the outer walls bounding the service passages are 70 cm thick. Inside the large storage cells are four smaller cells for the reception of waste with a high gamma activity. The waste is collected in barrels and subsequently stored in the cells. After completion of the decontamination plant, which is to be built during the next two years, the weakly radioactive waste already in store will be concentrated by incineration or by pulverization and compression. The realization of the waste store permits the storage of gamma emitters with an activity of up to about 1 million curies. The total cost of the erection of the store inclusive of the transporting arrangements and the measuring equipment amounted to DM 1.9 million. The storage capabilities of this Land collection center should suffice for all radioactive waste until such time as the problem of a central store for the Federal Republic has been solved. Then the radioactive waste from all the intermediate storage places will be transferred to a final store. (auth)

883 DETERMINATION OF CONCENTRATION OF WASTE WATER FROM LOW-ACTIVITY INSTALLATIONS AND LIQUID WASTE STORAGE TANK. T. Sekiny. Genshiryoku Kogyo, 10: No. 1, 54-7(1964). (In Japanese)

Experiments were carried out to demonstrate that process water may be safely stored in tanks. Becasue of the relatively high permissible concentrations of ^{35}S, ^{131}I, and ^{32}P, process water from equipment where these isotopes are present need not be treated prior to storage. Water from equipment used with long-lived isotopes may be higher than permissible. The possibility of leakage into the ground through the walls of concrete tanks, both unlined and lined with epoxy resin, was studied. The first tank contained ^{32}P in a concentration 200 times the MPC in water, and although water seeped through, no activity appeared outside. No leakage occurred with the second type, and, because ^{32}P would precipitate along with impurities, the ^{32}P concentration would be greater near the bottom. (IIC-M.G.)

884 HEAT GENERATION BY RADIOACTIVE WASTES. E. F. Gloyna (Univ. of Texas, Austin), H. J.

Longley, and C. W. Christenson. J. Sanit. Eng. Div., Am. Soc. Civil Engrs., 90: 43-66(Feb. 1964).

Generalized numerical and analytical methods were developed for estimating temperatures resulting from the storage of high-level wastes. Parametric relations were considered; and solutions for the packed-bed, multi-region, variable-geometry, and variable-effective thermal conductivity systems were obtained. Solutions of the heat equation in one space dimension were obtained for plane, cylindrical, and spherical geometries. Three computer codes were developed, tested for accuracy, and compared. The codes represent three approximations to the actual physical situation. An auxiliary code was written to provide an estimate of the waste that could be put into a cavity when a maximum permissible temperature is specified. The accuracy of the solutions depends chiefly on the values of the input data and the variations of the parameters affecting the finite mesh and boundary conditions. The size of storage sites, the concentration of wastes, and the age of wastes have the greatest effect on the magnitude of the temperature. (Publ. Health Eng. Abstr.)

885 ON THE QUANTITY OF RADIOISOTOPES IN LIQUID WASTE AND SOME PROBLEMS WITH REGARD TO THE STORAGE TANK OF LOW LEVEL WATER. K. Hirose (Tokyo Univ.), M. Sago, et al. Kuki Chowa, Eisei Kogaku, 37: 444-50(June 1963).

Two concrete waste storage tanks with 10-cm wall thicknesses were fabricated. One was unlined; the other was lined with epoxy-resin paint. The tanks were filled with water containing P^{32} and I^{131}, respectively, and the leakage was monitored. After two months the amount of isotope adsorbed in the concrete wall was examined. In the first tank, though the seepage of water was found, no activity existed on the outside surface. However, examination of the concrete showed the activity reached almost to the outer surface. In the second tank no leakage was observed. It was concluded that tanks for liquid waste storage higher than MPC should be lined with some materials such as plastic paint. (auth)

886 DEVELOPMENT, CONSTRUCTION, AND OPERATION OF THE CENTER FOR RADIOACTIVE WASTE AND SEWAGE AS AN EXAMPLE FOR STUDY OF A SOLUTION TO THE PROBLEM OF RADIOACTIVE WASTE IN A COUNTRY WITH HIGH POPULATION DENSITY. H. Weiss. Symp. Tech. Zagadnien Ochrony Promieniowaniem, Warsaw, 468-75(1962). (In German)

In 1959, the Center for Radioactive Waste and Sewage was established to study the problem of radioactive waste disposal in a country with high population density. The present structure of the Center and the distribution of studies to the various divisions are reported. The organization of the transportation service and the storage of radioactive wastes are described. The Lohmen project and the results obtained from it up to the present are reported. The international connections of the Center are discussed. (tr-auth)

Subterranean

See also the following references:

7, 19, 24, 31, 34, 35, 36, 41, 97, 98, 106, 122, 135, 185, 196, 210, 214, 215, 220, 222, 226, 233, 235, 242, 244, 247, 249, 878, 884, 979, 980

887 (A/CONF.28/P/783) SUBTERRANEAN DISPOSAL OF RADIOACTIVE WASTE IN CAVITIES MADE IN SALT FORMATIONS. G. Richter-Bernburg (Germany. Bundesanstalt fur Bondenforschung, Hanover). May 1964. 16p.

Only one kind of rock is truly impermeable to ground water: halite or rock salt. Some of the physical and geological properties of halite are described, and it is shown that domes or diapirs are better for waste storage than table formations. The creation of cavities in salt deposits for waste storage is considered, and some salt formations in Europe suitable for such applications are discussed. (D.L.C.)

888 (AEC-tr-6544) GEOLOGICAL REPORT ON THE SUITABILITY OF MINE CAVITIES IN THE ASSE II ROCK SALT MINE FOR THE DISPOSAL OF RADIOACTIVE WASTES. E. Hofrichter (Bundesanstalt für Bodenforschung, Hanover (West Germany)). Translated from report, File No. 3278/63. Nov. 26, 1963. 13p. Dep.(mn); $1.60(fs), $0.80(mf) JCL.

The geological features of the Asse I, II, and III mines of the Burbach Kaliwerke AG. are described. The mine spaces and working spaces in Asse II are discussed, and its pressure phenomena and hydro-geological situation are described. The applicability of the cavities to storage of radioactive wastes is discussed in detail. The question of long-term stability is considered. (D.L.C.)

889 (AEC-tr-6549) REPORT ON THE PROBLEM OF THE POSSIBILITY OF THE UNDERGROUND DISPOSAL OF RADIOACTIVE WASTES. (Bundesanstalt fuer Bodenforschung, Hanover (West Germany)). Translation. [nd]. 77p. Dep.(mn); $7.60(fs), $2.51(mf) JCL.

Studies were carried out on the possible underground storage of large amounts of wastes in West Germany. Both aquiferous and cavern depots were considered, and various possible places in Germany are discussed. Salt deposits, masses, and structures are described in some detail. (D.L.C.)

890 (DP-844) STORAGE OF RADIOACTIVE WASTES IN BASEMENT ROCK BENEATH THE SAVANNAH RIVER PLANT. R. J. Christl— F. H. Springer, ed. (Du Pont de Nemours (E. I.) & Co. Savannah River Lab., Aiken, S. C.). Mar. 1964. Contract AT(07-2)-1. 103p.

The ground beneath a three-square-mile area of the Savannah River Plant was explored to determine the attitude, continuity, types, and character of the basement rock; the hydrology of the basement rock; and the hydrology and character of the overburden. These studies show that radioactive waste solutions can be stored safely in the basement rock beneath the Savannah River Plant. A conceptual design was developed for constructing a 20,000,000-gallon, or larger, storage facility 1300 to 1700 feet beneath the ground. (auth)

891 (IDO-22046) HYDROLOGY OF SUBSURFACE WASTE DISPOSAL NATIONAL REACTOR TESTING STATION, IDAHO. Annual Progress Report, 1963. D. A. Morris, W. E. Teasdale, et al. (Geological Survey, Idaho Falls, Idaho). May 1964. 228p. Dep.(mn); $6.00(cy), 5(mn) CFSTI.

Results of investigations of the NRTS geological area are presented. Special studies are reported concerning

waste disposal. It was found that ground water underlying NRTS is part of the larger ground-water body underlying the entire Snake River plain. NRTS is underlain by basalt and unconsolidated sediments. Study during 1963 was devoted to determining chemical or radiometric changes that occurred as ground water containing radioactive waste moved through the geologic environments underlying the NRTS area. Results of water level studies and test hole sample analyses are included. (J.R.D.)

892 (IDO-22047-USGS) HYDROLOGY OF SUBSURFACE WASTE DISPOSAL—NATIONAL REACTOR TESTING STATION, IDAHO. Annual Progress Report, 1964. Morris, D. A.; Barraclough, J. T.; Chase, G. H.; Teasdale, W. E.; Jensen, R. G. (Geological Survey, Idaho Falls, Idaho). May 1965. 304p. Dep. mn; CFSTI, $7.00 cy, $1.50 mn.

Research was conducted on the hydrologic aspects of the disposal of low-level aqueous waste to the NRTS environment. Geologic and geohydrologic data were analyzed, and some new data are reported. Studies were made of the TRA waste-disposal pond area. The effects of the Alaskan earthquake on the water levels are described. (D.L.C.)

893 **(NP-15859) HYDRODYNAMICS OF THE DISPOSAL OF LOW-LEVEL LIQUID RADIOACTIVE WASTES IN SOIL. Final Report. Mackrle, Vladimir; Dracka, Oldrich; Svec, Jan (Czechoslovak Academy of Sciences, Prague. Inst. of Hydrodynamics). 1965. IAEA Contract No. 98. 142p. Dep. mn.**

The general aspects of liquid flow carrying suspension through a porous medium was studied in order to obtain data that might serve in evaluating the applicability of injection for the disposal of the radioactive sludge. Theoretical and laboratory studies were made on: injection of diluted suspension; injection of concentrated sludge; and two-dimensional filling of geological formations. The results offer a theoretical basis for the formulation of the requirements for a suitable geological structure as well as for the technology of injection.

894 (ORNL-3697(p.3-10)) LIQUID INJECTION INTO DEEP PERMEABLE FORMATIONS. D. G. Jacobs and M. U. Shaikh (Oak Ridge National Lab., Tenn.).

Movement of radionuclides through porous media is due to movement of the transporting water, but is also affected by the ion exchange properties of the medium. Calculations were made for the movement of the solution and radiostrontium through a slab of Berea sandstone and were confirmed by experimental data. The behavior of chloride as a water tracer is complicated by its repulsion by negatively charged soil colloids. The relative exclusion of chloride is especially complicated when clay minerals are present and restricted in their degree of lattice expansion. However, these effects would be decreased as the concentration of electrolyte is increased and would probably be rather insignificant in clean sandstones having low specific surface and high porosity, especially if most of the surfaces were sufficiently separated to permit full development of an electric double layer. Anion exclusion gives a good measure of lattice expansion and collapse of hydrobiotite systems, thereby providing a rather accurate method for measuring the distribution of surface charge density of these interstratified materials. (auth)

895 (ORNL-3697(p.10-19)) DISPOSAL BY HYDRAULIC FRACTURING. Wallace de Laguna, R. C. Sexton, E. G. Struxness, T. Tamura, and H. O. Weeren (Oak Ridge National Lab., Tenn.).

Water injections were made in the injection well at 986 and 966 ft. These tests confirmed that a fluid loss additive is not a necessary component of the mix to be used in the waste injections. On the last water injection, the sand-erosion technique for slotting failed to cut the casing after 20 min at a jet pressure of 2600 psi. An estimate of the probable operating cost of the shale fracturing plant shows that the cost of the mix and the well life are the significant variables. The well life can be significantly extended by making the batch size large. The Densometer controlled the density of the slurry to within 0.1 lb/gal of the desired density, which is entirely acceptable. Following construction of the surface plant, a series of five injections was made to determine the performance of the plant and the underground behavior of waste slurries of different compositions. In general, the operation of the surface plant was smooth and satisfactory. (auth)

896 (SAN-413-2) RADIOACTIVE WASTE-DISPOSAL POTENTIALS IN SELECTED GEOLOGIC BASINS—A RECONNAISSANCE STUDY. (American Assn. of Petroleum Geologists, Inc., Midland, Tex.). June 26, 1964. Contract AT(04-3)-413. 44p. Dep.; $2.00(cy), 1(mn) CFSTI.

The advantages of deep permeable formations for the storage of large volumes of low-level wastes from a fuel reprocessing plant, are pointed out. Descriptions are presented of the subsurface geology and hydrology of six basin provinces for the preliminary evaluation of their use for waste storage. The basins studied include the Appalachian, Valley-and-Ridge Province, Michigan, Salina, Denver, and San Juan Basins. (D.L.C.)

897 GEOLOGIC, HYDROLOGIC, AND SAFETY CONSIDERATIONS IN THE STORAGE OF RADIOACTIVE WASTES IN A VAULT EXCAVATED IN CRYSTALLINE ROCK. Proctor, J. F. (E. I. du Pont de Nemours and Co., Aiken, S. C.); Marine, I. W. Nucl. Sci. Eng., 22: 350-65(July 1965). (TID-20627)

A recent investigation established the technical feasibility and indicated the high degree of safety that could be afforded by the storage of high-level radioactive wastes in unlined vaults excavated in crystalline rock 1500 ft beneath the surface of the Savannah River Plant near Aiken, S. C. The crystalline rock at the proposed site is covered by 1000 ft of unconsolidated sediments consisting predominantly of sand and clay. A virtually impermeable layer of clay separates the rock from the overlying sediments in which several prolific water-bearing zones occur. The separation of the waters above and below this clay layer is confirmed by their different chemical composition and by the presence of dissolved helium-bearing gas only in the water in the rocks beneath the clay. Based on geologic and hydrologic information obtained in an intensive drilling and testing program, upper limits on the rates of water movement through the crystalline rock are calculated to be 1.5 to 7 ft/year, depending upon the degree of fracturing of the rock. Comparable data on the unconsolidated sediments lead to a calculated maximum rate of water movement of 350 ft/year. The most significant driving force for the migration of radionuclides from the storage site is derived from the natural water movement, coupled with effects due to dispersion and ion exchange. Characteristics of the waste, heat generation, and radiolysis have, by contrast, only small effects on migration. Three barriers prevent migration of the radionuclides: the very low permeability of the rock in which the storage vault is located, the virtually impermeable clay layer separating the rock and sediments, and the ion exchange properties of the sediments. Any one of these barriers is capable of confining the radionuclides well within the plant boundaries for a time much greater than the 600-year period required to render the wastes innocuous. (auth)

898 PROBLEMS IN DISPOSAL OF RADIOACTIVE LIQUID WASTES IN SALT. Kubota, Hisashi (Oak Ridge National Lab., Tenn.). Chem. Eng. Progr., Symp. Ser., 60: No. 53, 68-73(1964).

The disposal of radioactive liquid wastes in salt, and its problems are discussed. Laboratory studies on the cold

chemistry of the system, the role of radiation, alteration of cavity geometry, and results of field experiments using simulated waste solution in salt models and in 3000-gal. cavities are considered. (F.J.H.)

899 PIPE RADIOACTIVE WASTES TO SHALE BEDS. H. O. Weeren (Oak Ridge National Lab., Tenn.). Heating, Piping, Air Conditioning, 36: 122-3(Nov. 1964).

Piping employed in a radioactive waste disposal method, in which waste liquids are piped and injected into shale beds, is described. The experiment consists of injecting several 40,000 gal batches of intermediate level waste solution, mixed with cement and other additives, into a shale formation at a depth of approximately 900 ft. The mixer, surge tank, injection pump, and wallhead valving are installed in cells to reduce radiation exposure to operators. The process requires handling of radioactive and nonradioactive solutions at low pressure (up to 300 psi), slurries of sand or cement at high pressure (up to 7500 psi), and bulk cement. The greater part of the waste transfer line is cast iron pipe with mechanical joints. Some of it is welded carbon steel. (M.C.G.)

900 GROUND DISPOSAL. ITS ROLE IN THE U. S. RADIOACTIVE WASTE MANAGEMENT OPERATIONS. W. G. Belter (Atomic Energy Commission, Washington, D. C.). Colloq. Intern. Retention Migration Ions Radioactifs Sols, Saclay, 1962, 3-10(1963).

Ground disposal in the U. S. is reviewed. Some of the most important aspects of the Hanford operation are considered briefly, and the monitoring program required is outlined. Three types of geologic formations that may be technically feasible for the discharge of liquid radioactive wastes are recommended for investigations: salt dome or bed; deep permeable formations containing connate brine; and excavation in impermeable formations. Development work accomplished to date in this field is summarized. Basic geochemical and geophysical studies have been developed to obtain better understanding of fission products movement in the ground in order to provide a basis on which sound waste disposal practices can be established. (P.C.H.)

901 ION EXCHANGE IN THE DEEP-WELL DISPOSAL OF RADIOACTIVE WASTES. D. G. Jacobs (Oak Ridge National Lab., Tenn.). Colloq. Intern. Retention Migration Ions Radioactifs Sols, Saclay, 1962, 43-54(1963)

In order to evaluate the safety of injection of radioactive wastes into deep wells it is necessary to be able to predict the movement of the more hazardous radionuclides. Exchange of monovalent and divalent cations on numerous reference clay minerals and sandstone showed that the exchange potentials of the divalent alkaline earth cations are quite similar. Strontium is selectively sorbed over calcium by a factor of 1.25. Divalent cations are sorbed in preference to sodium by the clay minerals and the exchange is adequately described by the laws of mass action when the divalent cations occupy more than 10% of the total exchange sites. The sorption of trace quantities of strontium from a calcium-free sodium salt solution is greater than predicted for simple ion exchange, suggesting the presence of small quantities of sorption sites highly selective for divalent cations. Normalized strontium breakthrough curves coincide with normalized chloride breakthrough curves and show that radionuclide migration can be predicted by observing the dispersion of water in the disposal formation and correcting for the sorptive behavior of specific radionuclides. Exchange studies with Richfield sand, a typical deep-well disposal formation, suggest that in the disposal of ORNL low-level and intermediate-level wastes strontium will move only 1 and 10% as fast, respectively, as the transporting water due to ion exchange; calcium-strontium precipitation reactions will provide additional restrictions to radiostrontium movement. Cesium would move much more slowly than strontium because of highly selective exchange reactions. Ruthenium may move rapidly but is not likely to be hazard controlling. (auth)

902 PRINCIPLE OF "GROUND DISPOSAL INJECTION" OF RADIOACTIVE SLUDGE. C. Caron, C. Gailledreau, and Y. Belot (Centre d'Etudes Nucleaires, Saclay), France). Colloq. Intern. Retention Migration Ions Radioactifs Sols, Saclay, 1962, 167-80(1963). (In French)

Research on radioactive sludges and cement mixed grouts injected according to miscellaneous techniques is described. The characteristics of several grout studied in terms of the kind of sludge incorporated and of the ground to be injected are given along with grout behavior in lixiviation by natural waters. (P.C.H.)

903 RESEARCH BY THE UNITED STATES GEOLOGICAL SURVEY RELATED TO THE MOVEMENT OF RADIONUCLIDES IN THE EARTH ENVIRONMENT. Alfred Clebsch, Jr., and Alan E. Peckham (Geological Survey, Washington, D. C.). Colloq. Intern. Retention Migration Ions Radioactifs Sols, Saclay, 1962, 189-97(1963).

Geologic and hydrologic research by the U. S. Geological Survey related to the movement or retention of radionuclides embraces several scientific disciplines, a wide variety of geohydrologic terranes, and various types of radioactive materials. Field studies have been completed or are in progress at all the major nuclear energy installations in the United States where large releases of radioactivity have been made, either in the form of radioactive wastes or as the consequence of nuclear explosions. Much of this work is designed to develop an understanding of the influences of geologic and hydrologic features on radionuclide movement. Chemical studies include supporting services for field projects as well as research on reactions between radioactive substances and earth materials. Descriptive geochemical research on the distribution of stable elements, such as strontium, and naturally occurring radioactive elements, such as radium and uranium, has been a small but important part of the work. Theoretical and laboratory investigation of the basic mechanisms of fluid movement as they affect the transport and dispersion of radionuclides has emphasized the effects of the abundant departures in nature from idealized systems. Work related to the development of new techniques for disposal of wastes, such as deep-well injection, ultimate storage of fluids in cavities mined in rocks of low permeability, and storage of solids or fluids in salt, ranges from the compilation of broad summaries of the geology of geologic basins to detailed geologic and hydrologic investigations at the particular site. 111 references are included. (P.C.H.)

904 SELECTIVE ION EXCHANGE REACTIONS FOR CESIUM AND STRONTIUM BY SOIL MINERALS. Tsuneo Tamura (Oak Ridge National Lab., Tenn.). Colloq. Intern. Retention Migration Ions Radioactifs Sols, Saclay, 1962, 95-104(1963).

The basis for discharge of low and intermediate activity wastes into the ground is the retention of the nuclides, particularly cesium and strontium, by ion exchange on the soil particles. Studies at Oak Ridge have shown that minerals with the mica structure are efficient in selectively sorbing cesium. For the low concentration of cesium generally found in wastes, ion-exchange capacity is seldom the controlling parameter for its removal. Biotite, a primary mica mineral, will sorb cesium even more efficiently than vermiculite even though biotite has about 1% the exchange capacity of vermiculite. In a demineralized water system, montmorillonite with 90-meq-per-100 g exchange capacity gave a distribution coefficient (Kd) of 1700 ml per g, whereas illite with 20-meq-per-100 g gave a Kd of 49,000. In the acidic pH range, ion exchange originating from isomorphous substitutions is the dominant mechanism for strontium removal by soils. In neutral and alkaline systems, additional reactions may occur. Hydrous oxides of iron and aluminum show high selectivities for strontium in this pH region. A Conasauga shale soil did not sorb as much strontium after being subjected to free iron oxide removal in spite of an increase in capacity resulting from the removal of the iron coating. An aluminum saturated montmorillonite showed the lowest removal of strontium at pH 6.0: when the pH was raised to 10, this sample removed more strontium than potassium, sodium, or calcium saturated samples. Knowledge of the mechanisms responsible for selective ion exchange reactions made possible the preparations of materials that show even greater selectivities than their natural counterparts. Thus, in solutions high in sodium nitrate concentrations, montmorillonite heated to 600°C and vermiculites saturated with potassium ions remove more cesium than the natural minerals. Gibbsite, an aluminum hydroxide mineral commonly found in soils, was made 50 times more selective for strontium by heating to temperatures that decomposed the mineral to the oxide. (auth)

905 SIGNIFICANCE OF CLIMATE IN RELATION TO THE DISPOSAL OF RADIOACTIVE WASTE AT SHALLOW DEPTH BELOW GROUND. R. M. Richardson (Geological Survey, Washington, D. C.). Colloq. Intern. Retention Migration Ions Radioactifs Sols, Saclay, 1962, 207-11(1963).

Climate is of supreme importance in evaluating the suitability of sites for the disposal of radioactive waste at shallow depth below ground. To illustrate the profound importance of climate in relation to ground disposal operations, a number of sites in humid eastern United States where radioactive wastes are disposed at shallow depth below ground are described and compared with each other, and with two sites in arid parts of western conterminous United States. Consideration of conditions prevailing in these sites indicates that from the standpoint of results consequent to disposal or release, either deliberate or accidental, of radioactive waste to shallow depth below ground, sites in arid regions are in general infinitely more satisfactory than those in humid regions. In humid regions where radioactive waste is disposed at shallow depth below ground, the sorptive action of certain minerals in the soil is the only mechanism that prevents the rapid reappearance of radioactive ions in nearby surface water courses. Where the local soils do not immobilize the hazardous ions, structures of some sort must be provided for containment. (P.C.H.)

Surface

See also the following references:

13, 19, 24, 32, 34, 35, 87, 92, 95, 97, 98, 122, 135, 164, 182, 186, 196, 236, 239, 241, 247, 325, 337, 517, 855, 983, 1009

906 (A/CONF.28/P/813) REVIEW OF THE STUDIES UNDERTAKEN IN THE UAR FOR EVENTUAL DISPOSAL OF LOW-ACTIVITY WASTES OF STRONTIUM, RUTHENIUM, CESIUM AND CERIUM INTO THE GROUND. M. A. El-Guebeily, M. Y. Farah, Aida E. Sinbel, M. S. El-Alfy, S. Shakour, H. Fouad, and G. El-Shobaky (United Arab Republic. Atomic Energy Establishment, Inshas, Egypt). May 1964. 12p.

Studies were made on the uptake of radioactive Sr, Cs, Ce, and Ru by the following Egyptian minerals: Mokattam and Sinai clays, Inshas sand, hematite, phosphorite, and zircons. The effects of particle size and pH on the uptake were studied. The efficiency of the minerals for decontamination was studied. (D.L.C.)

907 (A/CONF.28/P/814) UPON THE PHYSICO-CHEMICAL LAWS GOVERNING THE DISPOSAL OF SOME HEAVY RADIOELEMENTS INTO THE GROUND. M. A. Elguebeily and N. Z. Misak (United Arab Republic. Atomic Energy Establishment, Inshas, Egypt). May 1964. 12p.

Studies were made of the adsorption of U and Th on sand and vermiculite. $UO_2(NO_3)_2$ in HNO_3, uranyl carbonate complex in Na_2CO_3 solution, and $Th(NO_3)_4$ in HNO_3 were used. The effects of initial pH, grain size, temperature, and concentration on uptake were studied. Correlations and interpretations of the data were made, and it is concluded that the uptake proceeds by an ion exchange mechanism and is particle diffusion controlled. The data for sand can be described by a Langmuir isotherm, while those for vermiculite can be correlated by a Freundlich isotherm. (D.L.C.)

908 (BNWL-47) GEOPHYSICAL SEISMIC EVALUATION STUDY AT HANFORD. Brown, R. E.; Raymond, J. R. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). Dec. 1964. Contract AT(45-1)-1830. 49p. Dep. mn. CFSTI $3.00 cy, $0.50 mn.

A geophysical research program was conducted at Hanford to determine the feasibility and desirability of using seismic methods in geohydrologic studies. Detection and delineation were desired of seven different geological features that in some sites affect the movement of liquid radioactive wastes discharged to the ground. Rotary drilling methods exceeded expectations. They permitted in-well logging methods and produced information not obtainable from cable-tool methods. Results obtained reinforced information received from wells drilled by cable-tool methods and validated concepts from cable-tool well data. By extending the information laterally, the seismic program provided line data where only point data had been available. The reliability and significance of the information and concepts derived from it thus were greatly increased. The importance of seismic methods as an adjunct to other exploration methods used at Hanford was proven. Refraction methods proved the most usable and least expensive seismic techniques for Hanford studies and also minimized drilling needs. Rotary method drilling and in-hole logging provided adequate control on the geology at tie points. The use of fertilizer-grade ammonium nitrate primed with diesel oil was fully adequate and cheaper than other explo-

sives. Surface detonation was completely satisfactory and minimized the drilling of shot holes.

909 (BNWL-63) A SEQUENCE FOR PREDICTING WASTE TRANSPORT BY GROUND WATER. Nelson, R. W. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). Apr. 15, 1965. Contract AT(45-1)-1830. 12p. Dep. mn; CFSTI, $1.00 cy, $0.50 mn.

Tabular field data and fitting functions submitted to a digital computer give the ground water potential equations for heterogenous soils. Added to special boundary conditions, this potential equation is the key for computing the permeability distribution. The second computer program has four sequential operations. The first two provide permeability distribution and the last two give water travel time and flux distribution which enables water waste transport calculations. The nonlinear boundary value problem interrelates all pertinent variables, and when solved, gives the new ground water potential for flow system changes. The nonlinear equations are then solved digitally by a piecewise method for computations of predicted distribution of ground water potentials. The present computer program has a limit on problem size (8000 equations), but an enlarged program is being developed. Travel paths and times of flow and subsequently, the water arrival distribution are determined from the water potential function. An arrival curve is then used to plot the concentration of wastes entering streams. (auth)

910 (BNWL-243) SOIL SURVEY—HANFORD PROJECT IN BENTON COUNTY, WASHINGTON. Hajek, B. F. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). Apr. 1966. Contract AT(45-1)-1830. 19p. Dep. mn. CFSTI $1.00 cy, $0.50 mn.

A soil map and a descriptive report of Hanford Area soils grouped according to morphologic and genetic characteristics are presented.

911 (BNWL-SA-406) A SEQUENCE FOR PREDICTING WASTE TRANSPORT BY GROUND WATER. Nelson, R. William (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). Apr. 15, 1965. Contract AT (45-1)-1830. 17p. (CONF-651207-1). ORAU.

From American Water Resources Assn., 1st Annual Meeting, Chicago.

An analysis sequence is presented to enable rational predictions of containment concentrations entering streams, rivers, or other potable waters traversing the region of study by ground water transport. The purposes are to give an over-all description of the sequence, briefly describe the methods developed, and indicate the computer programs available to assist in accomplishing the analysis. Throughout the analysis, major emphasis was on realistically evaluating actual situations. The natural sediments are assumed heterogeneous with respect to the permeability, potential measurements can be irregularly located, and boundary conditions may be both mixed and nonuniform.

912 FURTHER STUDIES ON THE UPTAKE OF LONG-LIVED FISSION PRODUCTS ON SOME CLAYS FROM THE UNITED ARAB REPUBLIC. Mitry, E.; Gawad, A.; Emara, S.; Farah, M. Y. (Atomic Energy Establishment, Cairo). Vienna, International Atomic Energy Agency, 1965, Preprint No. SM-71/63(C), 10p. (CONF-651202-32). ORAU.

From Symposium on Practices in the Treatment of Low and Intermediate Level Radioactive Waste, Vienna.

Further studies in the UAR program on the feasibility of ground disposal of radioactive wastes are reported. New clays from the Nile and Méadi have been studied as well as those of Mokattam and Ainai. Their eventual domestication was demonstrated for long-lived isotopes as exemplified by ^{90}Sr. The effects of competing ions and heat treatment of clays was particularly emphasized, and an interpretation is proposed for the reported behavior. Further studies are being carried out for cesium on some deep mineralizations at Inshas and Borg el Arab reactor sites.

913 (EUR-2481.f) MIGRATION DE RADIO-ELEMENTS DANS LE SOL. Rapport Final. (Migration of Radio Elements in Soil. Final Report). Baetsle, L.; Maes, W. F.; Souffriau, J.; Staner, P. I. (Centre d'Etude de l'Energie Nucleaire, Mol (Belgium)). Mar. 1966. 66p. Dep. mn.

Meteorologic and hydrologic observations at the Mol site were studied and led to a correlation between both phenomena. From these data the recharge rate of the aquifer and the real infiltration velocities were computed. A new device called Multicounter was developed for the in situ measurement of ground water velocity and direction in non-consolidated media. The apparatus is made of a stainless steel cylinder with four counter tubes piercing the bottom and one injection needle at the center of the bottom plate. The activities of the four counters are recorded simultaneously and provide Activity-Time plots which characterize the movement. Fundamental studies on the nature of the dispersion phenomena showed the existence of two distinct mechanisms being mainly influenced by the molecular diffusion. The dispersion coefficients vary between 10^{-3} and 10^{-5} cm^2/sec for velocities ranging from 10^{-2} to 10^{-4} cm/sec. Two techniques were developed in order to increase the soil's retention capacity for ^{90}Sr. The first one is based on the introduction of lignite treated with $K_4Fe(CN)_6 + FeCl_3$, and the other consists of a treatment with HF followed by $K_2H_2Sb_2O_7$. Both "chemical barriers" catch all the ^{90}Sr found in 200 and 60 volumes, respectively flowing through the barrier.

914 (HW-72335(Pt.2)) STEADY DARCIAN TRANSPORT OF FLUIDS IN HETEROGENEOUS PARTIALLY SATURATED POROUS MEDIA. PART 2. THE COMPUTER PROGRAM. A. E. Reisenauer, R. W. Nelson, and C. N. Knudsen (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Oct. 1963. Contract AT (45-1)-1350. 84p.

The computer program, "Steady Darcian Flow in Soils," provides a means of obtaining numerical solutions to problems in the general class involving steady flow through porous media. Based on the mathematical and numerical treatment presented previously, the program was designed to handle the widest possible variety of boundary conditions. Potential distributions in saturated and partially-saturated, homogeneous or heterogeneous soils may be obtained. Combined saturated and partially-saturated flow cases may also be solved, since the equations used for one part are reducible to solve the other. Such a solution proceeds smoothly with little concern for the water table position which can be located easily in the problem solution. Moisture contents may be obtained from the results. The program was written for the IBM 7090 in FORTRAN and FAP languages. It can solve one-, two-, and three-dimensional and axisymmetrical problems with up to 8000 grid points. As many as 15 different soils may be included in a heterogeneous, partially-saturated flow problem. Optimum underrelaxation and overrelaxation techniques are used to increase the speed of convergence and to maintain stability. Other methods of maintaining stability are also discussed. (auth)

915 (HW-80086) GROUND WATER ANALOG TEST MODEL. M. D. Erickson and R. A. Walker (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Jan. 2, 1964. Contract AT(45-1)-1350. 14p.

A test model of an electric analog simulation of the ground water beneath the Hanford Project was constructed to determine the safety of radioactive waste ground disposal operations. The 662-node model operated satisfactorily, and no major problems were encountered in the construction. The analog data were compared with predicted results with a maximum error of 10%; an inter-grid reaction was found to be causing errors, and a uniform grid size should be used throughout the network. (D.L.C.)

916 (HW-81746(p.4.32-6)) INVESTIGATION OF THE DISPOSITION AND MIGRATION OF GROSS GAMMA EMITTERS BENEATH LIQUID WASTE DISPOSAL SITES. J. R. Raymond (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.).

Wells immediately adjacent to liquid waste disposal sites were logged with a gamma scintillation probe to determine the penetration depth of radionuclides. Information is presented to show contamination magnitude and migration changes at several "typical" sites. Well scintillation logs are shown to illustrate changes in waste penetration at three waste disposal facilities that have different operational histories. (auth)

917 (HW-81746(p.4.39-42)) MEASUREMENT OF SOIL PARAMETERS FOR PARTIALLY SATURATED FLOW. L. G. King (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.).

A program is in progress to measure simultaneously the relationships between the three soil parameters: permeability, capillary pressure, and moisture content. Moisture content measurements utilize the characteristics of soils having different degrees of wetness to attenuate the low-energy gamma rays from an ^{241}Am source. Capillary pressure and permeability are measured simultaneously using conventional techniques. These data, collected for typical project soils, were used as inputs to the "steady-state" computer program, which solves for the equilibrium waste-flow patterns beneath ground disposal facilities. (auth)

918 (HW-84549) CHEMICAL EFFLUENTS TECHNOLOGY WASTE DISPOSAL INVESTIGATIONS, JANUARY–DECEMBER 1964. D. J. Brown (General Electric Co., Richland, Wash. Hanford Atomic Products Operation). 1964. Contract AT(45-1)-1350. 23p. Dep.(mn); $1.00(cy), 1(mn) CFSTI.

Results of radioactive nuclide surveillance in the ground at Hanford are reported. Sections are included on analysis of ground water monitoring data from 200-east, 200-west, and 600 areas. Information is also included on fission-product tritium in ground water and proposed well drilling. Data are included on the average concentration of radioisotopes in various wells, and on concentration of radioisotopes found in ground water that exceed the routine detection limit of 0.08 pCβ cc. (J.R.D.)

919 (HW-84577) SCINTILLATION PROBE RESULTS ON 200 AREA WASTE DISPOSAL SITE MONITORING WELLS. J. R. Raymond and V. L. McGhan (General Electric Co., Richland, Wash. Hanford Atomic Products Operation). Dec. 17, 1964. Contract AT(45-1)-1350. 62p. Dep.(mn); $3.00(cy), 2(mn) CFSTI.

Scintillation well probes were used to monitor the subsoil contamination below radioactive liquid waste ground disposal facilities. Results are discussed for the 200-East and 200-West facilities. The results indicate that significant lateral spread of wastes occurs in the sediments beneath the 200 Areas, and that the downward migration rates of gross gamma emitters is relatively slow. (D.L.C.)

920 (ORNL-3697(p.43-8)) MOVEMENT OF RADIONUCLIDES IN TERRESTRIAL ENVIRONMENT. T. F. Lomenick, Tsuneo Tamura, Ferrucio Gera, W. M. McMaster, R. M. Richardson, and D. A. Gardiner (Oak Ridge National Lab., Tenn.).

Core samples taken in and around waste pits show that most of the ^{137}Cs and ^{90}Sr is concentrated in sludge and precipitates in the pits and on the first few inches of shale comprising the side walls and bottoms of the pits. By the use of tritiated water as a tracer, it was observed that the normal rate of ground-water movement in the waste pit area is about 0.5 ft/day. Most of ^{137}Cs in the bed of White Oak Lake occupies highly selective exchange sites on the illitic clay fraction of the lacustrine sediment and can be desorbed only by disruption of the clay lattice structure. Analyses of selected samples obtained from the upper 0.2 ft of White Oak Creek bottom sediment indicate that less than 5×10^5 $\mu\mu$c of ^{137}Cs and substantially smaller quantities of ^{60}Co and ^{106}Ru are stored in this zone. The inference is drawn that the residence time for the less-than-63-μ sediment in this zone is relatively short and that the material is scoured during higher-than-normal stages and transported into White Oak Lake. As an extension of information gained from laboratory studies on radionuclide sorption, field studies on deliberately contaminated plots, using ^{137}Cs, were started in order to correlate areal and vertical redistribution of the nuclide with natural factors, including biologic, meteorologic, and pedologic factors. (auth)

921 (ORNL-3697(p.65-84)) RADIOACTIVE WASTE AREA AND RADIATION EFFECTS STUDIES. S. I. Auerbach, G. N. Brown, D. A. Crossley, Jr., et al. (Oak Ridge National Lab., Tenn.).

Statistical analyses of sedges growing on White Oak Lake bed indicated that length of inflorescence was mostly correlated with the air dose rate of gamma radiation at 1 m above ground surface. Some of the natural vegetation around the unshielded Health Physics Research Reactor exhibited the first visible effects of radiation damage. Radiation effects studies completed this year included fast-neutron exposures from the HPRR, and gamma-radiation exposures from both a ^{137}Cs field source and the White Oak Lake bed. Among the findings were that all acute fast-neutron doses ranging from 100 to 500 rads resulted in 100% mortality of pine seedlings. Pine seedlings subjected to chronic gamma irradiation from White Oak Lake bed for 178 days exhibited 100% needle mortality. A study was initiated to determine the seasonal changes in nuclear volumes of local tree species. Shoot apexes were measured monthly from 14 local species. An analysis of variance showed that most species increased in size before shoot elongation. Initial spring activity of nucleic acids in shoot apexes is thus earlier in eastern Tennessee than would be anticipated from any external evidence. Continuing work on the use of metaphosphate glass dosimeters demonstrated that low-Z glass rods are subject to fading in sunlight. Mud-dauber wasps that used radioactive mud from the liquid waste pits were found to be affected by the ionizing radiation. Continuing studies on radiocesium accumulation and elimination by various species of insects demonstrated the usefulness of these techniques for measuring food consumption and insect production under field conditions. Energy flow through the herbivorous insect population was estimated by using food consumption rates calculated with steady-state equilibria for radiocesium. Preliminary studies of radiation response of several insect species including crickets and grain beetles were reported. Hematological studies of native mammals were completed. The objective of these studies was to determine the effects of endogenous and environmental factors on the blood of indigenous species. These studies have shown that factors such as age, sex, certain diseases, certain parasites, injuries, confinement, and season govern hematopoietic changes. The largest and most consistent change in normal serum patterns is characteristically demonstrated by albumin. A coefficient of correlation of -0.360 between total white cell count and

albumin–globulin ratio was calculated for the white-footed mouse ($0.05 > P > 0.02$).

922 THE OCCURRENCE AND RETENTION OF RADIONUCLIDES IN THE SEDIMENTS OF WHITE OAK LAKE. T. F. Lomenick and D. A. Gardiner (Oak Ridge National Lab., Tenn.). Health Phys., 11: 567-77(June 1965). (ORNL-P-331)

As a result of the discharges of large volumes of low-level radioactive liquid wastes to surface streams at the Oak Ridge National Laboratory, large quantities of radionuclides have accumulated in the bottom sediments of White Oak Lake. Ruthenium-106 (1038 C) and cesium-137 (704 C) account for more than 90% of the total activity now present at the site, while ^{60}Co(152 C), the rare earths (17 C, exclusive of ^{90}Y), and ^{90}Sr(15 C) make up the remainder. More than half of the activity is associated with the upper 6-in. sediment layer, while progressively smaller quantities of activity are found with depth. The ruthenium, which is restricted to a small area in the now dry upper lake bed, is partially water soluble; however, its rate of movement through the soil is slow enough so that radioactive decay reduces the concentration of that reaching surface streams to insignificant levels. Most of the ^{137}Cs occupies highly selective exchange sites on the illitic fraction of the clay in the sediment and can be desorbed only by disruption of the lattice structure. Only a small fraction of the ^{60}Co in the soil was found to be exchangeable. It is, therefore, unlikely that any large fraction of the ^{137}Cs or ^{60}Co would move from the area except through erosion of the sediment. About one-half of the ^{90}Sr and the rare earths in the sediment appears to be exchangeable, while the other half is in the form of slightly soluble salts. Through leaching by ground water, a slow depletion of strontium from the dry part of the lake bed occurs. The accumulation of radionuclides in the sediments of White Oak Lake illustrates the effectiveness of relatively quiescent bodies of water in concentrating activity in stream beds and in retarding the downstream movement of these materials.

923 NATURALLY OCCURRING FIXATION OF ^{137}Cs ON SEDIMENTS OF LACUSTRINE ORIGIN. Lomenick, T. F.; Tamura, T. (Oak Ridge National Lab., Tenn.). Soil Sci. Soc. Am. Proc., 29: 383-7(July–Aug. 1965). (ORNL-P-737).

White Oak Lake formerly served as a final settling basin for contaminated waste water discharged from the Oak Ridge National Laboratory. The site, which now serves as an ecological study preserve, covers an area of 44 acres and contains approximately 1,000,000 ft^3 of contaminated sediment. The most abundant radionuclide present in the sediment of lacustrine origin is ^{137}Cs, 704 ± 35 Curies. Concentrations of ^{137}Cs on the sediment were observed to be as high as 77×10^{-3} μC/g. More than 80% of the ^{137}Cs is associated with the clay fraction of the material, which was found to be predominantly illite. Significant quantities of ^{137}Cs were removed from the material only by treatment with strong acids. Studies of several selected clays showed illite to have a high affinity for ^{137}Cs at levels of the same order of magnitude as those observed in lake bed sediment and that desorption is only accomplished after disruption of the lattice structure.

924 (ORNL-TM-1108) GEOLOGY AND SOILS OF WHITEOAK CREEK BASIN, TENNESSEE. W. M. McMaster and H. D. Waller (Oak Ridge National Lab., Tenn.). May 1965. Contract W-7405-eng-26. 36p. Dep.(mn); $2.00 (cy), 1(mn) CFSTI.

Since Oak Ridge National Laboratory began operation in 1943, radioactive wastes have been released on a continuous basis to the environment of Whiteoak Creek Basin. The natural factors affecting movement or retention of the radionuclides released are the water moving in and through the basin and the materials over and through which the water and its associated radionuclide load pass in transit. Because the geology and soils of Whiteoak Creek Basin are the principal modifying agents in water movement and the soil selectively fixes radionuclides, a knowledge of their characteristics is necessary for an understanding of the manner in which radionuclides move through or are stored in the basin. The geologic setting, the characteristic soils, and the general hydrologic characteristics of the geology and soils of the basin are described.

925 (RL-SA-15) RADIATION CONTROL OF ACCIDENTALLY CONTAMINATED SEEPAGE PONDS. Backman, G. E.; Roddy, L. W. (General Electric Co., Richland, Wash. Hanford Atomic Products Operation). June 3, 1965. Contract AT(45-1)-1350. 18p. (CONF-650616-2). Dep.(mn); $1.00(cy), 1(mn) CFSTI.

From Health Physics Society, Annual Meeting, Los Angeles.

In June 1964, a leak developed in the cooling coil of a process waste inventory tank in the Purex Plant, resulting in contamination of two ponds. An account is given of the immediate and long-term radiation control measures taken; they included reduction of high radiation levels, radioactivity determinations in water fowl, elimination of algae and invertebrate life, and soil sterilization. It is concluded that the problem is well under control, as most of the radioactivity is adsorbed on the bottom sediments.

926 (STI/PUB/103) RADIOACTIVE WASTE DISPOSAL INTO THE GROUND. Safety Series No. 15. (International Atomic Energy Agency, Vienna (Austria)). 1965. 116p. IAEA: $2.50; S52,50; 15/-stg; F.Fr 10,-; DM 8,-.

Information is given on: site characteristics affecting ground disposal; chemical reactions and physical behavior of wastes in the ground; modes of release; evaluation of sites and methods for ground disposal; and standards and control techniques. 138 references are cited.

927 (TID-19618(p.5-37)) MINERAL CYCLING ON A SITE CONTAMINATED WITH RADIOACTIVE WASTE PRODUCTS—WHITE OAK LAKE BED, OAK RIDGE, TENNESSEE. Final Report. H. R. DeSelm (Tennessee. Univ., Knoxville).

Data relevant to the dispersal of radioactive waste were collected as the result of studies of the bed of the former White Oak Lake, Oak Ridge, Tennessee. White Oak Lake served as a depository for low-level radioactive wastes from 1943 to 1955, when it was drained. Data are presented on biotic invasion, the chemical content of natural vegetation, site factors influencing plant chemical content, and the cycling of certain elements in the lake bed ecosystem. Emphasis was placed on the movement of K, Ca, Mg, Cs^{137}, Cs^{60}, Sr^{90}, Ru^{106}, Zr^{95}, Nb^{95}, and Ce^{144}. The cycling of materials through the ecosystems dominated by herbaceous taxa is discussed from the standpoint of amount of material moved per year, proportion of available material moved per year, and number of years required to return to the soil one year's input. Using the vegetation-sampling data, chemical concentrations, and litter accumulation data, a predictive equation was derived regarding rate of accumulation and movement of certain elements through soil-vegetation-litter systems. Calculated equilibrium figures approximate those of mesic and moist forests of the area.

928 (UCRL-13010) INTERIM REPORT ON RESEARCH ON RADIOACTIVE CONTAMINATION OF GROUND-WATER AQUIFERS. First Annual Report. (Texas. Agricultural and Mechanical Coll., College Sta-

tion. Research Foundation; Texas. Agricultural and Mechanical Coll., College Station. Engineering Experiment Station and California. Univ., Livermore. Lawrence Radiation Lab.). June 15, 1961. Contract W-7405-Eng-48. 51p.

Preliminary results are reported for studies of the rate of migration of tracer amounts of Sr^{90} through water-saturated quartz sand in vertical exchange columns and ion exchange measurements on Ottawa sand. A potentiometric model was developed for studying liquid flow in sand-packed columns that was based on the similarity of equations for liquid and electrical flow. Results are reported from applications of the model on two types of columns. (C.H.)

929 (UJV-1107/64) PEREMESHCHENIE RADIOIZOTOPOV V POCHVAKH. (Migration of Radioisotopes in Soil). Z. Dlougi and I. Zaiits (Československá Akademie Věd. Ústav Jaderného Výzkumu, Řež). [1963]. 13p. Dep.(mn).

Liquid radioactive waste materials can be disposed directly in the soil only under special geological conditions, such as those prevailing at Hanford, Wash. Under the conditions prevailing in Czechoslovakia where the water table in most places is close to the surface, direct disposal is considered practically impossible. It was found however, that the lithological properties of certain quaternary deposits are favorable toward sorption; therefore, the interaction of these soils with certain long-lived isotopes was investigated. Because of their importance as contaminants of the drinking water as the result of their long half life and their similarity with the biogenic elements Ca and K, ^{90}Sr and ^{137}Cs were chosen for the study. Soil samples from Central and Northwestern Bohemia, including loess, clay, and sandy specimens were used for the tests involving the determination of their sorption capacity by chromatography. It was found that the migration of the ^{137}Cs in all the media examined is negligible. The migration of ^{90}Sr was found to depend to a great extent on the Ca content of the natural waters (or of the water used for the tests). The penetrability of the formation also was found to play an important role in this respect. It was concluded that most of the Czech soil samples have a good sorptive capacity and may be considered for the safe disposal of liquid radioactive wastes. (TTT)

930 ALGAL INFLUENCE ON RADIONUCLIDES IN SETTLING PONDS. Echo, John; Hawkins, D. B. (Idaho Operations Office, Idaho Falls). Nature, 209: 1105-7 (Mar. 12, 1966).

The influence of algae growing on settling ponds, through which low-level aqueous radioactive waste solutions are allowed to seep into the ground, on the chemical state of radionuclides in solution, and on exchange reactions with the basin alluvium was investigated. Results of tracer studies using ^{137}Cs and ^{85}Sr and a synthetic alluvium showed that exchange reactions of the radionuclides on the soil are not affected by cycling through algal cells; that algae concentrate radioactivity from very low levels in water to rather high levels in algal aggregation; and that dried algae can become windborne and be transported a considerable distance from the pond. It is concluded that an algicide should be used to control the growth of algae in radioactive-liquid disposal ponds.

931 THE EFFECT OF SALT CONCENTRATION IN SOIL SOLUTIONS ON THE RATE OF DIFFUSION OF MICROQUANTITIES OF STRONTIUM IN THE SOIL. Prokhorov, V. M.; Frid, A. S. Radiokhimiya, 7: 496-8 (1965). (In Russian).

The effect of soil composition on the rate of diffusion of radioactive isotopes in the soil was investigated. Experiments were conducted with ^{90}Sr without a carrier. The soil (37 m^2/g, specific surface) was wet with concentrations of $CaCl_2$ solutions ranging from 3.3×10^{-2} to 8.5 g-equiv/l. The soil was moistened to the extent of 30% by weight. To correct for adsorption of Ca^{2+} the equilibrium concentration of $CaCl_2$ was determined by complexometric titration. The diffusion coefficient of ^{90}Sr as a function of the concentration of Ca^{2+} was measured. In the concentration range 4×10^{-3} to 5.8×10^{-1} N the diffusion coefficient of Sr increases by approximately a factor of 13. From 0.6 to 6.6 N the diffusion remains practically constant.

932 STRONTIUM FIXATION BY LIME CONTAINED IN SOILS. R. Mokady and M. Gal (National and University Inst. of Agriculture, Rehovoth, Israel). Science, 145: 154-5(July 10, 1964).

The suitability of soils for the disposal of radioactive strontium waste products is determined in part by the amount and properties of calcium carbonate in the soil. Analyses of the strontium ion concentrations in the effluent solution of soil columns and x-ray diffraction patterns suggest that a fraction of the strontium ions is precipitated as strontium carbonate. (auth)

933 NATURALLY OCCURRING FIXATION OF ^{137}Cs ON SEDIMENTS OF LACUSTRINE ORIGIN. Lomenick, T. F.; Tamura, T. (Oak Ridge National Lab., Tenn.). Soil Sci. Soc. Amer., Proc., 29: 383-7(July-Aug. 1965).

The distribution of ^{137}Cs in sediments was studied in White Oak Lake, which formerly served as a final settling basin for contaminated waste water discharged from the Oak Ridge National Laboratory. Core samples of sediments were taken to a depth of 24 in., and assayed for ^{106}Ru, ^{60}Co, and ^{137}Cs. The most abundant radionuclide present in the sediment of Lacustrine origin is ^{137}Cs, with 704 C in the entire lake bed. Concentrations of ^{137}Cs on the sediment were observed to be as high as 77×10^{-3} $\mu C/g$. More than 80% of the ^{137}Cs is associated with the clay fraction of the material, which was found to be predominantly illite. Significant quantities of ^{137}Cs were removed from the material only by treatment with strong acids. Studies of several selected clays showed illite to have a high affinity for ^{137}Cs at levels of the same order of magnitude as those observed in lake bed sediment and that desorption is only accomplished after disruption of the lattice structure. The desorption behavior of the sediments and the corroborative behavior of the reference minerals are evidence that Cs in the sediment will not be leached by percolating ground water to any appreciable degree. However, movement of Cs can occur by erosion of the sediment.

934 SORPTION OF CESIUM BY SOIL LAYER MINERALS. Titlyanova, A. A. (Ural Branch, USSR Academy of Sciences, Sverdlovsk). Soviet Soil Sci. (Engl. Transl.), No. 12, 1313-18(Dec. 1964).

The adsorption and desorption of ^{137}Cs was studied in various types of soils and pulverized minerals. Samples of each were shaken with ^{137}Cs solutions at concentrations of 10^{-9} to 10^{-2} eq/l and the radioactivity of the sediment was measured. Desorption of Cs was measured in water to which various cations were added. The specific adsorption of microquantities of cesium was found to occur in the mineral part of the soil. The percentage of cesium desorption from the humic acids was high and constant throughout the entire range of change in cesium concentration. The minerals biotite and phlogopite were characterized by a stable, low percentage of cesium desorption with all the cations tested in concentrations ranging from 4×10^{-11} to 4×10^{-8} eq Cs/g of mineral and by a sharp rise in the percentage of desorption in the region corresponding to increased concentrations of cesium. The desorption of cesium from muscovite by all three cations was effected much more easily. The

desorption of cesium from vermiculite was very low in concentrations ranging from 4×10^{-11} to 4×10^{-7} eq/g of mineral. Desorption rose sharply with a Cs concentration of 10^{-7} to 10^{-5} eq/g of mineral. Desorption again decreased sharply with a cesium content of about 4×10^{-4} eq/g. The desorption curve for chlorite was linear over cesium concentrations ranging from 10^{-10} to 10^{-5} eq/g of mineral. Desorption rose sharply in concentrations close to saturation. Askanite and gumbrin adsorbed cesium more firmly than did bentonite. A feature of the desorption curves for most of the minerals investigated was the presence of a lower desorption plateau in cesium concentrations ranging from 10^{-11} to 10^{-8} or 10^{-7} eq/g of mineral. No clear-cut relation was noted between the amount of stably adsorbed cesium and the size of the surface of the mineral. It was concluded that the specific adsorption of microquantities of cesium is effected by secondary layer clay and primary micaceous minerals that are present in all soils. Therefore, the behavior of microquantities of cesium in each soil is determined by its mineralogic composition.

935 RADIOTRACER MEASUREMENTS OF THE HYDRAULIC BEHAVIOR OF A LARGE SHALLOW POND. B. A. Fries and I. M. Whittemore (California Research Corp., Richmond). Trans. Am. Nucl. Soc., 8: 78-9(May 1965).

936 RETENTION AND MIGRATION OF RADIOACTIVE ISOTOPES IN THE LITHOSPHERE AT THE NATIONAL REACTOR TESTING STATION-IDAHO. Bruce L. Schmalz and W. Scott Keys (Atomic Energy Commission, Idaho Falls, Idaho). Colloq. Intern. Retention Migration Ions Radioactifs Sols, Saclay, 1962, 243-56(1963).

Aqueous liquids containing mixed fission and activation products in a concentration of the order of 10^3 μC/ml, plus tritium in a concentration of 10^3 μC/ml, are routinely discharged to ponds excavated in the alluvial regolith. Migration of radioisotopes through the regolith and basalt is delineated by samples from drill holes. The water and isotopes have been detected in the regolith at distances of 150 meters from the MTR-ETR disposal pond. The waste has been detected in perched water in the basalt at distance up to 1.5 kilometers. Migrating isotopes that have been identified are: tritium, chromium-51, cobalt-60, iodine-131, cerium-144, ruthenium-106, silver-110. Tritium, chromium-51, and cobalt-60 have been observed to be particularly migratory. The migration of cobalt-60 and cesium-137 has also been observed at other plant areas. The concentration of tritium diminishes with distance from the disposal pond. Because disposal is to the zone of aeration this cannot be explained on the basis of dilution. Ion exchange has been substantiated by laboratory studies. Columns filled with natural earth materials have shown that significant attenuation of tritium does occur despite relatively small column capacities. Migration of contaminated ground water resulting from disposal of waste at the NRTS is most extensively mapped on the basis of tritium analyses. Concentrations of 10^6 tritium units (T.U.) have been discovered at distances of 210 meters from a disposal well. Attenuation of the concentration to 1,200 T.U. at distances of 5 kilometers from the disposal point is observed. Attempts to trace contamination in ground water beyond 400 meters on the basis of isotopes other than tritium have been unsuccessful. Chemically contaminated wastes have also been traced to a distance of 5 kilometers. (auth)

RECOVERY OF CONSTITUENTS

See also the following references:

1, 2, 3, 13, 14, 26, 42, 43, 65, 69, 72, 76, 78, 90, 96, 207, 218, 219, 228, 243, 244, 246, 247, 248, 253, 590, 599, 619, 649, 650, 720, 731, 770, 772, 780, 788, 803, 805, 806, 992

937 (A/CONF.28/P/252) LARGE-SCALE FISSION-PRODUCT RECOVERY. R. E. Tomlinson (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.), B. F. Judson, and L. L. Zahn. 15p.

Prepared for the United Nations Third International Conference on the Peaceful Uses of Atomic Energy, 1964.

Methods used for separating and transporting large quantities of fission products (Ba–La, Sr, Ce, Pm, Zr, Cs, Tc, Kr, and Xe) are outlined. Future production plans are discussed. (T.F.H.)

938 (A/CONF.28/P/609) PROCESSES FOR THE RECOVERY AND PURIFICATION OF FISSION PRODUCTS FROM IRRADIATED URANIUM. Sten Ahrland and Karl Erik Holmberg (Aktiebolaget Atomenergi, Studsvik, Sweden) May 1964. 14p.

Among the various methods for processing waste solutions, ion exchange using inorganic compounds, and freezing out, are studied. Zirconium phosphate and the hydrous oxides of zirconium and silicium (silica gel) have been judged as the inorganic ion exchangers of greatest promise. Zirconium phosphate is moderately acidic, the hydrous oxides only weakly. In fact the zirconium hydrous oxide is amphoteric, acting mainly as an anion exchanger when pH $<$ 5. The selectivity between ions of different charges generally increases with decreasing acidity. On the other hand, a certain degree of sorption is reached at a lower pH value, the more acidic the gel which may be important for easily hydrolyzed ions. For pH $>$ 8, zirconium phosphate is heavily hydrolyzed and is thus not useful as an ion exchanger. Silica gel can be used up to a pH = 9, where dissolution becomes appreciable. The workable ranges of pH, as well as other properties of the investigated ion exchangers, thus supplement each other rather well. On partial freezing of a concentrated solution of uranyl nitrate containing a small amount of fission products, uranyl nitrate and ice crystallize leaving the fission products in the mother liquid. The phases may be separated by centrifugation, leaving the solid somewhat contaminated by mother liquid. By repeated partial freezing it should be possible to purify the uranyl nitrate further. The more important fission products have been tested and do not appear to cocrystallize with uranyl nitrate. The freezing out of uranyl nitrate and ice may also be performed in the presence of moderate concentrations of nitric acid, which reduces the solubility of the uranyl nitrate. It thus seems physically possible to recover a considerable fraction of uranium in a rather pure form from a concentrated uranyl nitrate solution containing small amounts of fission products by repeated fractional freezing. Simultaneously the high activity waste is concentrated in a reduced liquid volume.

939 (A/CONF.28/P/772) THE USE OF INORGANIC ION EXCHANGERS IN ACID MEDIUM FOR THE RECOVERY

OF Cs AND Sr FROM REPROCESSING SOLUTIONS. L. Baetsle, D. Van Deyck, D. Huys, and A. Guery (Centre d'Etude de l'Energia Nucleaire, Mol, Belgium). May 1964. 15p.

The search for new synthetic mineral ion exchangers exhibiting selective uptake of long-lived fission products has led to the elaboration of a separation scheme for cesium an strontium from reprocessing solutions. Zirconium phosphate, whose chemical, physical, and radiochemical properties are reported in connection with its structure, shows selective ion exchange properties for cesium in acid medium. The loading of cesium reaches a maximum of 0.2 me/g when a solution containing 0.5 g/l cesium and 0.5 N in acidity is fed to the zirconium phosphate in the H^+ form. That corresponds to a concentration factor of about 60. The results are presented for several types of reprocessing solutions, a.o., those reported by the Hanford, Windscale, and Marcoule plants, and those expected at the Eurochemic plant. A new family of synthetic ion exchangers was found by precipitating in acid medium Na_2MoO_4 or Na_2WO_4 solution with $H_4Fe(CN)_6$. The ferrocyanide molybdate ion exchanger has an extremely high affinity for cesium, which allows the separation from reprocessing waste solution of 2M HNO_3. A concentration factor of nearly 100 may be achieved leading to a maximum loading of 0.5 me/g at this high acidity. The difficulties encountered with the strontium separation in acid medium and in the presence of bulk ions are outlined; some promising results were obtained with clinoptilolite after dilution of the feed solution. An experimental set-up built for the separation of strontium and cesium with mineral ion exchangers is reported with emphasis on its technology. (auth)

940 (ANL-7020(p.100-49)) FUEL CYCLE APPLICATIONS OF VOLATILITY AND FLUIDIZATION TECHNIQUES. (Argonne National Lab., Ill.).

The development of the fluidized-bed fluoride volatility process for enriched fuels is described. Laboratory studies included the recovery of Pu from alumina beds, sorption of PuF_6, volatilization of Mo, and fluorination of wastes. Engineering-scale investigations were conducted on the processing of fuels of low- and high-enrichment fuels. Corrosion and fluidization studies are also described. (D.L.C.)

941 (BLG-267) THE USE OF INORGANIC ION EXCHANGERS IN ACID MEDIUM FOR THE RECOVERY OF Cs AND Sr FROM REPROCESSING SOLUTIONS. L. Baetsle, D. Van Deyck, D. Huys, and A. Guery (Centre d'Étude de l'Énergie Nucléaire, Brussels). Aug. 1964. 14p. Dep. (mn).

Among the many existing separation techniques, ion exchange on mineral ion exchangers seems to be the most promising one for the quantitative separation of long-lived fission products from acid reprocessing solutions. Zirconium phosphate (ZrP) is suitable for the separation of ^{137}Cs from reprocessing solutions desacidulated to 0.5 M HNO_3. The specific activity amounts to 0.5 to 1 Ci/g ZrP when the feed solution contains 20 Ci/l. Ferrocyanide molybdate (FeMo), a new mineral ion exchanger, allows the selective ^{137}Cs separation from reprocessing solutions after their desacidulation to 2 M HNO_3 by formaldehyde. The loading capacity for ^{137}Cs amounts to 2 to 4 Ci/g FeMo. A new Sr-selective ion exchanger, called polyantimonic acid, was developed and showed to be indicated for the ^{90}Sr recovery from acid reprocessing effluents. (auth)

942 (BNWL-4) RECOVERY OF STRONTIUM FROM STORED PUREX WASTE BY LEAD SULFATE CARRIER PRECIPITATION—PILOT PLANT STUDIES. G. L. Richardson (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). Jan. 15, 1965. Contract AT(45-1)-1830. 32p. Dep.(mn); $2.00(cy), 1(mn) CFSTI.

Pilot plant runs that were designed to test and optimize a flowsheet for lead sulfate scavenging process for isolating strontium crud fraction from wastes on a 100 to 700-liter scale and to evaluate alternate methods of separating the precipitate from the supernatant liquid are described. The tests were restricted to Purex-type wastes, but the results should be approximately the same for Redox wastes. Strontium losses of 5% or less were demonstrated. High iron concentration (0.35M) in one series of runs was believed to have contributed to high strontium losses (ranging from 6 to 28%) through sulfate depletion by complexing and ferric sulfate precipitation. The precipitate, after digestion, proved to be quite insoluble and could be washed with cold water or very dilute nitric acid (pH > 1) with no significant solubility losses for either strontium or cerium. The precipitate was satisfactorily metathesized to the acid-soluble carbonate by using a mixture of sodium hydroxide and carbonate as the metathesis solution. Best results were obtained when the hydroxide/lead mole ratio was 15 or above. As little as 0.2M carbonate was sufficient to prevent strontium solubility losses. Overall product decontamination factors for some of the feed constituents were obtained. Under optimum washing conditions, the DF's ranged from 30 to 460 for iron, aluminum, calcium, magnesium, manganese, and nickel. Lead decontamination factors of 3 and 8 were obtained using hydroxide/lead mole ratios of 15 and 17, respectively, in the metathesis solutions. Centrifugation, continuous decantation, and batch settling and decantation proved to be feasible methods for removing the supernate from the precipitate. (P.C.H.)

943 (BNWL-SA-504) FISSION PRODUCT RECOVERY RESEARCH AT BATTELLE-NORTHWEST. Moore, R. L. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). Dec. 27, 1965. Contract AT(45-1)-1830. 9p. (CONF-660305-6). Dep. mn. CFSTI $1.00 cy, $0.50 mn.

From American Nuclear Society Topical Meeting, Augusta, Ga.

The major recent developments in fission product recovery at the Pacific Northwest Laboratory (formerly Hanford Laboratories) are described. Process optimization studied for the compact band ion exchange separation of individual light rare earth elements, using DTPA, EDTA, or HEDTA as eluting agents. Equipment for purifying kilogram quantities of promethium has been installed recently, and was applied to the preparation of 62,500 Curies of very high-purity promethium. A process for recovery of fission product rhodium has been developed, and used for the purification of one gram of rhodium. Earlier work at Hanford Laboratories is briefly summarized by specific fission product, with respect to scientific and technical contributions, as well as the large-scale manufacturing production program which has resulted from application of this new technology.

944 (CONF-660305-5) RECOVERY AND PURIFICATION OF ^{147}Pm. Lowe, J. T. (Du Pont de Nemours (E. I.) and Co., Aiken, S. C. Savannah River Lab.). Jan. 1966. Contract AT(07-2)-1. 5p. Dep. mn. CFSTI $1.00 cy, $0.50 mn.

From American Nuclear Society Topical Meeting, Augusta, Ga.

A process to reclaim the rare earth fission products as a potential source of ^{147}Pm is described. The process is designed to have no significant interference to present plant operation. The rare earth fission products are extracted from Savannah River Plant wastes with 50% tributyl phosphate (TBP) in dodecane. Most of the fission product cerium is oxidized to Ce(IV) through isotopic exchange with inactive Ce(IV), and is retained in the 50% TBP while the trivalent rare earths (including the ^{147}Pm) are back-ex-

tracted into the aqueous phase. Removal of ^{144}Ce eliminates the major source of heat in the rare earth fraction and permits easier storage of the ^{147}Pm fraction. After a suitable storage period to permit decay of the undesirable ^{148}Pm, the ^{147}Pm can be purified by ion exchange.

945 (EUR-2754.f) RECUPERATION PAR PRECIPITATION OXALIQUE DU ^{90}Sr ET DU ^{144}Ce A PARTIR DE SOLUTIONS CONCENTREES DE PRODUITS DE FISSION. (Recovery of ^{90}Sr and ^{144}Ce From Concentrated Fission-Product Solutions by Means of Oxalic-Acid Precipitation). Klose, A.; Brohee, H. (European Atomic Energy Community); Lefevre, J. (Commissariat a l'Energie Atomique, Saclay (France). Centre d'Etudes Nucleaires). Mar. 1966. 32p. Dep. mn.

With the aim of evaluating the savings that could be effected on the installation cost of a strontium recovery unit, consideration was given to the separation of strontium by means of oxalic acid precipitation in a plant envisaged for cesium recovery. A detailed study of the process showed that its selectivity and yield are too low to enable a saving to be achieved under the conditions applied, namely production of purified ^{90}Sr sources and decontamination of the fission product solutions.

946 (HW-71589) LABORATORY STUDIES OF THE PRECIPITATION OF CERIUM SULFATE FOR OFF-SITE SHIPMENT. H. H. Van Tuyl (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Nov. 1, 1961. Contract AT(45-1)-1350. 5p. Dep.(mn); $1.00(cy), 1(mn) OTS.

Several series of experiments were performed to determine the optimum conditions for precipitating cerium as the sulfate for loading into the cerium filter cask for off-site shipment. Two sets of conditions were determined to be optimum within the limitations imposed by plant equipment, one set for precipitation in the presence of tartrate to obtain additional purification of the cerium, and the other set in the absence of tartrate. Optimum conditions with tartrate are as follows: to one liter of feed, add 20 ml of 5 M tartaric acid and 500 ml of 2 M sodium sulfate; adjust to pH 0.5 to 1 with 50 percent sodium hydroxide; digest one hour at the highest temperature obtainable consistent with proper jet operation; centrifuge. In the absence of tartrate, the procedure is identical except that tartrate is omitted, and the pH range is 3 to 5. Loss with either procedure at 40°C is about 16 percent of the cerium. In either case, the precipitate may then be slurried with two liters of 0.5 M sodium sulfate, digested one hour as above, and filtered into the cask. Loss in the slurrying step is about ten percent of the original cerium when the precipitation and slurrying are both performed at 40°C. (auth)

947 (HW-74036) REPORT OF INVENTION—IMPROVEMENTS OF STRONTIUM SULFATE PRECIPITATION PROCESS. H. H. Van Tuyl (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). June 18, 1962. Contract [AT(45-1)-1350]. 7p. Dep.(mn); $1.00(cy), 1(mn) OTS.

A modification of the carrier precipitation process for the recovery of Sr from Purex wastes, is described in which the separation of Sr from Ce and other rare earths is improved. Besides the addition of sodium sulfate, alkali and lead nitrate, the process involves the addition of tartaric acid and H_2O_2 to the solution. Data are presented on the effects of peroxide, sodium sulfate, and tartaric acid on the sulfate precipitation. Possible mechanisms of the improved separation are discussed. (D.L.C.)

948 (HW-75537) REPORT OF INVENTION—SOLVENT EXTRACTION PROCESS FOR RECOVERY OF STRONTIUM, CERIUM, RARE EARTHS, AND CESIUM FROM RADIOACTIVE WASTE SOLUTIONS. Lane Allan Bray (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Nov. 5, 1962. Contract [AT(45-1)-1350]. 14p. Dep.(mn); $1.00(cy), 1(mn) OTS.

A solvent extraction process is described which removes Sr, Ce, Cs, and rare earths from acid solutions (pH 3 to 5) in one operation. The process is based on the use of a solvent composed of 4-sec-butyl-2-(α-methylbenzyl)phenol and di-2-ethylhexyl-phosphoric acid in a diluent such as SOLTROL-170. Data are presented on the extraction of Sr, Ce, and Cs by the process under varying conditions. (D.L.C.)

949 (HW-76449) ION EXCHANGE SEPARATION OF CESIUM FROM ALKALINE WASTE SUPERNATANT SOLUTIONS. J. L. Nelson and B. W. Mercer (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Jan. 30, 1963. Contract AT(45-1)-1350. 31p.

Cesium separation from Purex alkaline supernatant waste by ion exchange with inorganic zeolites or phenolic resins was investigated. High cesium loadings were obtained with AW-400, a synthetic zeolite manufactured by the Linde Company. A disadvantage of the AW-400 is the poor quality of the binder; efforts are being made to obtain a more suitable binder. Optimum loading conditions appear to be a flow rate of one to two column volumes of actual waste per hour, with 20-50 mesh exchanger at a temperature of 25°C. A scrub or sodium removal step can be accomplished with several solutions; dilute ammonium carbonate solutions are preferred for aged alkaline supernatant wastes. Cesium is eluted with a solution of ammonium carbonate and ammonium hydroxide. Exchangers that are most selective for cesium and have the highest loadings also require the greatest elution volumes. Ammonium carbonate can be volatilized by heating, leaving a cesium solution that is very satisfactory for subsequent packaging operations. Under practical conditions a Cs/Na DF of 10^2 and cesium recovery of >99% can be obtained. (auth)

950 (HW-78987(Rev.)) SEPARATION OF TRIVALENT RARE EARTHS FROM CERIUM. L. A. Bray and F. P. Roberts (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Oct. 1963. Contract AT(45-1)-1350. 17p.

A process was developed for the separation of Ce from the rare earths. The integrated process includes the coextraction of the rare earths and Ce(III) into 0.4M di(2-ethylhexyl)phosphoric acid (D2EHPA), diluted with 0.2M tri-n-butyl phosphate (TBP) and kerosene, from an aqueous feed solution containing a suitable complexing agent and pH range to free the wanted products from other ions. The Ce-rare earth loaded organic phase is then contacted either batch-wise or in a continuous contactor with an aqueous solution containing 0.2M potassium persulfate oxidant, 0.005 to 0.02M silver catalyst, and 1 to 2M nitric acid. Under these conditions, Ce is oxidized to Ce(IV) and remains in the organic phase while Pm and the other trivalent rare earths are stripped into the aqueous phase. Other concentrations of persulfate, silver, and HNO_3 were also found to be usable, but less favorable separations were obtained for batch extraction conditions. Following phase separation, pure Ce(IV) undergoes reduction and is removed from the organic phase (by contracting with a dilute acid) as a highly purified product. Pm and the other rare earths (now free from Ce), which have stripped into the aqueous solution, are readily available for further processing by a combination of concentration, solvent extraction, and ion exchange. The process produces a Ce product of high chemical and radiochemical purity re-

quiring minimum purification before packaging. The trivalent rare earth product containing Pm will be free from isotopes which generate penetrating radiations and large amounts of heat, thus minimizing the problems of heat dissipation and shielding. The product is also suitable as a feed for rare earth separation processes using organic ion exchange resins. (auth)

951 (HW-77609(Sect.4)) CHEMICAL EFFLUENTS TECHNOLOGY. (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). 60p.

A group of several zeolites was examined for the ability to remove small amounts of Sr or Cs selectively from aquous wastes containing high concentrations of Na salts. Methods for the study of soil permeability are discussed. The use of electrical analog simulation to quantitatively analyze large, complex, saturated ground water flow systems is described. The use of scintillation and temperature probes in geophysical investigations and measurements of ground water flow is discussed. The removal of I^{131} from process gaseous exhaust streams, the hazards of accidentally heating recovered Cs^{137}, and fission product release from heated uranium are discussed. (C.H.)

952 (HW-80876) SOLVENT EXTRACTION OF CESIUM FROM HANFORD ALKALINE SUPERNATE BY 4-SEC-BUTYL-2 (α-METHYLBENZYL)PHENOL — LABORATORY STUDIES. L. A. Bray (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Mar. 1, 1964. Contract AT(45-1)-1350. 21p. Dep.(mn); $1.00(cy), 1(mn) OTS.

A cesium solvent extractant, 4-sec-butyl-2(α-methylbenzyl) phenol, has been investigated for possible use at Hanford. The substituted phenol, diluted with a kerosene, was found to be a very selective extractant for cesium, both in the laboratory and, subsequently, in pilot-plant-scale pulse column demonstrations. The extractant provides an efficient method for cesium recovery from the very large volumes of alkaline supernate now being stored at the separations facilities. The extraction of cesium was found to depend upon the concentration of phenol, the pH of the aqueous waste solution, the temperature of the equilibration, and the aqueous Na^+ concentration. Approximately 0.3 of a mole of additional OH^- was required per liter of synthetic alkaline supernate (9M Na^+, pH 10.3) for maximum cesium extraction. Cesium distribution values are favored by a decrease in the equilibration temperature. The heat of extraction, $-\Delta H$, was found to be approximately 19 kcal. The cesium distribution ratio was also found to be inversely dependent on the sodium concentration with a slope of slightly less than −2 over the range of 1 to 8M sodium. (auth)

953 (HW-81091) THE CSREX PROCESS FOR CO-EXTRACTION OF CESIUM, STRONTIUM, AND RARE EARTHS — PILOT PLANT STUDIES. G. L. Richardson (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Mar. 2, 1964. Contract AT(45-1)-1350. 50p. Dep.(mn); $2.00(cy), 2(mn) OTS.

The CSREX process was successfully demonstrated in pulse columns containing the same nozzle plate cartridges recommended for use with the D2EHPA process, and it was concluded that the same equipment could be used interchangeably for either process. The preferred IA column cartridge contained 23% free area stainless steel nozzle plates, the other columns contained 10% free area nozzle plates, all on two-inch spacing. Demonstrated volume velocities were over 800 gph/ft^2 for the IA and IB columns and over 500 gph/ft^2 for the IS and IC columns. Higher capacities were possible but infeasible because of low efficiency at high rates. Performance was studied in runs made at ambient temperature under near-optimum operating conditions. The feed was a tartrate-complexed Purex acid waste. A 50% reduction in flow rates had little effect on the HTU's or losses; however, a 50% rate increase nearly tripled the HTU's in the IB column and increased the IA column strontium and cerium HTU's by 30 to 70%. Strontium recovery from a synthetic Purex alkaline sludge waste was also demonstrated. The acidified sludge was first put through a lead sulfate carrier precipitation process to concentrate strontium and to remove iron and other impurities. EDTA (ethylenediaminetetraacetic acid) was then added to complex lead, and the feed was processed as before. Flowsheets and equilibrium data for both types of wastes are presented. The major limitation to the use of the CSREX process at the present time is the unusually high susceptibility of the solvent to radiation and nitrous acid degradation. Sulfamic acid and hydrazine have been successfully used in non-radioactive solutions to inhibit nitrous acid formation and solvent degradation, but their performance in preventing degradation in a radiation field has not yet been satisfactorily demonstrated. Other than loss of solvent, the major effect of degradation of pilot plant solvent appeared to be decreased efficiency of strontium and cesium stripping in the IB column. (auth)

954 (HW-82771(Pt.1)) RECOVERY OF ALUMINUM NITRATE NONAHYDRATE FROM REDOX ACID WASTE. PART 1: COMPUTER STUDY. W. L. Godfrey and R. D. Benham (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). July 1964. Contract AT(45-1)-1350. 18p.

A counter-current crystallization process was developed and demonstrated on the laboratory scale for recovering aluminum nitrate nonahydrate from Redox waste. Equations describing this process were developed and simulated on a conventional analog computer (a Beckman EASE 1132) and an iterative analog computer (a Beckman EASE 2133). The results from both computers checked satisfactorily with laboratory data. Slow "control-relay" operating speeds associated with the EASE 1132 computer caused timing problems; as a result, reproducible solutions were obtained only by using excessively long computer cycle times. The speed of the control relays on the EASE 2133 computer allowed solution of the problem about 600 times faster. (auth)

955 (HW-83461) ION EXCHANGE RECOVERY OF CESIUM FROM ALKALINE SUPERNATANT WASTE — SCALE-UP STUDIES. R. P. Popovich (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Sept. 1964. Contract AT(45-1)-1350. 45p. Dep.; $3.00 (OTS).

A study was conducted on the operating parameters (kinetics and scale-up factors) for the ion exchange removal of cesium with the inorganic zeolite, AW-500. A synthetic Purex alkaline supernatant waste was used with a 2-in. inside diameter variable height column. The study resulted in the development of a model from which the breakthrough history of a tracer concentration solution may be estimated to within 10% over a wide range of operating parameters. The design parameters of 2.0 gal/min/ft^2 and 2.0 column volumes per hour have been specified for the removal of cesium from the waste. The estimated breakthrough history is defined for these parameters with AW-500. It was found that greater than 60 column volumes of influent can be processed with less than 1% accumulated breakthrough. This is well within the established design limits and indicates the feasibility of reducing the design volume of the column by 30%. (D.L.C.)

956 (HW-84105) REMOVAL OF CESIUM FROM REDOX ALKALINE SUPERNATANT WASTES BY ION EX-

CHANGE. K. C. Knoll (Hanford Atomic Products Operation, Richland, Wash.). Oct. 1, 1964. Contract AT(45-1)-1350. 16p. Dep.(mn); $1.00(cy), $(mn) CFSTI.

A flow sheet was developed for separating cesium from Redox alkaline waste supernate which uses Duolite C-3 resin as an ion exchange material. The waste is diluted with three parts of water and loaded onto the resin column at 25C, at a flow rate of 2 column volumes an hour. The column is washed with water to remove all waste solution and the cesium is then eluted from the resin with 3N $(NH_4)_2CO_3 \cdot 1N\ NH_4OH$. The resin is regenerated in the column with sodium hydroxide which also increases the pH of the bed to prevent aluminum hydroxide precipitation upon further contact with the waste. The resin is then ready for reloading. This flow sheet worked successfully with synthetic waste and with supernate from the Redox 115-SX tank. The radiation stability of the C-3 resin was measured by loading a sample with about 5 curies of ^{137}Cs. After 14 days the cesium was eluted and the irradiated resin, along with a control resin column, was loaded with ^{137}Cs. About 15 percent loss of capacity was observed for the test column. The radioactivity to which the resin in the test column was exposed was greater than that to which it would be subjected in plant practice by a factor of 3000. (auth)

957 (MLM-734) BISMUTH RECOVERY PROCESS. Engineering Research Final Report. W. J. Raiff, P. M. Engle, and M. McEwen (Mound Lab., Miamisburg, Ohio). Aug. 5, 1952. Contract AT(33-1)-Gen-53. 45p. Dep.(mn); $2.00(cy), 1(mn) CFSTI.

Waste solutions from the concentration cells of the primary process of Mound Laboratory are disposed of by removing the bismuth as an oxychloride sludge and allowing the liquid wastes to enter the general plant waste-disposal system. This removal operation calls for an annual expenditure with no return. It also calls for the scrapping of bismuth, which is a critical and valuable metal. The bismuth-recovery program is designed to salvage this valuable metal. It also provides for the recovery of bismuth from the accumulated oxychloride sludge from past operations. The bismuth-recovery program promises not only to save money on current operations when compared to the present system, but also to produce an income while the stored sludge is being recovered. The process adopted for the bismuth recovery, after consideration of pyrometallurgy and electrochemical reduction, is the electrodeposition of the bismuth from solution. This process is completed with the melting and casting of the metal in the required slug form. These slugs will then be reirradiated and recycled through the process so that the requirements of this project for new bismuth will be greatly reduced. Due to the small size of the proposed plant, a full-size plant is proposed with all necessary developmental changes to be made on the full plant itself. The best location for this plant is the "HH" Building, where the necessary installation can be made with a minimum of interference to present operations. Flowsheet, equipment layout plan, charts, specifications for equipment, and a general process description for plant operation for the electrolytic section of the plant are included, along with a history and justification. (auth)

958 (ORNL-3518) RECOVERY OF FISSION PRODUCTS FROM WASTE SOLUTIONS BY SOLVENT EXTRACTION. D. E. Horner, K. B. Brown, D. J. Crouse, and B. Weaver (Oak Ridge National Lab., Tenn.). Dec. 9, 1963. Contract W-7405-eng-26. 39p.

Increasing industrial, military, and space requirements for certain fission products place emphasis on large-scale methods for their recovery from waste liquors produced by the fuel recovery processes. With development of organic extractants for separating and recovering these fission products, solvent extraction offers a versatile alternative to ion-exchange, precipitation, and crystallization procedures heretofore used for small-scale recoveries. A process flowsheet was developed for the recovery of strontium and mixed rare earths from adjusted Purex 1WW solution by the use of di(2-ethylhexyl)phosphoric acid (D2EHPA). After adding tartrate to complex the iron and caustic to adjust the pH to 6, strontium and rare earths are co-extracted and then co-stripped with nitric acid in the first cycle. In additional D2EHPA extraction cycles, they are isolated as concentrated fractions. The mixed rare earths from this process can be treated by tributyl phosphate (TBP) extraction to separate the individual rare earth elements of which promethium and cerium are of greatest interest. Zirconium–niobium also may be recovered with D2EHPA from acidic waste liquors. An amine pretreatment extraction process was developed for removing nitric acid, iron sulfate, zirconium–niobium, and ruthenium from waste solutions to furnish a feed liquor more amenable to subsequent processes for recovering fission products. The separate recovery of ruthenium, zirconium–niobium, and rare earths may also be possible by amine extraction. A process based on the use of substituted phenols was developed for recovering cesium from alkaline waste solutions. These compounds are selective extractants for cesium, giving high decontamination from other fission products and sodium, and they are readily stripped with dilute nitric acid. Previous studies also outlined methods for recovering neptunium, plutonium, and technetium with solvents similar to those described. When appropriately combined, these developments offer a versatile integrated solvent extraction flowsheet for the recovery of all important fission products and other components from waste liquors. (auth)

959 (ORNL-3765) RECOVERY OF ^{137}Cs FROM FISSION-PRODUCT WASTES AND TRANSPORT BY AN ALUMINOSILICATE ION EXCHANGER. R. E. Lewis, T. A. Butler, and E. Lamb (Oak Ridge National Lab., Tenn.). May 1965. Contract W-7405-eng-26. 15p. Dep.; $1.00 (cy), 1(mn) CFSTI.

A method for recovering fission-product cesium from aged basic fission-product waste with an aluminosilicate ion exchanger (Decalso) was developed and demonstrated. Cesium is selectively adsorbed on Decalso from Hanford Purex Supernatant 103A, a basic 9 M $NaNO_3$ solution, and concentrated by a factor of about 8.5 The Decalso column is transported from HAPO to ORNL in shielded containers formerly used for shipping solutions, and the cesium is eluted with ammonium salts for purification and source fabrication. In the period April 1961 to August 1964, 2 Mcuries of ^{137}Cs was recovered from Hanford wastes and transported to ORNL by this procedure. (auth)

960 (ORNL-TM-889) ISOTOPES DIVISION QUARTERLY REPORT, JANUARY–MARCH 1960. J. H. Gillette (Oak Ridge National Lab., Tenn.). June 1964. Contract W-7405-eng-26. 41p.

Routine production of ^{14}C, ^{137}Cs, ^{60}Co, ^{131}I, ^{192}Ir, and ^{32}P is reported. Special samples of ^{131}I, ^{113}Sn, ^{55}Fe, ^{63}Ni, ^{125m}Te, ^{147}Pm, and ^{234}Th were prepared; methods are reviewed. Shutdown installation and maintenance of the Fission Products Pilot Plant are discussed. The macroscopic cross section for the reaction $^{147}Pm(n,\gamma)^{148}Pm$ was determined for both 5.2- and 42.0-day ^{148}Pm. Percent fission yields for neodymium isotopes were determined. Inorganic ion exchange materials were tested for ^{137}Cs removal from Savannah River waste, which contained sodium ions at 6M: Decalso (an aluminosilicate) was found to perform satis-

factorily. Minerals were tested for their efficiency in removing ^{137}Cs, ^{90}Sr, and ^{106}Ru from dilute alkaline wastes prior to ground disposal. The removal of ruthenium tetraoxide from air by aqueous scrubbing is discussed. Pm-147 separation by solvent extraction, ^{99}Tc purification by solvent extraction, preparation of Tc-metal by H_2 reduction of the salt, preparation of strontium titanate by precipitation, and determination of ^{33}P in ORR-produced ^{32}P are discussed. Electromigration methods and results for ^{137}Cs separation are described. Developments and revisions in the F3P cells during shutdown are discussed. Calutron separations of indium, barium, lead, and chromium isotopes are described. High-purity-chemistry techniques are discussed. (R.E.U.)

961 (ORNL-TM-890) ISOTOPES DIVISION QUARTERLY REPORT, JANUARY–MARCH 1961. J. H. Gillette (Oak Ridge National Lab., Tenn.). June 1964. Contract W-7405-eng-26. 37p.

Routine processing of ^{14}C, ^{137}Cs, ^{60}Co, ^{131}I, ^{192}Ir, ^{85}Kr, ^{32}P, and ^{99}Tc, and nonroutine processing of ^{14}C, ^{131}I, ^{125}I, ^{55}Fe, ^{204}Pb, ^{147}Pm, ^{90}Sr, ^{125m}Te, and ^{137}Cs are reported. The effect of sodium concentration on the batch separation of strontium and sodium was investigated for a typical strontium, calcium, and barium feed; iron extraction from the feed prior to strontium removal was studied. Foam separation process developments are discussed. Iodine-125 was produced from ^{124}Xe adsorbed on coconut charcoal. The separation of ^{147}Pm from ^{147}Sm by DTPA ion exchange is described. Conditions were established for electroplating plutonium on platinum. Uranium was electroplated on platinum and stainless steel. The properties of Decalso (an aluminosilicate cation exchanger) were studied as a component of leach-resistant glass radiation sources; leaching tests were run at room temperature on 0.1-mC sources of ^{137}Cs, ^{89}Sr, ^{204}Tl, and ^{144}Ce. Technetium chemistry is discussed. Some equipment modifications for the Fission Products Development Laboratory are described. The operation of a thermal diffusion column for ^{36}Ar, and the development for ^{21}Ne, ^{13}C, ^{85}Kr, and ^{124}Xe are reported. Calutron operations for the period covered are discussed: isotopes included are Ca, Ni, Dy, K, Yb, Zr, Cl, Mg, and Sn. (R.E.U.)

962 (RFP-396) RECOVERY OF PLUTONIUM FROM INCINERATOR ASH. F. E. Butler, J. D. Moseley, and G. F. Molen (Dow Chemical Co. Rocky Flats Div., Golden, Colo.). July 31, 1964. Contract AT(29-1)-1106. 22p.

A method for the recovery by dissolution of Pu from ash produced by combustion of Pu-contaminated burnables is described. The ash should be finely ground and completely oxidized prior to dissolution. Nitric acid and fluoride ion are the dissolution agents. The dissolution rate, which is directly proportional to the fluoride ion concentration, increases rapidly with temperature. (auth)

963 (RFP-442) RECOVERY AND PRELIMINARY PURIFICATION OF AMERICIUM OBTAINED FROM WASTE MATERIAL. P. G. Hagan and H. W. Miller (Rocky Flats Div., Dow Chemical Co., Golden, Colo.). Mar. 29, 1965. Contract AT(29-1)-1106. 11p. Dep.; $1.00(cy), 1(mn) CFSTI.

Americium was recovered from miscellaneous residues by leaching with hot concentrated hydrochloric acid. The solution was filtered, diluted to about 5% acidity, and precipitated as an oxalate. The oxalate was then calcined to americium oxide. Since the oxide contained a considerable amount of lead, calcium, cerium, potassium, and plutonium, further purification was necessary. The method employed utilized both simple cation-anion ion exchange and precipitation techniques. Ammonium thiocyanate was used as both an eluting and a complexing agent. The americium oxide content of the product was increased from approximately 42 to 99%. (auth)

964 (RFP–551) RECOVERY OF PLUTONIUM FROM MISCELLANEOUS ANALYTICAL LABORATORY WASTES. Molen, G. F. (Rocky Flats Div., Dow Chemical Co., Golden, Colo.). Sept. 24, 1965. Contract AT(29-1)-1106. 28p. Dep. mn; CFSTI $2.00 cy, $0.50 mn.

A method for the recovery of plutonium from miscellaneous analytical laboratory wastes is described. Both oxalate precipitation and anion exchange were investigated as means of recovery. Oxalate precipitation is the preferred method of recovery.

965 (RL-SA–40) STATUS OF FISSION PRODUCT RECOVERY AT HANFORD. Beard, S. J.; Smith, P. W. (General Electric Co., Richland, Wash. Hanford Atomic Products Operation). Aug. 16, 1965. Contract AT(45-1)-1350. 20p. (CONF–650902-11). Dep. mn; CFSTI $1.00 cy, $0.50 mn.

From American Chemical Society, 150th National Meeting, Atlantic City.

The recovery of fission products was initiated at Hanford in August 1960 with the production of strontium-90 and cesium-137. The program has since included production of custom quantities of cerium-144 and promethium-147. The current program involves separation of strontium and rare earths from Purex acid waste by lead carrier sulfate precipitation and subsequent separation of rare earths from strontium by oxalate precipitation. Strontium is further purified by di(2-ethylhexyl)phosphoric acid solvent extraction. Cesium-137 is recovered from aged alkaline wastes by processing the alkaline supernatant solution through a bed of alumino-silicate resin. Several other processes have been demonstrated with full level process solutions in plant equipment. Cesium has been recovered from alkaline wastes by zeolite ion exchange and nickel ferrocyanide precipitation and from acid wastes by phosphotungstic acid precipitation. Technetium-99 has been recovered and purified by anion exchange. Cerium-144 has been separated from the rare earths by di(2-ethylhexyl)phosphoric acid solvent extraction using silver catalyzed persulfate oxidation of cerium(III) to cerium(IV). Promethium-147 was subsequently separated from the cerium-free rare earths by chromatographic ion exchange. Future plans include the recovery of megacurie quantities of Sr, Cs, Ce, and Pm in a waste processing plant scheduled for startup in 1967. (auth)

966 (RL-SA–53) PLANT SCALE DEMONSTRATION OF PROMETHIUM RECOVERY AND PURIFICATION PROCESSES. Ritter, G. L.; Buckingham, J. S. (General Electric Co., Richland, Wash. Hanford Atomic Products Operation). Sept. 17, 1965. Contract AT(45-1)-1350. 39p. (CONF-651010–1). Dep. mn; CFSTI $2.00 cy, $0.50 mn.

From Electrochemical Society, 6th Regional AIChE Meeting, Union, Wash.

A slurry from an underground radioactive waste storage tank containing about 0.003 weight percent promethium-147 (total cation basis) was processed for recovery of the promethium. The promethium obtained was semi-purified by solvent extraction techniques to yield 75,000 curies. Approximately 45,000 curies of promethium at 98 weight percent purity was obtained from a chromatographic ion exchange final purification process. Process technology demonstrated during this effort included: (a) a solvent extraction flowsheet for the successful separation of prome-

thium and other rare earths from principally iron, lead, and strontium contaminants; (b) a solvent extraction flowsheet for the removal of cerium, including the extremely radioactive cerium-144 and its decay daughter praseodymium-144, from the other rare earths; and (c) a chromatographic ion exchange system capable of handling kilocurie quantities of promethium. Results of the test program have been helpful in defining areas for additional laboratory investigations and in designing flowsheets and equipment for the large-scale recovery and purification of promethium at Hanford.

967 (RL-SEP-518) SOLVENT EXTRACTION RECOVERY OF AMERICIUM WITH DIBUTYL BUTYL PHOSPHONATE. Kingsley, R. S. (General Electric Co., Richland, Wash. Hanford Atomic Products Operation). June 21, 1965. Contract AT(45-1)-1350. 35p. Dep. mn; CFSTI $2.00 cy, $0.50 mn.

A flow sheet for the recovery of americium from the Plutonium Reclamation Facility's aqueous waste (CAW) stream was developed and demonstrated in the laboratory. A study of the basic recovery chemistry showed the following: CAW can be neutralized to <0.05M H^+ without irreversible precipitation occurring. An extractant containing 30 vol% DBBP–70 vol% CCl_4 is an effective organic solvent for americium recovery. Americium extraction coefficients obtained with DBBP containing as high as 36% dibutyl methyl phosphonate impurity are comparable with those obtained with 99% pure DBBP. Dilution of DBBP with TBP drastically lowers the americium extraction coefficient. Dilution of neutralized CAW with 0.3M HNO_3 caused the americium coefficient to decrease rapidly. The order of effectiveness of NO_3^- metal salts added to the CAW to increase salting strength is $LiNO_3$ > $Mg(NO_3)_2 \cdot 6H_2O$ > $Al(NO_3)_3 \cdot 9H_2O$ > $NaNO_3$ > $Ca(NO_3)_2 \cdot 4H_2O$. Below 0.8M HNO_3, americium extraction coefficients increase rapidly with decreasing CAW acidity. Less than 0.1M F^- in CAW does not affect americium extraction. $NaNO_3$ and $Al(NO_3)_3$ are effective scrub-deentrainment reagents. Water and dilute HF are excellent agents for stripping americium from 30 vol% DBBP–70 vol% CCl_4. Greater than 90% of the plutonium entering a dibutyl butyl phosphonate solvent extraction flow sheet is retained by the organic phase through subsequent extraction-strip cycles.

968 THE CONCENTRATION OF ^{231}Pa FROM URANIUM PROCESSING WASTES. Vikt. I. Spitsyn, and R. A. D'yachkova. At. Energ. (USSR), 16: 134-7(Feb. 1964). (In Russian)

The carrying of trace amounts of ^{231}Pa on phosphoric acid resins, MnO_2, SnO_2, Al_2O_3, Sb_2O_3, and Cr_2O_3 was determined in 0.1, 0.25, 0.5, 1, 2, and 3N HNO_3. MnO_2 (10 mg/ml) was found to be the best carrier for ^{231}Pa, even in the presence of large amounts of extraneous salts such as iron (30 g/l), Al (15 g/l), Ca (35 g/l), and Mg (6 g/l). The amount of iron, Al, or Cu carried down with the MnO_2 carrier rapidly decreased with increasing acidity. It is recommended that ^{231}Pa be carried on MnO_2 from a >1N HNO_3 solution. A sharp separation of ^{231}Pa from manganese can be carried out by extracting the ^{231}Pa into β-β'-dichlorodiethyl ether from a 9.0N HCl solution. The manganese remains in the aqueous phase. Tributylphosphate (TBP) also extracts ^{231}Pa quantitatively from a 2 to 8N HNO_3 solution. A total of 98.4% ^{231}Pa is carried by a precipitate of zirconium salicylate, while only 4.5% of the titanium is carried down under these conditions (2.5N H_2SO_4 + 1% H_2O_2). The salicylate method was the most suitable for separating Zr from Ti. (TTT)

969 INVESTIGATION OF ADSORPTION OF TECHNETIUM FROM AQUEOUS SOLUTIONS ONTO RUSSIAN-MADE ANION EXCHANGE RESIN. Ryabchikov, D. I.; Pozdnyakov, A. A. (Inst. of Geochemistry and Analytic Chemistry, Academy of Sciences, USSR). Dokl. Akad. Nauk SSSR, 161: 896-8(Apr. 1, 1965). (In Russian).

Technetium separation from various waste aqueous solutions in the nuclear energy industry was studied. Adsorptive properties of some anion exchange resins were examined. Separation coefficients for technetium isotopes in acid solution were determined. Concentrations of technetium isotopes were measured by monitoring γ and β radiation. Isotopic equilibration was achieved after 50 to 60 min contact between the solution and the anion exchange resin. One anion exchange resin extracted technetium from weakly acidic, neutral, and basic concentrated solutions of $NaNO_3$, from solutions of HCl of any concentration, and also from 6 to 7M of H_2SO_4 and from 3 to 4M HNO_3. Because of small adsorption of technetium from perchlorate solutions, these solutions could be used for desorbing technetium from the anion exchange resin. All technetium can be desorbed from a resin to a 1.5 to 2M aqueous solution of $HClO_4$. Adsorption of technetium on strongly basic anion exchange resins was reversible. High sorptivity of TcO_4^- ions was due to small hydration of these ions.

970 A NEW INORGANIC ION EXCHANGER FOR THE SEPARATION OF CESIUM FROM HIGH LEVEL RADIOACTIVE WASTES. Balon, G.; Beaudet, C. A.; Piret, J. (SERAI, Brussels). EuroNuclear, 3: 85-8(Feb. 1966).

Studies were made to determine the resistance of titanium phosphate (ABEDEMTiA) to gamma radiation and its selective retention of cesium from five different reprocessing solutions. It was found to be possible to perform at least four complete cycles on titanium phosphate with reprocessing solutions that are similar to Marcoule's or Hanford's wastes. The yields at breakthrough were increased for the second and third cycles. This increase was explained as resulting from a saturation of the column with ammonium ions from the eluent after the first elution.

971 SEPARATION OF CESIUM FROM FISSION-PRODUCT WASTES BY ION EXCHANGE ON AMMONIUM MOLYBDOPHOSPHATE. Smit, J. Van R.; Jacobs, J. J. (National Chemical Research Lab., Pretoria). Ind. Eng. Chem., Process Design Develop., 5: 117-22(Apr. 1966).

A process is described for the large-scale recovery of ^{137}Cs from acid fission product solutions by selective adsorption on columns of the crystalline inorganic ion exchanger ammonium 12-molybdophosphate (AMP). Specially prepared coarse AMP was used, which allowed suitably high flow rates. At a linear flow rate of 2.5 cm per min with Butex 150-fold waste concentrate, relatively shallow beds of AMP showed a saturation capacity of about 0.65 equivalents of cesium per liter. Cesium was recovered from the column, after washing out impurities with dilute nitric acid, by stripping with hot concentrated ammonium nitrate solution, followed by chloride-catalyzed destruction of ammonium nitrate in nitric acid solution. Principal advantages of the process are small size of equipment, the small quantities of active waste produced, and applicability to virtually any nonalkaline waste, including those of very high acid or salt content.

972 THE EXTRACTION OF MOLYBDENUM FROM RADIOACTIVE WASTES. I. L. Jenkins and A. G. Wain (Atomic Energy Research Establishment, Harwell, Berks, Eng.). J. Appl. Chem. (London), 14: 449-54(Oct. 1964).

973 AN ALUMINOSILICATE ION EXCHANGER FOR RECOVERY AND TRANSPORT OF ^{137}Cs FROM FISSION-PRODUCT WASTES. Lewis, R. E.; Butler, T. A.; Lamb, E. Nucl. Sci. Eng., 24: 118-22(1966). (ORNL-P-1411).

A method for recovering fission-product cesium from aged basic fission-product waste with an aluminosilicate

ion exchanger (Decalso) was developed and demonstrated. Cesium is selectively absorbed on Decalso from Hanford Purex Supernatant 103A, a basic 9 M $NaNO_3$ solution, and concentrated by a factor of about 8.5. The Decalso column is transported from HAPO to ORNL in shielded containers formerly used for shipping solutions, and the cesium is eluted with ammonium salts for purification and source fabrication. In the period April 1961 to August 1964, 2MCi of ^{137}Cs was recovered from Hanford wastes and transported to ORNL by this procedure.

SOLID WASTES

GENERAL

See also the following references:

18, 35, 153, 162

974 (HW-79393) FEASIBILITY OF REMOTELY SEAL WELDING FISSION PRODUCT WASTE CONTAINERS. L. J. Rousseau (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Nov. 1, 1963. Contract AT(45-1)-1350. 21p.

A study was conducted to determine the feasibility of remotely seal welding stainless steel caps to filler necks for fission-product waste containers. Complete remote application of the welding process, high temperatures in the weld area, gas pressure inside the container, weld joint contamination, and reproducibility of the welds were considered. It was concluded that repetitive welds can be readily made under the conditions considered. (M.C.G.)

BALING

See also the following references:

34, 82, 87, 121, 242

975 NEW DEVICE FOR PROCESSING SOLID WASTE AT THE SACLAY NUCLEAR RESEARCH CENTER. Cerre, Pierre; Mestre, Emile; Barral, Samuel; Lebrun, Pierre (CEN, Saclay, France). Bull. Inform. Sci. Tech. (Paris), No. 99, 73-82(Dec. 1965). (In French).

The services which produce radioactive waste nearly always use drums for its storage and transportation to the processing installation. The large number of drums containing metals, screenings and materials contaminated by fission products coming mainly from the dismantling of the Fontenay-aux-Roses plutonium plant has led to development of a press capable of compressing the drum and its contents. This installation which makes it possible to obtain a large reduction factor and to reduce considerably the cost of processing solid waste is described.

BURIAL AND STORAGE

See also the following references:

5, 19, 24, 34, 36, 77, 79, 80, 87, 95, 97, 98, 121, 122, 129, 140, 148, 155, 156, 163, 166, 196, 209, 210, 214, 215, 217, 234, 238, 243, 244, 247, 248, 570, 872, 887, 888, 889

976 (BMwF-FBK-65-03) THEORETISCHE UNTERSUCHUNG DES GROESSTEN ANZUNEHMENDEN UNFALLES BEI DER LAGERUNG HOCHRADIOAKTIVER SPALTPRODUKTE IN SALZFORMATIONEN. (Theoretical Investigation of the Maximum Credible Accident with Regard to the Storage of Highly Radioactive Fission Products in Salt Formations). Brandes, S. (Technische Hochschule, Aachen (West Germany). Institut fuer Elektrische Anlagen und Energiewirtschaft). Oct. 1965. 47p. Dep. mn.

In order to obtain information for dimensioning salt cavities filled with waste, a maximum accident was considered. The spherical heap of fuel elements separated from salt by a layer of air will reach a temperature of more than 100°C so that water, which runs slowly into the cavity, will boil and evaporate. The temperature distributions are calculated for stationary and instationary cases with a given maximum temperature of 100°C in the waste. Fuel elements of the Schulten-type were taken, for those will probably be the first, which might be taken for storage. Because the formation of critical arrangements of waste in salt is possible, the critical radius of the sphere was calculated and found to be 2.9 m.

977 (DPSPU-62-30-3A) BENTONITE AS A PROTECTIVE COVER FOR BURIED RADIOACTIVE WASTE. Hawkins, R. H. (Du Pont de Nemours (E.I.) and Co., Aiken, S. C. Savannah River Plant). May 1962. Contract AT(07-2)-1. 9p. (CONF-32-3). Dep. mn; CFSTI $1.00 cy, $0.50 mn.

From Health Physics Society 7th Annual Meeting, Chicago, June 1962.

Laboratory tests, lysimeter tests, and field tests indicate that bentonite is an effective barrier to the movement of soil moisture. As such it appears to offer improved protection of buried solid radioactive waste from this primary cause of radionuclide migration. Bentonite studies are being continued. (auth)

978 (HW-77274) BURIAL OF HANFORD RADIOACTIVE WASTES. G. E. Backman, P. C. Jerman, R. E. Roberts, and J. W. Vanderbeek (General Electric Co., Richland, Wash. Hanford Atomic Products Operation). Apr. 15, 1963. Contract AT(45-1)-1350. 13p. Dep.(mn); $1.00(cy), 1(mn) CFSTI.

The current policies and procedures and future plans of Hanford relative to the disposal of solid radioactive waste are reported. The feasibility and practicability of burial ground consolidation are discussed and maps are included that indicate the location of both active and retired burial grounds.

979 (KFK-205) SCHRIFTTUMSUBERSICHT UBER ENDLAGERUNG RADIOAKTIVER ABFALLSTOFFE IN SALZFORMATIONEN. (Literature Search on Ultimate Disposal of Radioactive Wastes in Salt Formations). H. Ramdohr (Kernforschungszentrum, Karlsruhe, Germany). Mar. 1964. 43p.

A bibliographic study was made of the disposal of radioactive wastes in underground salt formations. The physical properties of salt and rock salt were given. The type and occurrence of rock salt is considered. The possibilities of deposition in rock salt by preparation of caverns or installation of a mine are examined. The practical research on the deposition of wastes in salt is reviewed, and the cost of such depositions was evaluated. Conclusions on the feasibility of waste disposal in salt formations for Germany were drawn from the data presented. 82 references. (J.S.R.)

980 (KFK-357) STUDIENGRUPPE FUER TIEFLAGERUNG RADIOAKTIVER ABFAELLE. Jahresbericht,

1964. (Study Group for Deep Storage of Radioactive Waste. Annual Report, 1964). Krause, H.; Ramdohr, H. (Kernforschungszentrum, Karlsruhe (West Germany)). May 1965. 26p. Dep. mn.

Studies on the development of safe and economic methods for the long-term removal of radioactive waste and for the design and construction of an installation for the final collection and processing of radioactive wastes are reported. The basic concepts used in the determination of the requirements for long-term storage of radioactive wastes are reported. Experimental work on the storage of radioactive wastes in salt caverns is described. The economic and safety factors to be considered in the disposal of waste under the sea were also considered.

981 (LADC-5940) EVALUATION OF TEMPERATURE RISES IN RADIOACTIVE WASTES. E. F. Gloyna, M. Asce, H. J. Longley, and C. W. Christenson (Los Alamos Scientific Lab., N. Mex.). [1962]. Contract [W-7405-eng-36]. 50p.

Generalized numerical and analytical methods were developed for estimating rises in temperatures resulting from the storage of high-level wastes. Parametric relationships were considered, and solutions for the packed-bed, multi-region, variable-geometry, and variable-effective thermal conductivity system were evaluated. Solutions of the heat equation in one space dimension were obtained for plane, cylindrical, and spherical geometries. Three computer codes were developed, tested for accuracy, and compared. The codes represent three approximations to the actual physical situation. An auxiliary code was written to provide an estimate of the waste that could be put into a cavity when a maximum permissible temperature is specified. The accuracy of the solutions depends chiefly on the values of the input data and the variations of the parameters affecting the finite mesh and boundary conditions. The size of storage sites, concentration of wastes, and age of wastes have the greatest effect on the magnitude of the temperature. (auth)

982 (ORNL-3697(p.19-31)) DISPOSAL IN NATURAL SALT FORMATIONS. W. J. Boegly, Jr., R. L. Bradshaw, F. M. Empson, W. C. McClain, F. L. Parker, W. F. Schaffer, Jr., and Hisashi Kubota (Oak Ridge National Lab., Tenn.).

Renovation of the topside facilities at the Carey Salt Mine, Lyons, Kansas, is essentially complete. Drilling has started on the 19.1-in.-ID waste-charging shaft. Cleanup of the main access tunnels in the mine is in progress. Plastic flow gages are being installed along the access tunnels and near the experimental area to determine the effect of increased temperature on stability of the mine. Design and fabrication of equipment for handling and storing the radioactive fuel assemblies is essentially complete. Laboratory tests of 1000-hr duration on pillar models made from Lyons salt were run at pillar stresses ranging from 4000 to 10,000 psi. From the results of these tests, an empirical equation was derived which correlates well with measured vertical closure rates in the Lyons and Hutchinson mines in openings up to 70 years old. The isolated $2 \times 8 \times 10$ ft heated model room (wall temperature approximately 140°C), after more than a year of heating, has a closure rate ten times higher than it had at ambient temperature. (auth)

983 (ORNL-3697(p.54-61)) ENGINEERING, ECONOMIC, AND SAFETY EVALUATIONS. R. L. Bradshaw, D. G. Jacobs, J. J. Perona, J. O. Blomeke, and K. E. Cowser (Oak Ridge National Lab., Tenn.).

Costs of permanent storage of calcined radioactive wastes in concrete vaults were five to seven times greater than those for storage in salt, and costs for storage in rooms mined out of granite formations were twice those for storage in salt mines. This economic advantage, plus greater safety, makes salt the preferred choice. Excavation costs were calculated as a function of the amount of salt mined yearly, and $2/ton was found to be a reasonable figure for a relatively large operation. Estimates were made of the movement of ^{90}Sr, ^{106}Ru, and ^{137}Cs in Conasauga shale following a hypothetical release of reactor fuel-reprocessing waste from a tank. Computations were based on hydrologic and exchange characteristics of the local formation. In Conasauga shale, ^{90}Sr would apparently move quite rapidly in an acid waste system, even if the acid would be neutralized by calcite ($CaCO_3$) in the formation. Movement of ^{90}Sr in an alkaline waste solution would be much slower due to the formation of insoluble compounds, in addition to the greater holdup by ion exchange. Ruthenium-106 would be expected to move rapidly in either acid or neutralized waste systems, but it presents little long-term hazard due to its relatively short half-life. The movement of ^{137}Cs would be expected to be slow because of its strong fixation by the ion exchange sites of the illitic fraction of the Conasauga shale. (auth)

984 (ORNL-P-1064) DISPOSAL OF HIGH-LEVEL POWER REACTOR WASTES IN A SALT MINE. W. J. Boegly, Jr., R. L. Bradshaw, F. M. Empson, W. C. McClain, F. L. Parker, and W. F. Schaffer, Jr. (Oak Ridge National Lab., Tenn.). [1964]. Contract [W-7405-eng-26]. (CONF-650203-3). Dep.(mn); $2.00(cy), 1(mn) CFSTI.

From National AIME Meeting, Chicago.

The major conclusions of previous studies on the disposal of high-level liquid and solid wastes in salt mines are outlined. A field experiment called Project Salt Vault is being planned to demonstrate the feasibility of such disposal. The experiment, using irradiated fuel assemblies in lieu of actual wastes, will be conducted in the Carey salt mine at Lyons, Kansas. The objectives, design, instrumentation, etc., of the experiment are described. (D.L.C.)

985 ACCELERATED DEFORMATION OF ROCK SALT AT ELEVATED TEMPERATURE. Lomenick, T. F.; Bradshaw, R. L. (Oak Ridge National Lab., Tenn.). Nature, 207: 158-9(July 10, 1965). (ORNL-P-1163).

Uniaxial tests of scale-model specimens of rock salt were performed to predict the effects on rock-salt mine stability of supporting-pillar stresses and elevated temperatures, in order to be able to design a radioactive waste disposal facility in a salt mine. Cylindrical specimens were tested for axial loads of 2000, 4000, 6000, 8000, and 10,000 lb/in^2 and at 22.5, 60, and 100°C, for a few hours to several thousand hours. It was found that pillar deformation increased markedly with increasing load, and creep rates of the salt were greatly accelerated at elevated temperatures. Sample data are presented. The results were fitted by empirical equations which predict strain rate and cumulative deformation. (M.J.T.)

986 DISPOSAL OF HIGH ACTIVITY POWER REACTOR WASTES IN SALT MINES: A CONCEPT AND FIELD SCALE DEMONSTRATION. Bradshaw, R. L.; Blomeke, J. O.; Boegly, W. J. Jr.; Empson, F. M.; Parker, F. L.; Perona, J. J.; Schaffer, W. F. Jr. (Oak Ridge National Lab., Tenn.). Nucl. Struct. Eng., 2: 438-46(Oct. 1965). (ORNL-P-1202).

One of the most important advantages of a natural salt formation as a repository for radioactive wastes is its essential impermeability due to its plastic properties. To prove the feasibility of disposal in salt mines a conceptual design has been developed and a demonstration of the storage concept is being prepared in a 1000 ft deep inactive mine in Lyons, Kansas. Economic studies show that costs for an actual disposal operation would fall well within the range allowable for competitive nuclear power. In the dem-

onstration irradiated fuel assemblies from the Engineering Test Reactor will serve as a source of radiation in lieu of actual solidified wastes. Two radioactive arrays (one in the existing floor to check problems in the use of abandoned mines, and one in a specially excavated area) will each contain 14 assemblies in 7 cans. An electrically heated array will serve as a control. Mine renovations and additions are complete, and initiation of the two-year test is scheduled for late 1965.

987 (ORNL-P-1568) DEMONSTRATION OF DISPOSAL OF HIGH-LEVEL RADIOACTIVE SOLIDS IN SALT. Empson, F. M.; Boegly, W. J. Jr.; Bradshaw, R. L.; McClain, W. C.; Parker, F. L.; Schaffer, W. F. Jr. (Oak Ridge National Lab., Tenn.). [1964]. Contract W-7405-eng-26. 33p. (CONF-650554-1). Dep. mn; CFSTI $2.00 cy, $0.50 mn.

From 2nd Symposium on Salt, Cleveland.

A description is given of Project Salt Vault, a demonstration using irradiated ETR fuel assemblies and the inactive mine of the Carey Salt Company at Lyons, Kansas. The engineering and scientific objectives of Project Salt Vault are outlined. The instrumentation and the canning and shipment of the ETR fuel assemblies are described. The status of Project Salt Vault is given. (D.L.C.)

988 (ORNL-P-1987) PROJECT SALT VAULT: DESIGN AND DEMONSTRATION OF EQUIPMENT. Schaffer, W. F. Jr.; Boegly, W. J. Jr.; Parker, F. L.; Bradshaw, R. L.; Empson, F. M.; McClain, W. C. (Oak Ridge National Lab., Tenn.). [1964]. Contract W-7405-eng-26. 36p. (CONF-660208-4). Dep. mn. CFSTI $3.00 cy, $0.50 mn.

From Symposium on Solidification and Long-Term Storage of Highly Radioactive Wastes, Richland, Wash.

Handling, transfer, and storage techniques applicable to high-level solid waste disposal operations were successfully demonstrated with short-cooled ETR fuel assemblies. The design of Project Salt Vault equipment for the encapsulation of the fuel assemblies, cross country shipment, transfer from the surface to the mine level, movement within the mine, and transfer into storage is described. The operations are unique in that high-level radioactive materials are transferred from a shipping cask on the surface to the mine level 1000 feet below without the use of hot cells.

989 (ORNL-P-2058) PROJECT SALT VAULT: DESIGN AND OPERATION. Empson, F. M.; Bradshaw, R. L.; Boegly, W. J.; McClain, W. C.; Parker, F. L.; Schaffer, W. F. Jr. (Oak Ridge National Lab., Tenn.). [1965]. Contract W-7405-eng-26. 20p. (CONF-660208-15). Dep. mn. CFSTI $1.00 cy, $0.50 mn.

From Symposium on Solidification and Long-Term Storage of Highly Radioactive Wastes, Richland, Wash.

The demonstration disposal of high-level radioactive solids, initiated in November 1965 in the Carey salt mine at Lyons, Kansas, is known as Project Salt Vault. The objectives of the demonstration are: confirmation of feasibility and safety of disposal in salt mines; demonstration of required waste-handling equipment and techniques; determination of the stability of salt under the influence of heat and radiation; and collection of information on creep and plastic flow of salt, which is needed for the design of an actual disposal facility. Fourteen irradiated fuel assemblies from the Engineering Test Reactor serve as a source of radiation in lieu of actual solidified wastes. The assemblies, contained in seven cans, have been placed in an array of holes in the floor. During the course of the 2-year test, four sets of assemblies will be used to achieve a peak dose to the salt of about 8×10^8 rad and the temperature of the adjacent salt will be raised to 200°C with auxiliary electrical heaters. A second array will be installed in less pure salt at the end of the first 6 months, using the fuel assemblies which are now in the main array. A third array, consisting only of heaters, is being operated as a control to determine those effects due solely to heat. In addition to the radioactive and control arrays, a rib-pillar will be heated by electrical heaters at its base to produce additional information on salt flow characteristics at elevated temperatures.

990 (ORNL-TM-734) DEMONSTRATION DISPOSAL OF HIGH-LEVEL RADIOACTIVE SOLIDS IN LYONS, KANSAS, SALT MINE: BACKGROUND AND PRELIMINARY DESIGN OF EXPERIMENTAL ASPECTS. R. L. Bradshaw, J. J. Perona, and J. O. Blomeke (Oak Ridge National Lab., Tenn.). Jan. 10, 1964. Contract W-7405-eng-26. 41p.

A demonstration of the disposal of high-level radioactive waste solids to be carried out in a salt mine at Lyons, Kansas, will have as its objectives: (1) the demonstration of required waste-handling equipment and techniques, (2) the determination of the stability of salt under the influence of heat and radiation, and (3) the collection of information on creep and plastic flow of salt which is needed for the design of an actual disposal facility. As presently conceived, 14 irradiated fuel assemblies from the Engineering Test Reactor will serve as a source of radiation in lieu of actual solidified wastes. The assemblies will be placed in a circular array of holes in the floor with one can in the center and other six cans located peripherally, spaced 5 ft on centers. During the course of the 2-year test, four sets of assemblies will be used to achieve a peak dose to the salt of about 8×10^8 rad and the temperature of the adjacent salt will be maintained at 200°C with electrical heaters. A second array, consisting only of heaters, will be operated as a control to determine those effects due solely to heat. In addition to the radioactive and control arrays, a rib-pillar located between the two arrays will be heated electrically around its base to produce significant information on salt flow characteristics at elevated temperatures.

991 (ORNL-TM-1327) GEOHYDROLOGICAL EVALUATION OF SOLID WASTE STORAGE AREA 6. Lomenick, T. F.; Wyrick, H. J. (Oak Ridge National Lab., Tenn.). Dec. 1965. Contract W-7405-eng-26. 16p. Dep. mn. CFSTI $1.00 cy, $0.50 mn.

The proposed site of Solid Waste Storage Area 6 in Melton Valley is underlain mostly by multicolored shales and interbedded silty limestones that are generally sufficiently weathered to permit easy excavations by power equipment. About 25% of the 70-acre tract is unsuitable for burial, since it lies in areas where the water table is within 6 ft of the land surface. In order to minimize leaching and the subsequent migration of radioactive materials from the buried waste, it is essential that the trenches be constructed so as to suspend the waste above the water table, minimize rainfall infiltration into the holes once they are covered, and provide maximum seepage rates of waters from the trenches.

992 (RL-SA-58) STORAGE OF LONG-LIVED FISSION PRODUCTS ON ALUMINO-SILICATE ZEOLITES. Mercer, B. W. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.); Schmidt, W. C. (General Electric Co., Richland, Wash. Hanford Atomic Products Operation). Dec. 3, 1965. Contract AT(45-1)-1851. 12p. (CONF-660208-3). Dep. mn. CFSTI $1.00 cy, $0.50 mn.

From Symposium on Solidification and Long-Term Storage of Highly Radioactive Wastes, Richland, Wash.

Technology for the encapsulation and storage of long-lived radioisotopes ^{137}Cs and ^{90}Sr was developed. This technology includes adsorption of the isotopes on alumino-silicate zeolites, drying the zeolites to prevent excessive pressure after a container is sealed, sealing the containers by welding, and elution of the containers to recover the isotopes. Several alumino-silicate zeolites that possess the chemical properties to adsorb these cations (1.5 to 2 equiv/kg) from the

mother liquor were investigated. Dehydration was found to be a satisfactory method of reducing the moisture content of the zeolite to less than 0.5 wt %. Purging the container with helium prior to final closure increased the thermal conductivity of the zeolite and permitted use of the mass spectrometer for detecting minute leaks after final closure. Final closure was made by tungsten inert gas welding to form a container which is capable of withstanding several hundred pounds per square inch pressure without yielding.

993 (RL-SA-60) HIGH INTEGRITY BULK PACKAGING FOR LONG-TERM STORAGE OF RADIOACTIVE WASTES. Kolb, J. W. (General Electric Co., Richland, Wash. Hanford Atomic Products Operation). Feb. 9, 1966. Contract AT(45-1)-1851. 67p. (CONF-660208-12). Dep. mn. CFSTI $3.00 cy, $0.75 mn.

From Symposium on Solidification and Long-Term Storage of Highly Radioactive Wastes, Richland, Wash.

Facilities for packaging megacurie quantities of radioactive wastes (strontium-90, cesium-137) in all welded stainless steel helium leak-tight containers for long-term storage (many decades) or retrieval at any time are presented. Unique features of the facilities include: remote operation in a separations plant canyon without the use of manipulators or dynamic penetration between the operating gallery and contaminated zone; remote encapsulation and replacement of equipment by canyon crane and impact wrench without shutdown, cell clean-up or contamination spread; equipment components except those that must contact container or waste stream located in clean zone; and remote clean-up of contaminated portion of equipment for subsequent contact maintenance at glove box.

994 (SRIA-8) AN ANALYSIS OF LAND DISPOSAL OF PACKAGED RADIOACTIVE WASTE IN SEVEN WESTERN STATES. Final Report. R. B. Hoy, E. M. Kinderman, and R. R. Tarrice (Stanford Research Inst., Menlo Park, Calif.). Sept. 30, 1959. Contract AT(04-3)-115. 120p.

A preliminary study was made of possible areas in Western U. S. which might be used for future land burial of solid or packaged low-level radioactive wastes. The volume and nature of wastes generated in the seven western states were investigated. Fourteen areas in the states were selected as the best areas for further studies for disposal of packaged wastes. The costs of land disposal were also analyzed and compared with those for sea disposal. (D.L.C.)

995 ENVIRONMENTAL MONITORING AROUND A LONG-TERM STORAGE PLACE FOR SOLID RADIOACTIVE WASTES IN CZECHOSLOVAKIA. Kortus, J. (Chemoprojekt, Prague). pp 55-62 of STI-PUB-98.

Experience in operating the central radioactive waste storage facility in Czechoslovakia and the procedures and equipment used for environmental control are described.

996 **PROJECT SALT VAULT: A DEMONSTRATION DISPOSAL OF HIGH-LEVEL RADIOACTIVE SOLIDS IN LYONS, KANSAS, SALT MINE. Boegly, W. J. Jr.; Bradshaw, R. L.; Empson, F. M.; Schaffer, W. F. Jr.; Parker, F. L.; Blomeke, J. O. (Oak Ridge National Lab., Tenn.). Health Phys., 12: 417-24(Mar. 1966).**

The demonstration of the disposal of high-level radioactive waste solids in salt mines is reported. Fourteen irradiated fuel assemblies from the Engineering Test Reactor will serve as a source of radiation, since high-level solidified wastes do not exist at the present time. The assemblies will be placed in a circular array of holes in the floor of a newly mined area (14 ft above the existing mine floor) with one can in the center and the other six cans located peripherally, spaced 5 ft on centers. During the course of the 2-year test, four sets of assemblies will be used to achieve a peak dose in the salt of about 8×10^8 rad and the temperature of the salt adjacent to the center hole will be maintained at 200°C with supplementary electrical heaters. A second radioactive array, located in the existing mine floor, will receive the fuel canisters which are removed from the main array at the end of each 6-month period. The purpose of this array is to study the problems that may be encountered if the waste containers are located in salt interbedded with water-bearing shale. A third array, consisting only of heaters, will be operated as a control to determine those effects due solely to heat. In addition to the radioactive and control arrays, a rib-pillar located between the main radioactive array and the control array will be heated electrically around its base to produce significant information on salt flow characteristics and mine stability at elevated temperatures. Preparation of the four experimental rooms will involve the mining of approximately 19,000 tons of salt. Placing of the first fuel assembly cans in the mine is expected to take place in November 1965.

997 HEAT GENERATION BY RADIOACTIVE WASTES. Bradshaw, R. Louis (Oak Ridge National Lab., Tenn.). J. Sanit. Eng. Div. Amer. Soc. Civil Eng., 90: 55-7(Oct. 1964).

A discussion is given of an article by Gloyna, et al. (ibid., 90: 43-66(Feb. 1964)), concerning the storage of radioactive wastes in salt mines, and additional data on the subject are provided, particularly of temperatures of such stored wastes. Equations have been developed which allow the calculation of temperature rises for the case of an infinite array of finite length cylindrical sources, an array of seven finite length cylindrical sources, and an isolation finite cylinder. These equations assume a line source to obtain the temperature rise in the medium (rock salt in this case) and at the wall of the hole containing the cylinder, and assume a constant salt wall temperature (400°F, typically) to obtain the temperature rise at the center of the waste cylinder. By these equations, it was possible to calculate the spacing required between waste cylinders to meet the criteria for both maximum allowable waste temperature and salt temperature. Calculations were performed for storage of wastes of various ages and compositions in 10-ft long cylinders of 6- to 24-in. in 50 by 300-ft rooms in a salt mine. The maximum allowable temperature at the center-line of the waste cylinder was taken to be 1650°F, and the maximum allowable salt temperature was taken as 400°F. For wastes less than 3 to 6 yr old at the time of storage, peak salt temperatures were reached in a few hundred to a few thousand hours, but, for older wastes, the peak salt temperatures were not reached for several years, and, in some cases, for several decades. Of the estimated costs for processing and ultimate disposal of high-activity power reactor wastes in a salt mine, total costs were found to be higher for disposal of the wastes in a form that produced a larger volume of solids per unit of fission-product activity.

998 A CONCEPT AND FIELD-SCALE TEST OF DISPOSAL OF HIGH-ACTIVITY POWER REACTOR WASTES IN SALT MINES. R. L. Bradshaw, J. O. Blomeke, W. J. Boegly, Jr., F. L. Parker, J. J. Perona, and W. F. Schaffer, Jr. (Oak Ridge National Lab., Tenn.). Trans. Am. Nucl. Soc., 7: 399-400(Nov. 1964).

999 CHEMICAL PROCESS DEVELOPMENT. p.130-6 of "Fundamental Nuclear Energy Research, 1964." Washington D. C., Atomic Energy Commission, 1964.

The development of chemical processing methods for nuclear reactor fuels is reviewed, with emphasis on the fuel-salvage, reconversion, credit, and useful-waste aspects. The processes reviewed include solvent extraction, volatility separation, pyrometallurgy, analysis and chemonuclear research. (T.F.H.)

CHEMICAL DISSOLUTION

See the following references:

50, 137

INCINERATION

See also the following references:

31, 32, 34, 87, 99, 107, 148, 185, 186, 389, 962

1000 (AECL-2248) CRNL ACTIVE WASTE INCINERATOR. D. W. McQuade (Atomic Energy of Canada Ltd., Chalk River, (Ontario)). Feb. 1965. 52p. Dep.; $1.00 (AECL).

At CRNL, the daily collection of 1200 pounds of active combustible waste is burned in a refractory lined multichamber incinerator. Capacity is 500–550 pounds per hour; volume reduction 96%. Combustion gases are cooled by air dilution and decontaminated by filtration through glass bags in a baghouse dust collector. A description is presented of the incinerator plant, its operation, construction and operating costs, and recommendations for future designs. (auth)

1001 (AEET–220) THE USE OF INCINERATORS FOR TREATMENT OF COMBUSTIBLE WASTES. Amalraj, R. V.; Khan, A. A. (Atomic Energy Establishment, Trombay (India)). 1965. 12p. Dep. mn.

The use of incinerators for waste disposal is discussed, and use of incinerators in India for various wastes is reported.

1002 WASTE INCINERATOR FOR BURNABLE PLUTONIUM WASTE (Engineering Materials). (Dow Chemical Co. Rocky Flats Plant, Devner). (CAPE-1182)

The incinerator is used as a production unit to efficiently recover plutonium values from wastes and to reduce the volume of necessary storage. The facility includes the incinerator, 3 heat exchangers, slag wool filters, tar sump, and a walk in plenum. The plenum facilitates changing filters and monitoring filter conditions. Also included are drawings describing a glove box, piping, and a fire detection system. Thirty-eight drawings are included. (P.C.H.)

1003 (HW-68788) INCINERATOR AND OFF-GAS SCRUBBER TESTS. George Rey (General Electric Co. Hanford Atomic Products Operation, Richland, Wash.). Mar. 9, 1961. Contract AT(45-1)-1350. 18p.

An experimental incinerator with an off-gas system consisting of a course fly ash remover, Peabody scrubber, and Absolute filter, was installed and tested in the 321 A Building. Typical "Z" plant combustible wastes (a mixture of rags, paper, wood and plastics, thoroughly shredded) were burned during the tests. Test conditions, coupled with a minimum equipment system, constituted a severe test of the incineration process. The process under the tested conditions has two major problem areas—(1) frequent filter pluggage with incinerator "soot" and (2) poor absorption of acidic vapors. If the recommendations can be feasibly incorporated into the existing "Z" plant design and operation these problems can be lessened, but would still predominate. Assuming optimum conditions, filter replacements can be expected on a weekly basis (8 hour day) and some five to ten percent of the nitric (HNO_3 + NO_2 mixture) vapors may not be absorbed. Greater than 99 percent of the acid vapors of chloride (possibly sulfates also) can be removed with the scrubber-filter combination. The estimate of weekly filter changes assumes the use of a secondary combustion chamber in the incinerator. To further delay filter pluggage a tertiary combustion may be required, or an exceptionally efficient secondary combustion, after which filter pluggage rates would be controlled by scrub liquor entrainment (40 lbs liquor 10^6 lbx off-gas). Use of in-place filter cleaning techniques would delay filter changes to one per month or two. An overall plutonium dust decontamination factor of 10^7 should be readily attained from incinerator to filter exhaust, with 99.9 percent remaining within the incinerator, mostly in the ash itself. (auth)

1004 (TID-7677(p.331-47)) A RADIOACTIVE WASTE INCINERATOR. Ernest W. Bloore and David G. Lachapelle (Army Nuclear Defense Lab., Edgewood Arsenal, Md.).

A 50 lb/hr incinerator for the concentration of low-level radioactive combustible waste was designed and fabricated. The experimental facility utilizes auxiliary gas firing to maintain high combustion gas temperatures of 2,000 to 2,500°F, and dry collection methods for gas cleaning. An evaluation program is being initiated. (auth)

1005 THE INCINERATION OF RADIOACTIVE WASTES AND ITS INDUSTRIAL ASPECTS. Jean Rodier, Guy Leifillatre, and Michel Charbonneaux. Energie Nucl., 7: 25-37(Jan.-Feb. 1965). (In French)

The incineration of radioactive waste considerably reduces their volume and this is especially necessary in countries with a small area and a high population density. The principles of the design of the incineration unit project at the Marcoule Centre (where the solid wastes are the most abundant) are reviewed along with the factors that determine the choice of the installation characteristics; the importance of the problems of corrosion, fire, and the necessity of purifying the gas formed are emphasized. A description of the installation is given as well as some data on the efficiency of the gas purifying system, on the operation of the installation in general, and on operational costs. (auth)

1006 THE USE OF COMMERCIAL INCINERATORS FOR THE VOLUME REDUCTION OF RADIOACTIVELY CONTAMINATED COMBUSTIBLE WASTES. H. Glauberman and P. Loysen (Atomic Energy Commission, New York). Health Phys., 10: 237-41(Apr. 1964).

AEC contractors were surveyed to determine the extent to which commercial incinerators are used in reducing the volume of contaminated combustible wastes. The incinerators in use are performing successfully both operationally and economically. Along with descriptions of the installation, data are presented for several installations showing initial costs, air dust measurements, volume reduction, and rentention of radioactivity in the ash. The main conclusion is that commercial incinerators are practical for the treatment of combustible wastes generated by fabricators of normal and enriched uranium fuel elements. Units are being operated with high reduction of waste volume, high retention of radioactivity in the ash and with the creation of no appreciable off-site air contamination problems. (auth)

1007 WASTE BURNING AT CHALK RIVER. CANADIAN INCINERATOR PLANT DEALS WITH 1200 lb/day. Nucl. Eng., 10: 472-3(Dec. 1965).

The procedures used for the incineration of low-activity waste (200 mR/hr maximum and about 5 mR/hr average)

at Chalk River are described. The incinerator plant layout, construction, and operation are discussed. The handling of flue gas, using baghouse filters to remove fly ash, is described; the performance of absolute filters, originally intended for use besides the baghouse filters, was unsatisfactory. Instrumentation used is mentioned briefly. Results of testing and operation are discussed. Costs of construction and operation of the plant are considered.

1008 RADIOACTIVE WASTE MATERIAL INCINERATOR. Berghout, Christian F.; Dauer, Maxwell; Lanard, Francis W.; Minx, Ramon P.; Senoski, Walter E.; Ward, Ural T. (to U. S. Army). U. S. Patent 3,218,997. Nov. 30, 1965. Filed Feb. 6, 1963.

In a proposed incinerator for combustible radioactive material, the combustion chamber has charging means for the waste material, fuel injection means, means to support the material, and air supply means. A controlled ash disposal means at the base of and contiguous to the combustion chamber includes an ash chute, a control valve in and for the ash chute, means for liquid spraying the ash in the chute, and dust removal means for disposal of the dust created by the combustion by discharging the contents of the combustion chamber. Dynamic precipitation means purify the waste gases emitted from the combustion chamber, and water cooled exhaust conduit means are provided between the combustion chamber and the dynamic precipitation means. A controlled precipitation disposal means are provided contiguous to the precipitation means and includes a valve controlled hopper and dust removal means. A radiation detector alarm and recording device is mounted in the stack and at the ash disposal means.

INCORPORATION IN CONCRETE

1009 COMBINED BURIAL OF SOLID AND LIQUID RADIOACTIVE WASTES. Sobolev, I. A.; Khomchick, L. M. Gigiena i Sanit., No. 4, 79-80(Apr. 1965). (In Russian).

Previously liquid and solid wastes were buried separately in underground concrete vaults, but this system suffered from a number of obvious disadvantages such as leakage danger and improper utilization of space (50 to 55%) in solid storage. In order to improve the waste handling system a concrete mixer was installed above the underground concrete vault. The solid wastes were placed in the vault as usual. Then the liquid radioactive wastes containing 10^{-4} to 10^{-6} C/l was mixed with portland cement, and the cement solution was poured on top of the solid radioactive waste. The space utilization increased from 50% to 80%. Each vault has a volume of 50 to 100 m^3. The cost of storing the combined waste was decreased by one-half. (TTT)

1010 A STATION FOR PROCESSING AND CONDITIONING OF SOLID RADIOACTIVE WASTES. P. Cerré and E. Mestre (Commissariat a l'Energie Atomique, Paris). Health Phys., 10: 651-4(Sept. 1964). (In French)

Processing and conditioning of solid radioactive wastes have been resolved at the Saclay Nuclear Research Center. The processing and conditioning plant is divided in three zones: an inactive zone where the wastes are conditioned by incorporation in a concrete block in order to constitute a monolith. Every operation is automatically performed and, in this report, details are given on the successive operations of waste coating; a semi-active zone where the wastes of various nature and activity are received and through which they are introduced into the active zone; an active zone where the wastes are processed by compacting and shearing and then introduced into baskets constituting the concrete block core. Details are given on the various stages. (auth)

1011 CONVERSION OF IRRADIATED FUEL SWARF TO A STABLE CERAMIC SUITABLE FOR LONG-TERM STORAGE. Hilton, D. A.; Buddery, J. H. (Berkeley Nuclear Labs., Gloucestershire, Eng.). Ind. Eng. Chem., Process Design Develop., 4: 400-3(Oct. 1965).

A method is described for the conversion of irradiated fuel swarf, of the type obtained in the postirradiation examination of uranium-Magnox fuel elements, into a stable ceramic by reaction with a 5 wt % cupric oxide—32 wt % alumina—63 wt % lead borate glass mixture. The process requires a minimum of handling and is carried out by a three-stage heating sequence for periods of 1 hr each at 350, 50^ and 700°C. Losses of fission products throughout the reaction were less than 20 ppm of the gross β and γ activities and the product formed was suitable for long-term storage. Brief details of the resistance of the ceramic to water leaching and radiation are given.

OCEAN DISPOSAL

See the following references:

19, 24, 34, 87, 844, 845, 859, 980

MISCELLANEOUS

NEUTRON BURNING

1012 (BNL-8558) NEUTRON BURNING OF LONG-LIVED FISSION PRODUCTS FOR WASTE DISPOSAL. Steinberg, Meyer; Wotzak, Gregory; Manowitz, Bernard (Brookhaven National Lab., Upton, N. Y.). Sept. 1964. Contract AT-30-2-GEN-16. 27p. Dep. mn; CFSTI $2.00 cy, $0.50 mn.

A discussion is presented of the economics of ^{85}Kr, ^{90}Sr, and ^{137}Cs waste processing by in-pile nuclear "burning" to yield stable or short-lived products. In the process, the Cs and Kr effluents from a reactor are fed into gaseous diffusion plants to increase the ^{137}Cs and ^{85}Kr contents, respectively, to 90%. ^{90}Sr is fed into the "burner" reactor directly, with no isotopic separation, along with the 90%-pure Cs and Kr radioisotopes. After suitable burning times, the irradiated radioisotopes are recycled through the separation processes and back into the "burner."

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ORAU	Available on interlibrary loan from the Library, Oak Ridge Associated Universities, Post Office Box 117, Oak Ridge, Tennessee 37830.
ORNL	Oak Ridge National Laboratory, Oak Ridge, Tennessee 37830.

(The symbols "mc", "mf", and "fs" are used to designate that copy is available only in microcard, microfilm, or facsimile form. The symbol "mn" is used to designate that copy is available in micronegative (microfiche) form. The symbol "cy" is used to designate full size or photographically enlarged copy (two-thirds of original size.)

Report No.	Reference	Availability
AAEC-TM		
307	434	Dep. (mn)
A/AC.82/G/L		
986	266	Dep. (mn)
A/CONF.28/P		
27	432	
86	81	
88	82	
187	83	
188	732	
252	937	Dep. (mn); $1.00(CFSTI)
282	84	Dep. (mn); $1.00(CFSTI)
287	810	Dep. (mn); $1.00(CFSTI)
587	733	
609	938	
714	433	
772	939	
774	693	
775	85	
783	887	
784	584	
794	86	
813	906	
814	907	
849	300	
868	7	Dep. (mn); $1.00(CFSTI)
869	87	Dep. (mn); $1.00(CFSTI)
870	301	
ACNP		
63628	435	Dep. (mn); $1.00(cy), $0.50(mn) CFSTI
65544(Vol.I)	80	Dep. (mn); $7.00(cy), $2.00(mn) CFSTI
AD		
404279	669	$7.60(CFSTI)
606056	338	$6.00(CFSTI) as AFWL-TDR-64-81
AE		
182	811	Dep. (mn)
AEC-tr		
6128(p.231-47)	436	Dep.$4.00(CFSTI)
6365	437	Dep. (mc); $1.60(fs), $0.80(mf) JCL
6459	397	Dep. (mc); $1.60(fs), $0.80(mf) JCL
6544	888	Dep. (mn); $1.60(fs), $0.80(mf) JCL
6549	889	Dep. (mn); $7.60(fs), $2.51(mf) JCL

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Report No.	Reference	Availability
AECL		
1918	741	Dep.; $0.50(AECL) as ERI-6
1975	91	Dep.; $0.50(AECL) as CRIO
2030	812	Dep.; $0.50(AECL)
2043	438	Dep.; $0.50(AECL)
2233	88	$0.50(AECL)
2248	1000	Dep.; $1.00(AECL)
2295	813	Dep.; $0.20(AECL)
AED-C-		
01-12	8	Dep.(mn)
01-13	9	Dep.(mn)
AEET		
220	1001	Dep.(mn)
AEET/HP/RWD		
3	814	Dep.(mn)
4	165	Dep.(mn)
AEET/HP/TH		
20	694	Dep.(mn)
AEEW-R		
227	439	Dep.(mc)
AERE-R		
3880	690	Dep.(mn)
4460	512	Dep.; $0.70(BIS); 3s.6d.(HMSO)
4493	770	Dep.; $0.80(BIS); 4s.(HMSO)
4685	734	Dep.; $0.70(BIS); 3s.6d.(HMSO)
4905	585	Dep.; $1.00(BIS); 5s.(HMSO)
AFWL-TDR		
64-81	338	$6.00(CFSTI)
AHSB(RP)-R		
42	440	Dep.; $1.00(BIS); 5s.(HMSO)
ANL		
6938	441	Dep.; $6.00(cy), 5(mn)CFSTI
6945	267	Dep.; $3.00(cy), 2(mn)CFSTI
7020	940	Dep.; $6.00(CFSTI)
ANL-Trans		
76	442	Dep.(mn); $1.10(fs), $0.80(mf)JCL
126	443	Dep.(mn); $10.10(fs), $3.92(mf) JCL
AP-Note		
439	356	Dep.(mn); $1.00(cy), 1(mn)CFSTI
442	89	Dep.(mn); $2.00(cy), 1(mn)CFSTI
BLG		
267	941	Dep.(mn)
365(Vol.1)	444	Dep.(mn)
BMwF-FBK		
65-03	976	Dep.(mn)
BNL		
799	178-79	Dep.; $2.25(CFSTI)
832	546	Dep.; $0.75(CFSTI)
841(p.18-42)	180	Dep.; $4.00(cy), 2(mn)CFSTI
853	586	Dep.; $0.50(CFSTI)
859	445	Dep.; $0.50(CFSTI)
900	181	Dep.; $6.00(CFSTI)
914	446-52	Dep.(mn); $5.00(cy), $1.00(mn) CFSTI
915	268	Dep.(mn); $1.00(cy), $0.50(mn) CFSTI
8558	1012	Dep.(mn); $2.00(cy), $0.50(mn) CFSTI
8645	461	ORAU as CONF-641207-2
10047	670	Dep.(mn); $1.00(cy), $0.50(mn) CFSTI
10109	735	Dep.(mn); $2.00(cy), $0.50(mn) CFSTI

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Report No.	Reference	Availability
10132	736	Dep.(mn); $1.00(cy), $0.50(mn) CFSTI
BNWL		
4	942	Dep.(mn); $2.00(cy), 1(mn)CFSTI
36	182-83, 453	Dep.(mn); $8.71(cy), $2.00(mn) CFSTI
47	908	Dep.(mn); $3.00(cy), $0.50(mn) CFSTI
63	909	Dep.(mn); $1.00(cy), $0.50(mn) CFSTI
68	801	Dep.(mn); $1.00(cy), $0.50(mn) CFSTI
72	587	Dep.(mn); $2.00(cy), 1(mn)CFSTI
76	737	Dep.(mn); $3.00(cy), 1(mn)CFSTI
80	738	Dep.(mn); $2.00(cy), $0.50(mn) CFSTI
90	269	Dep.(mn); $6.00(cy), $1.25(mn) CFSTI
101	762	Dep.(mn); $1.00(cy), $0.50(mn) CFSTI
122	270, 302-3	Dep.(mn); $6.00(cy), $1.25(mn) CFSTI
139	739	Dep.(mn); $2.00(cy), $0.50(mn) CFSTI
157	695	Dep.(mn); $4.00(cy), $0.75(mn) CFSTI
165	271	Dep.(mn); $2.00(cy), $0.50(mn) CFSTI
170	696	Dep.(mn); $2.00(cy), $0.50(mn) CFSTI
180	588	Dep.(mn); $2.00(cy), $0.50(mn) CFSTI
187	802	Dep.(mn); $2.00(cy), $0.50(mn) CFSTI
213	547	Dep.(mn); $2.00(cy), $0.50(mn) CFSTI
243	910	Dep.(mn); $1.00(cy), $0.50(mn) CFSTI
BNWL-SA		
53	304	Dep.(mn); $1.00(cy), 1(mn)CFSTI
123	454	Dep.(mn); $1.00(cy), 1(mn)CFSTI
176	815	Water Resour. Res. 2, 31-9(1966)
219	305	Nucl. Safety 7, 92-9; 102(1965)
293	306	Dep.(mn); $1.00(cy), $0.50(mn) CFSTI
406	911	ORAU
429	455	Dep.(mn); $1.00(cy), $0.50(mn) CFSTI
453	697	Dep.(mn); $1.00(cy), $0.50(mn) CFSTI
465	272	Dep.(mn); $3.00(cy), $0.75(mn) CFSTI
466	816	Dep.(mn); $2.00(cy), $0.50(mn) CFSTI
467	698	Dep.(mn); $1.00(cy), $0.50(mn) CFSTI
504	943	Dep.(mn); $1.00(cy), $0.50(mn) CFSTI
514	699	Dep.(mn); $1.00(cy), $0.50(mn) CFSTI
564	740	Dep.(mn); $1.00(cy), $0.50(mn) CFSTI
569	548	Dep.(mn); $1.00(cy), $0.50(mn) CFSTI
623	307	Dep.(mn); $1.00(cy), $0.50(mn) CFSTI
CAPE		
1100	719	EM
1147	868	EM
1150	549	EM
1182	1002	EM
1269	90	EM
CEA		
2237	398	Dep.(mc)
2331	700	Dep.(mc)

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Report No.	Reference	Availability
CEA-Bib		
53	308	Dep. (mn)
59	10	Dep. (mn)
CEA-R		
2553	513	Dep. (mn)
2851	589	Dep. (mn)
2894	869	Dep. (mn)
2908	401	Dep. (mn)
CEA-tr-A		
1413	701	Dep. (mc); $1.60(fs), $0.80(mf) JCL
1462	771	Dep. (mc); $3.60(fs), $1.13(mf) JCL
1490	166	Dep. (mn); $5.60(fs), $1.94(mf) JCL
CEND		
512	456	Dep. (mn); $4.00(cy), $1.00(mn) CFSTI
CENRD/HP		
62/2	184	Dep. (mc); $1.10(fs), $0.80(mf) CFSTI
CONF		
12-3	399	Dep. (mc); $1.60(fs), $0.80(mf) CFSTI; Gmelin, AED-CONF-62-105-135
32-3	977	Dep. (mn); $1.00(cy), $0.50(mn) as DPSPU-62-30-3A
173-3	469	Dep. (mc); $1.60(fs), $0.80(mf) CFSTI as HW-SA-3192
179-1	590	Dep. (mc); $1.60(fs), $0.80(mf) CFSTI; J. Inorg. Nucl. Chem. 27, 473-9(1965)
179-5	591	Dep. (mc); $1.60(fs), $0.80(mf) CFSTI; Gmelin, AED-CONF-64-033-26
179-10	592	ORAU. Gmelin, AED-CONF-64-033-16
196-1	807	Dep. (mc); $1.60(fs), $0.80(mf) CFSTI as HW-SA-3211
196-2	600	Dep. (mc); $2.60(fs), $0.83(mf) CFSTI as HW-SA-3233
243-33	457	ORAU. Gmelin, AED-CONF-63-168-34
CONF		
243-40	390	ORAU. Gmelin, AED-CONF-63-168-44
255-7	731	Dep. (mc); $1.10(fs), $0.80(mf) CFSTI as NLCO-894
255-9	470	Dep. (mc); $1.60(fs), $0.80(mf) CFSTI as HW-SA-3220
337-5	368	$3.00(cy), 1(mn)CFSTI
337-7	273	Dep. (mn)
337-11	357	ORAU
337-23	458	ORAU
337-41	459	Dep. (mn); $1.00(cy), 1(mn)CFSTI
337-48	371	ORAU
337-54	460	ORAU
337-55	400	ORAU
337-57	372	ORAU
365-1	344	ORAU
365-11	274	ORAU
365-13	275	ORAU
365-18	309	ORAU
365-19	276	ORAU
365-22	310	ORAU
365-23	311	ORAU
365-25	277	ORAU
393-3	817	ORAU
402-1	278	ORAU
488-7	550	ORAU
516-1	708	Dep. (mc); $3.60(fs), $1.25(mf) CFSTI as LADC-6398
561-1	358	Dep. (mc); $2.60(fs), $0.89(mf) CFSTI; Health Phys. 10, 899-907 (1964) as HW-SA-3434

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Report No.	Reference	Availability
566-2	593	Soil Sci. Am. Proc. 29, 25-8(1965)
635-1	1	Dep.(mc); $1.60(fs), $0.80(mf) CFSTI as HW-SA-3080(Rev.)
642-2	818	ORAU . Gmelin, AED-CONF-64-082-12
642-7	345	Dep.(mn); $1.00(cy), 1(mn)CFSTI; Gmelin, AED-CONF-64-082-11
642-29	471	Dep.(mn); $1.00(cy), 1(mn)CFSTI; Gmelin, AED-CONF-64-082-50; Health Phys. 11, 1009-15(1965) as HW-SA-3338
642-31	730	Dep.(mn); $1.00(cy), 1(mn)CFSTI as DPSPU-64-30-8B
644-1	551	ORAU
652-2	473	Dep.(mn); $1.00(cy), 1(mn)CFSTI as HW-SA-3685
672-1	250	ORINS. Gmelin, AED-CONF-64-077-1
708-1	95	Dep.(mn); $1.00(cy), 1(mn)CFSTI as HW-SA-3575
721-2	518	Dep.(mn); $2.00(cy), 1(mn)CFSTI as ORNL-P-447
727-4	287	Dep.(mn); $1.00(cy), 1(mn)CFSTI as NLCO-933
772-1	830	Dep.(mn); $1.00(cy), 1(mn)CFSTI as ORNL-P-605
620911(Vols.1,2,3)	819	New York, Pergamon Press, 1964
631104-9	93	Dep.(mn) as IEA-102
631005-1	168	Dep.(mn) as RT/PROT(65)-12
640413	820	Seminar on Protection Against Tritium Hazards, Le Vesinet, France, Apr. 16-18, 1964; Dep.(mn)
640506-1	514	Dep.(mn) as EUR-2118.f
640511	446	Dep.(mn); $5.00(cy), $1.00(mn) CFSTI as BNL-914
640530-1	312	Dep.(mn) as EUR-2407.f
641103-1	472	Dep.(mn); $1.00(cy), 1(mn)CFSTI as HW-SA-3683
641207-2	461	ORAU
650103-6	339	Dep.(mn); $2.00(cy), 1(mn)CFSTI; Gmelin, AED-CONF-65-001-8
650203-3	984	Dep.(mn); $2.00(cy), 1(mn)CFSTI as ORNL-P-1064
650407	373,402-4	International Symposium on Fission Product Release and Transport Under Accident Conditions, Oak Ridge, Tenn., Apr. 1965
650407-10	454	Dep.(mn); $1.00(cy),1(mn)CFSTI as BNWL-SA-123
650407-14	401	Dep.(mn) as CEA-R-2908
650515-2	821	ORAU
650554-1	987	Dep.(mn); $2.00(cy), $0.50(mn) CFSTI as ORNL-P-1568
650603-2	704	Dep.(mn); $2.00(cy), $0.50(mn) CFSTI as HW-SA-3604
650604-4	88	$0.50(AECL); p.667-76 of International Symposium on Working Methods in High Activity Hot Laboratories. Vol. II. Paris, Organization for Economic Cooperation and Development, 1965 as AECL-2233
650616-2	925	Dep.(mn); $1.00(cy), 1(mn)CFSTI as RL-SA-15
650616-4	304	Dep.(mn); $1.00(cy), 1(mn)CFSTI as BNWL-SA-53
650902-10	765	Dep.(mn); $1.00(cy), $0.50(mn) CFSTI as RL-SA-41
650902-11	965	Dep.(mn); $1.00(cy), $0.50(mn) CFSTI as RL-SA-40
650918-7	809	Dep.(mn); $1.00(cy), $0.50(mn) CFSTI as ORNL-P-1603
651010-1	966	Dep.(mn); $2.00(cy), $0.50(mn) CFSTI as RL-SA-53
651014-1	306	Dep.(mn); $1.00(cy), $0.50(mn) CFSTI as BNWL-SA-293
651101-35	428	Dep.(mn); $1.00(cy), $0.50(mn) CFSTI as RL-SA-50

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Report No.	Reference	Availability
651202-1	92	Dep.(mn); $1.00(cy), $0.50(mn) CFSTI as DPSPU-65-30-30
651202-2	121	Dep.(mn); $1.00(cy), $0.50(mn) CFSTI as UCRL-14320
651202-3	103	Dep.(mn); $1.00(cy), $0.50(mn) CFSTI as LA-DC-7249
651202-4	108	Dep.(mn); $1.00(cy), $0.50(mn) CFSTI as RFP-638
651202-10	11	ORAU
651202-11	20	Dep.(mn); $3.00(cy), $0.50(mn) CFSTI as ORNL-TM-1289
651202-32	912	ORAU
651202-44	12	ORAU
651207-1	911	ORAU as BNWL-SA-406
660208-1	878	Dep.(mn); $1.00(cy), $0.50(mn) CFSTI as ORNL-P-1926
660208-2	879	Dep.(mn); $1.00(cy), $0.50(mn) CFSTI as RL-SA-62
660208-3	992	Dep.(mn); $1.00(cy), $0.50(mn) CFSTI as RL-SA-58
660208-4	988	Dep.(mn); $3.00(cy), $0.50(mn) CFSTI as ORNL-P-1987
660208-5	697	Dep.(mn); $1.00(cy), $0.50(mn) CFSTI as BNWL-SA-453
660208-6	698	Dep.(mn); $1.00(cy), $0.50(mn) CFSTI as BNWL-SA-467
660208-7	699	Dep.(mn); $1.00(cy), $0.50(mn) CFSTI as BNWL-SA-514
660208-8	764	Dep.(mn); $1.00(cy), $0.50(mn) CFSTI as ISO-SA-1
660208-11	670	Dep.(mn); $1.00(cy), $0.50(mn) CFSTI as BNL-10047
660208-12	993	Dep.(mn); $3.00(cy), $0.75(mn) CFSTI as RL-SA-60
660208-13	548	Dep.(mn); $1.00(cy), $0.50(mn) CFSTI as BNWL-SA-569
660208-14	740	Dep.(mn); $1.00(cy), $0.50(mn) CFSTI as BNWL-SA-564
660208-15	989	Dep.(mn); $1.00(cy), $0.50(mn) CFSTI as ORNL-P-2058
660208-16	735	Dep.(mn); $2.00(cy), $0.50(mn) CFSTI as BNL-10109
660208-17	736	Dep.(mn); $1.00(cy), $0.50(mn) CFSTI as BNL-10132
660305-3	3	Dep.(mn); $1.00(cy), $0.50(mn) CFSTI as RL-SA-59
660305-4	2	Dep.(mn); $1.00(cy), $0.50(mn) CFSTI as ORNL-P-1944
660305-5	944	Dep.(mn); $1.00(cy), $0.50(mn) CFSTI
660305-6	943	Dep.(mn); $1.00(cy), $0.50(mn) CFSTI as BNWL-SA-504
660401-1	455	Dep.(mn); $1.00(cy), $0.50(mn) CFSTI as BNWL-SA-429
660507-1	836	Dep.(mn); $1.00(cy), $0.50(mn) CFSTI as RLO-1725-41
660507-4	323	Dep.(mn); $2.00(cy), $0.50(mn) CFSTI as ORNL-P-2021
660507-5	307	Dep.(mn); $1.00(cy), $0.50(mn) CFSTI as BNWL-SA-623
CRIO		
1194	91	Dep.; $0.50(AECL)
DP		
844	890	Dep.; $2.25(CFSTI)
876	594	Dep.; $0.50(CFSTI)
877	772	Dep.; $0.50(CFSTI)
908	391	Dep.; $1.00(cy), 1(mn)CFSTI
942	729	Dep.; $1.00(cy), 1(mn)CFSTI
966	870	Dep.(mn); $2.00(cy), $0.50(mn) CFSTI
990	871	Dep.(mn); $2.00(cy), $0.50(mn) CFSTI
1028	405	Dep.(mn); $2.00(cy), $0.50(mn) CFSTI

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Report No.	Reference	Availability
DPSPU		
62-30-3A	977	Dep.(mn); $1.00(cy), $0.50(mn) CFSTI
63-30-42	251	Dep.(mn); $1.60(fs), $0.80(mf) CFSTI
64-30-2	279	Dep.(mn); $1.00(cy), 1(mn)CFSTI
64-30-8B	730	Dep.(mn); $1.00(cy), 1(mn)CFSTI
65-30-30	92	Dep.(mn); $1.00(cy), $0.50(mn) CFSTI
EIR		
56	872	Dep.(mc)
ERI		
6	741	Dep.; $0.50(AECL)
ETR		
69	104	Dep.(mn); $1.00(cy), 1(mn)CFSTI as NP-8519
140	672	as NP-13289
173	480	Dep.(mn); $2.00(cy), $0.50(mn) CFSTI as NP-15754
EUR		
2118.f	514	Dep.(mn)
2199.e	595	Dep.(mn)
2407.f	312	Dep.(mn)
2481.f	913	Dep.(mn)
2497.e	596	Dep.(mn)
2535.f	773	Dep.(mn)
2754.f	945	Dep.(mn)
EURAEC		
202	185	Dep.(mc); $2.60(fs), $0.89(mf) CFSTI
300	186	Dep.(mc); $3.60(fs), $1.16(mf) CFSTI
HW		
55735	427	Dep.(mc); $1.10(fs), $0.80(mf) CFSTI
62455	463	Dep.; $0.50(CFSTI)
68788	1003	Dep.(mc); $1.60(fs), $0.80(mf) CFSTI
71589	946	Dep.(mn); $1.00(cy), 1(mn)CFSTI
72229(Rev.)	280	Dep.(mn); $1.00(cy), 1(mn)CFSTI
72335(Pt.2)	914	Dep.; $2.00(CFSTI)
72666	803	Dep.; $2.75(CFSTI)
74036	947	$1.00(cy), 1(mn)CFSTI
74040	536	$1.60(fs), $0.80(mf)CFSTI
75223	552	Dep.(mc); $3.60(fs), $1.34(mf) CFSTI
75537	948	Dep.(mn); $1.00(cy), 1(mn)CFSTI
76449	949	Dep.(mc); $3.60(fs), $1.13(mf) CFSTI
76973(Rev.)	582	Dep.; $0.50(CFSTI)
77163	94	Dep.(mc); $8.60(fs), $3.14(mf) CFSTI
77274	978	Dep.(mn); $1.00(cy), 1(mn)CFSTI
77380	537	Dep.(mn); $1.00(cy), 1(mn)CFSTI
77609	281,525, 950	Dep.; $4.00(CFSTI)
77979	553	Dep.(mn); $1.00(cy), 1(mn)CFSTI
78388(p.4.1-15)	187	Dep.; $2.50(CFSTI)
78395	282	Dep.; $0.75(CFSTI)
78987(Rev.)	951	Dep.; $1.00(cy), 1(mn)CFSTI
79140	188	Dep.; $1.00(CFSTI)
79174	597	Dep.; $2.00(CFSTI)
79279	313	Dep.(mc); $1.10(fs), $0.80(mf) CFSTI
79280(p.4.1-28)	189	Dep.; $2.75(CFSTI)
79393	974	Dep.(mc); $2.60(fs), $0.83(mf) CFSTI
79522	873	Dep.(mc); $2.60(fs), $0.98(mf) CFSTI
79622	583	Dep.; $0.75(CFSTI); Chem. Eng. Progr. 60, 74-82(1964)
79652	283	Dep.; $1.00(CFSTI)

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Report No.	Reference	Availability
79762(Pt.1)	804	Dep.; $1.50(CFSTI)
79762(Pt.2)	805	Dep.; $1.25(CFSTI)
80086	915	Dep.(mc); $1.60(fs), $0.80(mf) CFSTI
80204	464	Dep.(mc); $2.60(fs), $1.07(mf) CFSTI
80284(p.4.1-23)	190	Dep.; $2.50(CFSTI)
80500(p.205-8)	314	Dep.; $6.00(cy), 5(mn)CFSTI
80526	191	Dep.; $0.75(CFSTI)
80686	806	Dep.(mc); $2.60(fs), $0.83(mf) CFSTI
80848	874	Dep.(mc); $1.60(fs), $0.80(mf) CFSTI
80876	952	Dep.(mn); $1.00(cy), 1(mn)CFSTI
81091	953	Dep.(mn); $2.00(cy), 2(mn)CFSTI
81092	774	Dep.(mn); $1.00(cy), 1(mn)CFSTI
81481	13	Dep.(mc); $5.60(fs), $1.97(mf) CFSTI
81746	284,406,465-67, 822,916-17	Dep.; $7.00(cy), 6(mn)CFSTI; J. Appl. Meteorol. 3, 761-70 (1964)
81829	702	Dep.(mc); $2.60(fs), $0.98(mf) CFSTI
81862	703	Dep.; $1.00(CFSTI)
82771(Pt.1)	954	Dep.; $1.00(cy), 1(mn)CFSTI
83218	763	Dep.(mn); $2.00(cy), 1(mn)CFSTI
83298(p.4.1-25)	192	Dep.; $5.00(cy), 4(mn)CFSTI
83320	193	Dep.; $2.00(cy), 1(mn)CFSTI
83461	955	Dep.; $3.00(CFSTI)
83609	14	Dep.(mn); $2.00(cy), 1(mn)CFSTI
83610	15	Dep.; $2.00(cy), 1(mn)CFSTI
83614	823	Dep.; $2.00(cy), 1(mn)CFSTI
83896	554	Dep.; $4.00(CFSTI)
84105	956	Dep.(mn); $1.00(cy), 1(mn)CFSTI
84467	194	Dep.; $2.00(cy), 1(mn)CFSTI
84549	918	Dep.(mn); $1.00(cy), 1(mn)CFSTI
84577	919	Dep.(mn); $3.00(cy), 2(mn)CFSTI
84603	195,555-56,875	Dep.; $3.00(cy),1(mn)CFSTI
HW-SA		
2940	598	Dep.(mc); $1.60(fs), $0.80(mf) CFSTI; Am. Mineralogist 48, 1374-81(1963)
3080(Rev.)	1	Dep.(mc); $1.60(fs), $0.80(mf) CFSTI
3147	824	Limnol. Oceanog. 9, 540-5(1964)
3176	468	Dep.(mc); $2.60(fs), $0.92(mf) CFSTI
3192	469	Dep.(mc); $1.60(fs), $0.80(mf) CFSTI
3199	599	Dep.(mn); $2.00(cy), 1(mn)CFSTI
3211	807	Dep.(mc); $1.60(fs), $0.80(mf) CFSTI
3220	470	Dep.(mc); $1.60(fs), $0.80(mf) CFSTI
3229	465	J. Appl. Meteorol. 3, 761-70 (1964) as HW-81746(p.1.1-19)
3233	600	Dep.(mc); $2.60(fs), $0.83(mf) CFSTI
3279	16	Dep.(mc); $1.60(fs), $0.80(mf) CFSTI
3333	601	Dep.(mc); $2.60(fs), $1.10(mf) CFSTI
3338	471	Dep.(mn); $1.00(cy), 1(mn) CFSTI; Gmelin, AED-CONF-64-082-50; Health Phys. 11, 1009-15(1965)
3421	810	Dep.(mn); $1.00(CFSTI) as A/Conf.28/P/287
3434	358	Dep.(mc); $2.60(fs), $0.89(mf) CFSTI; Health Phys. 10, 899-907(1964)
3550	345	Dep.(mn); $1.00(cy), 1(mn)CFSTI; Gmelin, AED-CONF-64-082-11
3575	95	Dep.(mn); $1.00(cy), 1(mn)CFSTI
3604	704	Dep.(mn); $2.00(cy), $0.50(mn) CFSTI
3607	96	Isotopes Radiation Technol. 2, 99-123(1964-65)

Report No.	Reference	Availability
3683	472	Dep.(mn); $1.00(cy), 1(mn)CFSTI
3685	473	Dep.(mn); $1.00(cy), 1(mn)CFSTI
IDO		
12024	474	Dep.(mn); $4.00(cy), $1.00(mn) CFSTI
12032	374	Dep.(mc); $3.60(fs), $1.22(mf) CFSTI
12035	315	Dep.; $3.00(cy), 2(mn)CFSTI
12036	775	Dep.(mc); $2.60(fs), $0.92(mf) CFSTI
12037	196	Dep.; $3.00(cy), 2(mn)CFSTI
12040	97	Dep.(mn); $2.00(cy), 2(mn)CFSTI
12041	475	Dep.(mn); $2.00(cy), $0.50(mn) CFSTI
12042	557	Dep.(mn); $1.00(cy), 1(mn)CFSTI
14599	197	Dep.; $1.00(CFSTI)
14607	369	Dep.; $2.00(cy), 1(mn)CFSTI
14608	558	Dep.(mn); $4.00(cy), 3(mn)CFSTI
14616	198	Dep.; $0.75(CFSTI)
14618	559	Dep.; $2.50(CFSTI)
14620	560	Dep.; $5.00(CFSTI)
14621	199	Dep.; $1.00(CFSTI)
14622	561	Dep.; $1.75(CFSTI)
14624	602	Dep.; $1.50(CFSTI)
14625	200	Dep.; $1.25(CFSTI)
14626	562	Dep.; $0.50(CFSTI)
14627	563	Dep.; $2.75(CFSTI)
14629	564	Dep.; $0.50(CFSTI)
14631	565	Dep.; $2.00(cy), 1(mn)CFSTI
14633	566	Dep.; $1.00(cy), 1(mn)CFSTI
14635	201	Dep.; $1.00(CFSTI)
14636	202	Dep.; $6.00(cy), 3(mn)CFSTI
14638	203	Dep.; $0.75(cy), 1(mn)CFSTI
14646	204	Dep.; $2.00(cy), 1(mn)CFSTI
14653	567	Dep.(mn); $2.00(cy),$0.50(mn) CFSTI
14655	568	Dep.; $6.00(cy), 4(mn)CFSTI
14657	603	Dep.(mn); $1.00(cy), $0.50(mn) CFSTI
14666	569	Dep.(mn); $1.00(cy), $0.50(mn) CFSTI
14670	570	Dep.(mn); $1.00(cy), $0.50(mn) CFSTI
14672	742	Dep.(mn); $2.00(cy), $0.50(mn) CFSTI
16925	98	Dep.; $2.75(CFSTI)
22046	891	Dep.(mn); $6.00(cy), 5(mn)CFSTI
22047-USGS	892	Dep.(mn); $7.00(cy), $1.50(mn) CFSTI
IEA		
102	93	Dep.(mn)
INR		
609/XIX/D	604	Dep.
636/XIII/J	605	Dep.
IONAC		
5002	619	Dep.(mn); $3.00(cy), $0.75(mn) CFSTI as TID-22511
ISO-SA		
1	764	Dep.(mn); $1.00(cy), $0.50(mn) CFSTI
JAERI		
1048	99	
4025	17	
JPRS		
21816	316	$2.50(CFSTI)
23903	540	$0.50(CFSTI)
28343	476	$5.00(CFSTI)
34927	100	$1.00(cy), $0.50(mn)CFSTI
JUL		
249-RW	340	Dep.(mn); p.635-51 of International Symposium on Working Methods in High

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Report No.	Reference	Availability
		Activity Hot Laboratories. Vol. II. Paris, Organization for Economic Cooperation and Development, 1965
261-DE	101	Dep.(mn)
K		
1651	102	Dep.(mn); $1.00(cy), $0.50(mn) CFSTI
KFK		
205	979	Dep.(mc)
357	980	Dep.(mn)
KR		
86	705	Dep.(mn)
KY		
484	285	Dep.(mn); $1.00(cy), $0.50(mn) CFSTI
LA		
3150	706	Dep.; $1.00(cy), 1(mn)CFSTI
3363	392	Dep.(mn); $1.00(cy), $0.50(mn) CFSTI
LA-DC		
6106	477	$1.00(cy), 1(mn)CFSTI
6283	707	Ceram. Age 80, No. 8, 31-4; 37-41(1964)
7249	103	Dep.(mn); $1.00(cy), $0.50(mn) CFSTI
LADC		
5940	981	Dep.(mc); $4.60(fs), $1.70(mf) CFSTI
6398	708	Dep.(mc); $3.60(fs), $1.25(mf) CFSTI
6486	407	Rev. Sci. Instr. 35, 1534-5(1964)
LAMS		
2937	478	Dep.; $1.00(CFSTI)
MLM		
734	957	Dep.(mn); $2.00(cy), 1(mn)CFSTI
1151	515	Dep.; $2.60(fs), $1.04(mf)CFSTI
1152	205	Dep.; $2.60(fs), $0.86(mf)CFSTI
1155	206	Dep.; $4.60(fs), $1.49(mf)CFSTI
1210	286	Dep.; $1.00(cy), 1(mn)CFSTI
1266	207	Dep.(mn); $2.00(cy), $0.50(mn) CFSTI
NAVDOCKS-MO		
218	252	
NDL-TM		
20	671	
NLCO		
894	731	Dep.(mc); $1.10(fs), $0.80(mf) CFSTI
933	287	Dep.(mn); $1.00(cy), 1(mn)CFSTI
970	526	Dep.(mn); $3.00(cy), $0.75(mn) CFSTI
NP		
8519	104	Dep.(mn); $1.00(cy), 1(mn)CFSTI
13289	672	
13685	825	
13687	479	
13876	105	Dep.(mc)
14161	317	
14178	606	Dep.(mc)
15102	359	Dep.(mn)
15159	318	Dep.(mn)
15267	18	Dep.(mn); IAEA(Free)
15367	673	Dep.(mn)
15371	674	Dep.(mn)
15662	675	Dep.(mn)

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Report No.	Reference	Availability
15754	480	Dep.(mn); $2.00(cy), $0.50(mn) CFSTI
15859	893	Dep.(mn)
15874	676	Dep.(mn)
15916	319	Dep.(mn)
NRL		
5970	360	$1.10(CFSTI) as AD-421841
NYO		
841-1	208	Dep.(mn); $3.00(cy), 2(mn)CFSTI
841-2	389	Dep.(mn); $5.00(cy), $1.00(mn) CFSTI
2911-1	607	Dep.(mc); $4.60(fs), $1.64(mf) CFSTI
3086-1	167	Dep.(mn); $4.00(cy), $1.00(mn) CFSTI
8104	826	Dep.(mc); $10.10(fs), $3.92(mf) CFSTI
10038	720	Dep.(mc); $10.10(fs), $3.89(mf) CFSTI
10447	827	Dep.(mn); $6.00(cy), $1.25(mn) CFSTI
ORNL		
2873	876	Dep. mn; CFSTI $4.00 cy, $0.75 mn
3518	958	Dep.; $1.00(CFSTI)
3527	721	Dep. (mn); $6.00(cy), 5(mn) CFSTI
3530	320	Dep.; $9.10(fs), $3.32(mf) CFSTI
3531	321	Dep.; $8.10(fs), $2.84(mf) CFSTI
3578	481,877	Dep.; $2.75(CFSTI)
3585	608	Dep.; $1.00(CFSTI)
3591	425	Dep.; $3.50(CFSTI)
3612	743	Dep.; $1.50(CFSTI)
3627	106,209	Dep.; $7.00(cy), 5(mn)CFSTI
3629	744	Dep.; $0.50(CFSTI)
3640	745	Dep.(mn); $3.00(cy), 2(mn) CFSTI
3650	210	Dep.; $6.00(cy), 4(mn)CFSTI
3665	211	Dep.; $1.75(CFSTI)
3697	609,828,829, 894,895,920, 921,982,983	Dep.; $6.00(cy), 5(mn)CFSTI
3721(Suppl.1)	288	Dep. mn; CFSTI $3.00 cy, $0.75 mn
3744	516	Dep.; $3.00(CFSTI)
3765	959	Dep.; $1.00(cy), 1(mn)CFSTI
3776	408	Dep.; $5.00(cy), 3(mn)CFSTI
3782	517	Dep. mn; CFSTI $5.00 cy, $1.00 mn
3811	776	Dep. mn; CFSTI $2.00 cy, $0.50 mn
3820	212	Dep. (mn); $3.00(cy), 2(mn) CFSTI
3824	709	Dep.mn; CFSTI $4.00 cy, $0.75 mn
3830	213,214,808	Dep.mn; CFSTI $7.00 cy, $1.50 mn
3840	409	Dep.mn; CFSTI $1.00 cy, $0.50 mn
3849	215,322	Dep.mn; CFSTI $6.00 cy, $1.25 mn
3863	777	Dep. mn; CFSTI $2.00 cy, $0.50 mn
3868	216	Dep. mn; CFSTI $4.00 cy, $0.75 mn
3900	217	Dep.mn; CFSTI $6.00 cy, $1.25 mn
ORNL-NSIC		
3	482	Dep.; Free (ORNL)
ORNL-P		
131	610	Ind. Eng. Chem., Process Design Develop. 4, 333-7(1965)

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Report No.	Reference	Availability
331	922	Health Phys. 11, 567-77(1965)
447	518	Dep.(mn); $2.00(cy), 1(mn) CFSTI
605	830	Dep.(mn); $1.00(cy), 1(mn) CFSTI
737	923	Soil Sci. Soc. Am. Proc.29, 383-7(1965)
1064	984	Dep.(mn); $2.00(cy), 1(mn) CFSTI
1163	985	Nature 207, 158-9(1965)
1169	19	Ann. Rev. Nucl. Sci. 15, 151-74(1965)
1202	986	Nucl. Struct. Eng. 2, 438-46 (1965)
1411	973	Nucl. Sci. Eng. 24, 118-22 (1966)
1540	519	Microchem. J. 10, 301-14 (1966)
1568	987	Dep.mn; CFSTI $2.00 cy, $0.50 mn
1603	809	Dep.mn; CFSTI $1.00 cy, $0.50 mn
1926	878	Dep.mn; CFSTI $1.00 cy, $0.50 mn
1944	2	Dep.mn; CFSTI $1.00 cy, $0.50 mn
1987	988	Dep.mn; CFSTI $3.00 cy, $0.50 mn
2021	323	Dep.mn; CFSTI $2.00 cy, $0.50 mn
2058	989	Dep. mn; CFSTI $1.00 cy, $0.50 mn
ORNL-TM 515	218	Dep.(mc); $1.60(fs), $0.80 (mf)CFSTI
532	219	Dep.(mc); $3.60(fs), $1.22 (mf)CFSTI
603	220	Dep.(mc); $10.10(fs), $3.83 (mf)CFSTI
642	221	Dep.(mn); $3.00(cy), 2(mn) CFSTI
657	483	Dep.(mn); $1.00(cy), 1(mn) CFSTI
715	571	Dep.(mc); $13.00(fs), $5.66 (mf)CFSTI
734	990	Dep.(mc); $4.60(fs), $1.43 (mf)CFSTI
757	222	Dep.(mc); $13.50(fs), $5.96 (mf)CFSTI
785	223	Dep.(mc); $3.60(fs), $1.22 (mf)CFSTI
786	527	Dep.(mc); $1.10(fs), $0.80 (mf)CFSTI
798	224	Dep.(mc); $6.60(fs), $2.03 (mf)CFSTI
823	225	Dep.(mc); $6.60(fs), $2.18 (mf)CFSTI
827	778	Dep.(mc); $2.60(fs), $0.86 (mf)CFSTI
830	226	Dep.(mc); $10.10(fs), $4.07 (mf)CFSTI
836	227	Dep.(mc); $2.60(fs), $1.10(mf)
839	572	Dep.(mc); $1.10(fs), $0.80 (mf)CFSTI
851	484	Dep.(mc); $1.60(fs), $0.80 (mf)CFSTI
857	573	Dep.(mc); $3.60(fs), $1.34 (mf)CFSTI
882	746	Dep.(mn); $2.00(cy), 1(mn) CFSTI
886	228	Dep.(mn); $3.00(cy), 2(mn) CFSTI
889	960	Dep.(mc); $4.60(fs), $1.43 (mf)CFSTI
890	961	Dep.(mc); $3.60(fs), $1.31 (mf)CFSTI
936	229	Dep.(mn); $2.00(cy), 1(mn) CFSTI

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937	230	Dep.(mn); $2.00(cy), 1(mn) CFSTI
1027	231	Dep.(mn); $2.00(cy), 1(mn) CFSTI
1069	232	Dep.(mn); $2.00(cy), 1(mn) CFSTI
1081	233	Dep.(mn); $5.00(cy), 4(mn) CFSTI
1094	747	Dep.(mn); $2.00(cy), 1(mn) CFSTI
1108	924	Dep.(mn); $2.00(cy), 1(mn) CFSTI
1289	20	Dep.mn; CFSTI $3.00 cy, $0.50 mn
1327	991	Dep. mn; CFSTI $1.00 cy, $0.50 mn
1343	710	Dep. mn; CFSTI $1.00 cy, $0.50 mn
1384	361	Dep. mn; CFSTI $1.00 cy, $0.50 mn
1441	77	Dep. mn; CFSTI $1.00 cy, $0.50 mn
1442	528	Dep. mn; CFSTI $1.00 cy, $0.50 mn
ORNL-tr		
143	107	Dep.(mc); $2.60(fs), $0.86 (mf)JCL
916	779	Dep. mn. JCL $1.10 fs, $0.80 mf
965	831	Dep. mn. JCL $2.60 fs, $0.83 mf
OTS		
63-41112	316	$2.50(CFSTI) as JPRS-21816
64-21915	540	$0.50(CFSTI) as JPRS-23903
PAN		
453/XIX	611	Dep.
PG-Report		
500	832	Dep.; $0.60(BIS); 3s. (HMSO)
605	833	Dep.(mn); $2.00(BIS); 10s. (HMSO)
647	834	Dep.
PHS		
999-RH-13	835	Public Health Service Free
PNROO-DEV		
129	289	Dep.(mc); $1.10(fs), $0.80(mf) CFSTI
RAI		
116	722	Dep.(mc); $4.60(fs), $1.61(mf) CFSTI
350	723	Dep. mn; CFSTI $2.00 cy, $0.50 mn
RFP		
396	962	Dep.; $1.00(cy), 1(mn)CFSTI
442	963	Dep.; $1.00(cy), 1(mn)CFSTI
534	677	Dep. mn; CFSTI $2.00 cy, $0.50 mn
551	964	Dep. mn; CFSTI $2.00 cy, $0.50 mn
638	108	Dep. mn; CFSTI $1.00 cy, $0.50 mn
RHD		
Vol. IV, No. 3	290	$0.50(GPO)
RISO		
94	485	Dep.(mn)
RL-SA		
15	925	Dep.(mn); $1.00(cy), 1(mn) CFSTI
40	965	Dep. mn; CFSTI $1.00 cy, $0.50 mn

Report No.	Reference	Availability
41	765	Dep. mn; CFSTI $1.00 cy, $0.50 mn
50	428	Dep. mn; CFSTI $1.00 cy, $0.50 mn
53	966	Dep. mn; CFSTI $2.00 cy, $0.50 mn
58	992	Dep. mn; CFSTI $1.00 cy, $0.50 mn
59	3	Dep. mn; CFSTI $1.00 cy, $0.50 mn
60	993	Dep. mn; CFSTI $3.00 cy, $0.75 mn
62	879	Dep. mn; CFSTI $1.00 cy, $0.50 mn
RL-SEP		
65	766	Dep.(mn); $2.00(cy), 1(mn) CFSTI
496	78	Dep.(mn); $1.00(cy), 1(mn) CFSTI
RL-SEP		
518	967	Dep. mn; CFSTI $2.00 cy, $0.50 mn
RLO		
1725-41	836	Dep. mn; CFSTI $1.00 cy, $0.50 mn
RT/PROT(63)		
12	612	Rend. Acad. Sci. Fis. Mat., Soc. Nazl. Sci. Napoli(4), 29, 259-84(1962)
RT/PROT(65)		
12	168	Dep. mn
19	613	Dep. mn
29	614	Dep. mn
SAN		
413-2	896	Dep.; $2.00(cy), 1(mn)CFSTI
SB		
448(Rev.)	21	$0.10(CFSTI)
SC-T		
64-613	410	Dep.(mc); $1.10(fs), $0.80(mf) JCL
64-616	411	Dep.(mc); $1.10(fs), $0.80(mf) JCL
64-617	424	Dep.(mc); $1.10(fs), $0.80(mf) JCL
SGAE-PH		
31/1965	412	Dep. mn
SM		
71-24	103	Dep. mn; CFSTI $1.00 cy, $0.50 mn as LA-DC-7249
71-43	108	Dep. mn; CFSTI $1.00 cy, $0.50 mn as RFP-638
72-41	323	Dep. mn; CFSTI $2.00 cy, $0.50 mn as ORNL-P-2021
SRIA		
8	994	Dep.(mc); $9.60(fs), $3.80(mf) CFSTI
STI/DOC		
10/27	109	$3.00; Sch 63,-; 18s. stg; F. fr. 12,-; DM 10.50 IAEA
10/28	615	$1.00; Sch 21; 6s. std.; F. fr. 4; DM 3.20(IAEA)
STI/PUB		
87	257	$1.50; S 31.50; 9/-stg; F. fr. 6,-; DM 4.80(IAEA)
98	837,995	IAEA: $1.50; S 31,50; 9/-stg; F. fr. 6,-; DM 4,80
103	926	IAEA; $2.50; S52,50; 15/-stg; F.fr. 10,-; DM 8,-

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110	880	IAEA: $18.00; S 378,-; £ 5.8.0; F. fr. 72,-; DM 63,-.
SZS		
2	110	Dep.(mn)
2(1965)	375	Dep.(mn)
TID		
3910(Suppl.2)	22	Dep.; $2.50(CFSTI)
3910(Suppl.3)	23	Dep.; $2.50(cy), 2(mn)CFSTI
7677	352,370,377, 385,388,393, 413,414,415, 416, 1004	Dep.; $7.00(CFSTI)
7688	417	Dep.(mc); $9.60(fs), $3.80 (mf)CFSTI
7698(Pt.I)	4	Dep.; $6.00(cy), 5(mn)CFSTI
7699	574,575, 748	Dep.; $3.00(cy), 3(mn)CFSTI
8541	24	Dep.mn; CFSTI $4.50 cy, $2.00 mn
19562	324	Dep.; $1.25(CFSTI)
19608	838	Dep.(mc); $11.50(fs), $4.88 (mf)CFSTI
19618	325,927	Dep.(mc); $15.00(fs), $6.98 (mf)CFSTI
19941	111, 112, 113, 114,115	Dep.(mc); $8.60(fs), $3.14 (mf)CFSTI
20200	291	Dep.(mc); $3.60(fs), $1.25 (mf)CFSTI
20627	897	Nucl. Sci. Eng. 22, 350-65 (1965)
20662	616	$1.10(fs), $0.80(mf)CFSTI
20980	617	Dep.(mn); $3.00(cy), 2(mn) CFSTI
21279	292	Dep.(mn); $1.00(cy), 1(mn) CFSTI
21713	326	Dep.(mn); $4.00(cy), 2(mn) CFSTI
21786	116	Dep.(mn); $3.00(cy), 2(mn) CFSTI
21861	117	Dep.(mn); $3.00(cy), 2(mn) CFSTI
21882	749	Dep.(mn); $2.00(cy), 1(mn) CFSTI
21955	618	Dep.mn; CFSTI $1.00 cy, $0.50 mn
22168	118	Dep. mn; CFSTI $2.00 cy, $0.50 mn
22172	293	Dep. mn; CFSTI $1.00 cy, $0.50 mn
22427	119	Dep. mn; CFSTI $2.00 cy, $0.50 mn
22428	120	Dep. mn; CFSTI $4.00 cy, $0.75 mn
22511	619	Dep. mn; CFSTI $3.00 cy, $0.75 mn
22531	750	Dep. mn; CFSTI $2.00 cy, $0.50 mn
TRG-Report		
888	418	Dep.; BIS $0.60; HMSO 3s.
982	403	Dep. mn; CFSTI $12.75 cy, $5.00 mn as CONF-650407
TT		
65-30148	476	$5.00(CFSTI) as JPRS-28343
66-31365	100	CFSTI $1.00 cy, $0.50 mn as JPRS-34927
UCD		
472-112	620	Dep. mn; CFSTI $5.00 cy, $1.00 mn
UCRL		
13010	928	Dep.; $1.25(CFSTI)
14320	121	Dep. mn; CFSTI $1.00 cy, $0.50 mn

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UCRL-Trans		
1229(L)	486	Dep. mn; JCL $2.60 fs, $0.89 mf
UJV		
850/63	780	Dep. (mc)
936/63	621	Dep. (mc)
973/63	622	Dep. (mn)
1103/64	376	Dep. (mc)
1104/64	529	Dep. (mn)
1105/64	711	Dep. (mn)
1106/64	623	Dep. (mn)
1107/64	929	Dep. (mn)
1108/64	624	Dep. (mn)
1109/64	678	Dep. (mc)
1110/64	679	Dep. (mn)
1111/64	712	Dep. (mn)
1112/64	625	Dep. (mn)
1113/64	781	Dep. (mc)
1296/65	626	Dep. (mn)
1489	782	Dep. (mn)
1493	79	Dep. (mn)
UWFL		
86	327	Dep.; $3.00(CFSTI)
WASH		
1052	122	Dep.; $0.50(GPO)
ZFK-RCH		
2	627	Dep. (mc)
ZFK-RN		
17	628	Dep. (mn)

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RADIOACTIVE WASTE PROCESSING AND DISPOSAL

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INTRODUCTION

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COBALT ISOTOPES Co-60 - content of liquid waste disposal area at Chalk River Nuclear Laboratories
COBALT ISOTOPES Co-60 - uptake by edible plants in environs of Chalk River Nuclear Laboratories

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OSTEICHTHYES - metabolism in, effects of chlorinated hydrocarbons on calcium and strontium
OTTAWA RIVER - monitoring in environs of nuclear power plant during 1971 for cesium-137 and strontium-90
PERCH LAKE (ONTARIO) - radioecology of
PLANKTON - primary productivity of, carbon-14 tracer study
STRONTIUM ISOTOPES Sr-85 - metabolism in fish, effects of age and temperature on
STRONTIUM ISOTOPES Sr-90 - monitoring in Ottawa River in environs of nuclear power plant during 1971
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ENVIRONMENT - radioecology of marine, in Humboldt Bay, California
HUMBOLDT BAY POWER REACTOR - waste disposal practices at, effects on radioecology of marine environment of
IODINE ISOTOPES I-131; MANGANESE ISOTOPES Mn-54; NIOBIUM ISOTOPES Nb-95 - uptake by organisms and sediments in Humboldt Bay marine environment
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RUTHENIUM ISOTOPES Ru-106 - uptake by organisms and sediments in Humboldt Bay marine environment
SILTS - radioisotope uptake by marine, from waste released by Humboldt Bay power reactor
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PLUTONIUM - separation from aqueous Purex waste by solvent extraction using dibutyl butylphosphonate
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RADIOACTIVE WASTE - handling and storage of gaseous, from nuclear power industry, status of methods for
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SEAWATER - contamination of Indian, by low-level radioactive liquid waste disposal into coastal waters, (E)
SEAWATER; SILTS - radioisotope contents of Indian coastal, measurement of cesium-137 and strontium-90
STRONTIUM ISOTOPES Sr-90 - content in coastal seawater and sediments near Bombay, India

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MAN - radiation hazards to, from nuclear power plant radioactive wastes buried at Federal Repository, evaluation of long-term α
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ENVIRONMENT - radiation monitoring in Michigan, during 1970
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MICHIGAN - radiation monitoring of environment in, during 1970
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RADIOACTIVE WASTE - discharge from Big Rock Point Power Reactor during 1970
RADIOISOTOPES - content in atmosphere, atmospheric precipitation, bottom sands, fish, and waters near Big Rock Point Reactor during 1970
SANDS - radioisotope content of, near Big Rock Point Reactor during 1970
SILTS - radioisotope content of, near Palisades Power Reactor during 1970
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SEAWATER - radiation monitoring of coastal, in UK following waste disposal, 1970
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FOOD CHAINS - radioisotope content of human, in relation to levels of discharge to environment, estimation of average internal radiation dose to man from cesium-137 and strontium-90
MAN - radiation dose to, from cesium-137 and strontium-90 in human food chains, effects of age and diet on average internal
RADIATION - dose to man from cesium-137 and strontium-90 in human food chains, effects of age and diet on average internal
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CHARCOAL - sorptive properties of filters of, for iodine
FOOD - cesium-137 content of Belgian, during 1970
FOOD - strontium-90 content of Belgian, during 1970
NUCLEAR MATERIALS MANAGEMENT - safeguards for, research programs on
RADIOACTIVE WASTE - disposal of, research programs on
SAMPLING - method for smear-type, for radioactive materials on surfaces
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NEUTRON SOURCES - development of antimony--beryllium, for nuclear materials management

uses
NUCLEAR MATERIALS MANAGEMENT - methods for fuel reprocessing plants
PLUTONIUM - determination in radioactive waste by delayed neutron method
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RADIOACTIVE WASTE - separation of cesium from, by ion exchange, thermal decomposition of eluate in

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PLUTONIUM - separation from Purex acid sludge by solvent extraction using dibutyl butylphosphonate
PUREX PROCESS - radioactive waste from, recovery of americium, europium, neptunium, and plutonium from
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352 26:47912 ESTER PROGRAM: HIGH-LEVEL RADIOACTIVE WASTE SOLIDIFICATION. BOROSILICATE GLASS EXPERIMENTAL EVALUATION AND BENCH-SCALE VITRIFICATION WITH SIMULATED FISSION PRODUCT SOLUTIONS. Donato, A.; Bocola, W. (CNEN, Rome). Energ. Nucl. (Milan); 19: No. 7, 459-69(Jul 1972).

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RADIOACTIVE WASTE - solidification of high-level, Ester program borosilicate glass process for

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ARGON ISOTOPES Ar-41 - content in atmosphere in vicinity of Los Alamos Scientific Laboratory, July to Dec. 1971
ATMOSPHERE - radioactivity monitoring in vicinity of Los Alamos Scientific Laboratory, July to Dec. 1971
BETA PARTICLES - content in atmosphere, effluent waste water, ground and surface waters, silts, soils, and water supplies in vicinity of Los Alamos Scientific Laboratory, July to Dec. 1971
CESIUM ISOTOPES Cs-137 - content in effluent waste water, ground and surface waters, silts, soils, and water supplies in vicinity of Los Alamos Scientific Laboratory, July to Dec. 1971
CESIUM ISOTOPES Cs-138 - content in atmosphere in vicinity of Los Alamos Scientific Laboratory, July to Dec. 1971
GROUND WATERS - radioactivity monitoring in vicinity of Los Alamos Scientific Laboratory, July to Dec. 1971
IODINE ISOTOPES I-131 - content in atmosphere in vicinity of Los Alamos Scientific Laboratory, July to Dec. 1971
LOS ALAMOS AREA (N. MEX.) - radioactivity monitoring in vicinity of, July to Dec. 1971
PLUTONIUM ISOTOPES Pu-238; PLUTONIUM ISOTOPES Pu-239 - content in atmosphere, effluent waste water, ground and surface waters, silts, soils, and water supplies in vicinity of Los Alamos Scientific Laboratory, July to Dec. 1971
RADIUM ISOTOPES Ra-226 - content in effluent discharge areas in vicinity of Los Alamos Scientific Laboratory, July to Dec. 1971
RUBIDIUM ISOTOPES Rb-88 - content in atmosphere in vicinity of Los Alamos Scientific Laboratory, July to Dec. 1971
SEWAGE - radioactivity monitoring of aqueous effluents from, in vicinity of Los Alamos Scientific Laboratory, July to Dec. 1971
SILTS; SOILS - radioactivity monitoring in vicinity of Los Alamos Scientific Laboratory, July to Dec. 1971
STRONTIUM ISOTOPES Sr-90 - content in effluent discharge areas in vicinity of Los Alamos Scientific Laboratory, July to Dec. 1971
SURFACE WATERS - radioactivity monitoring in vicinity of Los Alamos Scientific Laboratory, July to Dec. 1971
TRITIUM - content in atmosphere, effluent waste water, ground and surface waters, soils, and water supplies in vicinity of Los Alamos Scientific Laboratory, July to Dec. 1971
URANIUM - content in atmosphere, effluent waste water, ground and surface waters, silts, soils, and water supplies in vicinity of Los Alamos Scientific Laboratory, July to Dec. 1971
URANIUM ISOTOPES U-234 - content in effluent discharge areas in vicinity of Los Alamos Scientific Laboratory, July to Dec. 1971
URANIUM ISOTOPES U-235 - content in atmosphere in vicinity of Los Alamos Scientific Laboratory, July to Dec. 1971
WATER SUPPLIES - radioactivity monitoring in vicinity of Los Alamos Scientific Laboratory, July to Dec. 1971

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HUMBOLDT BAY POWER REACTOR - waste discharge from, radioisotope transport in marine environment following liquid
IODINE ISOTOPES I-131; IRON ISOTOPES Fe-59; MANGANESE ISOTOPES Mn-54; NIOBIUM ISOTOPES Nb-95 - transport in Humboldt Bay marine environment following discharge of liquid reactor wastes
PLANKTON - radioisotope content in phyto-, in Humboldt Bay area following liquid reactor waste discharge
RADIOISOTOPES - transport in Humboldt Bay marine environment following liquid reactor waste discharge
RUTHENIUM ISOTOPES Ru-106 - transport in Humboldt Bay marine environment following discharge of liquid reactor wastes
SEAWATER - radioisotope transport in Humboldt Bay, following liquid reactor waste discharge
SILTS - radioisotope content in Humboldt Bay area, following liquid reactor waste discharge
SILVER ISOTOPES Ag-110; TRITIUM; ZINC ISOTOPES Zn-65; ZIRCONIUM ISOTOPES Zr-95 - transport in Humboldt Bay marine environment following discharge of liquid reactor wastes

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COLUMBIA RIVER - aquatic organisms from, trace element content of
COPPER - determination in fingernails by x-ray fluorescence methods
GAMMA SOURCES - determination of, coincidence-anticoincidence semiconductor spectrometers for
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RUTHENIUM - sorption on crystalline cement hydrates in cement milk
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CADMIUM NITRATES - transference numbers in highly concentrated aqueous melts of, pp. 61-62
CALCIUM SULFATES - solubility in aqueous sodium perchlorate solutions at 25 to 250°C, pp. 57-58
CALCIUM SULFATES - solubility in sodium perchlorate--sodium sulfate--water system from 0 to 350°C, pp. 58-59
CALCIUM SULFATES - solubility in brackish well waters, effects of polyphosphate additives on, pp. 59-60
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RADIATION MONITORING - equipment for alpha and beta, in air exhaust ducts and stacks in

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RADIOACTIVE WASTE - analysis of aqueous solutions of, from fuel reprocessing for americium, curium, and plutonium
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RUTHENIUM; STRONTIUM - separation from low-level liquid waste at West Valley Processing Plant
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PLUTONIUM - separation from water by flocculation, evaluation of coagulants for
RADIOACTIVE WASTE - processing of low-level, evaluation of coagulations for flocculation in
SODIUM HYDROXIDES - $Fe_2(SO_4)_3$--NaOH, properties as coagulant for separation of plutonium and uranium from water by flocculation
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PUREX PROCESS - plutonium reduction in
RADIOACTIVE WASTE - processing of salt, from pyrochemical head-end process in fuel reprocessing for recycle
RARE GASES - production and release in power reactors
REACTOR FUEL ELEMENTS - reprocessing of, pyrochemical processes in head-end
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STAINLESS STEEL - solubility in liquid zinc
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THORIUM OXIDES ThO_2 - ThO_2--UO_2, analysis for thorium and uranium, x-ray fluorescence method for in-line
TRITIUM - production and release in power reactors
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PERCH LAKE (ONTARIO) - evaporation studies on
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RADIOACTIVE WASTE - monitoring at Chalk River for measuring radioactivity discharged to Ottawa River
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STRONTIUM ISOTOPES Sr-90 - content of precipitation sample and surface streams in Chalk River area
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RADIOACTIVE WASTE PROCESSING, AND DISPOSAL

Prepared for
Division of Waste Management and Transportation

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INTRODUCTION

References to 345 publications on all aspects of waste processing and disposal that have appeared in *NSA* Volume 27 (1973) are contained in this bibliography.

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AIR - analysis for tritium as gas and tritiated water by liquid scintillation counting
NATURAL WATERS - plutonium chemical states in
OXYGEN - determination in steel by neutron activation
PLUTONIUM - chemical states of, multiple point problems in studies on
PLUTONIUM - chemical states in environmental waters
PROTACTINIUM - determination in St. Louis residues by γ counting
PROTACTINIUM ISOTOPES Pa-231 - production from St. Louis residues by Cotter process
RADIOACTIVE WASTE - thorium-230 recovery from St. Louis residues, by Cotter process
RADIOACTIVE WASTE - protactinium-231 recovery from St. Louis residues, by Cotter process
SOLUTIONS - analysis of, spectrographic
STEEL - analysis for oxygen by neutron activation
THORIUM ISOTOPES Th-230 - production from St. Louis residues by Cotter process
THORIUM ISOTOPES Th-230 - determination in St. Louis residues by α or γ counting
TRITIUM - determination in air as gas and tritiated water by liquid scintillation counting

2 27:139 APPLICATION OF THREE-PHASE SYSTEMS IN CHEMICAL TECHNOLOGY. Csovari, M. (Mecsek Ore Mining Enterprise, Pecs, Hungary); Czegledi, B.; Kuzin, I. A. pp 145-52 of Proceedings of the 2nd Conference on Applied Physical Chemistry. Vol. 1. /Buzas, I. (ed.). Budapest; Akademiai Kiado (1971). From 2. conference on applied physical chemistry; Veszprem, Hungary (2 Aug 1971). See CONF-710844--(Vol.1).

IRON - separation of scandium from waste solutions of uranium and, by solvent extraction using anion exchange resin--DEHPA--sulfuric acid
SCANDIUM - separation from iron and uranium in waste solutions by solvent extraction using anion exchange resin--DEHPA--sulfuric acid
URANIUM - separation of scandium from waste solutions of iron and, by solvent extraction using anion exchange resin--DEHPA--sulfuric acid

3 27:183 (HMI-B--124) NUCLEAR CHEMISTRY DIVISION AND REACTOR DIVISION ANNUAL RESEARCH REPORT, 1971. (Hahn-Meitner-Institut fuer Kernforschung, Berlin (West Germany)). Jul 1972. 42p. Dep. NTIS (U. S. Sales Only).

ECOSYSTEMS - radioisotope distribution and studies in
NEUTRONS - scattering of, research progress on
NUCLEAR CHEMISTRY - research progress on, 1971
RADIOACTIVE WASTE - solidification of, waste processes for
RADIOISOTOPES - distribution and studies in ecosystems

4 27:203 HIGH-VOLTAGE ELECTROPHORESIS OF ^{106}Ru: SEA-WATER SYSTEMS. Ishikawa, M.; Pucar, Z. (Inst. Rudjer Boskovic, Zagreb). J. Radioanal. Chem.; 11: No. 2, 197-207(Aug 1972).

RUTHENIUM; RUTHENIUM CHLORIDES - complexes, chemical states in seawater, effects of aging on
RUTHENIUM ISOTOPES Ru-106 - disposal in sea, physicochemical behavior following
RUTHENIUM NITROSYL NITRATES - complexes, chemical states in seawater, effects of aging on
SEAS - ruthenium-106 disposal in, physicochemical behavior following
SEAWATER - ruthenium complexes in, effects of aging on chemical states in

5 27:214 (DUN-SA--192) TREATMENT OF CHEMICAL WASTES FROM N FUELS PLANT. Tew, H. F. (Douglas United Nuclear, Inc., Richland, Wash.). 6 Sep 1972. Contract AT(45-1)-1857. 8p. (CONF-721030--1). Dep. NTIS. From Conference on pollution control and abatement at AEC facilities; Oak Ridge, TN. (25 Oct 1972).

HANFORD PRODUCTION REACTORS (N) - fuels for, processing and disposal of liquid wastes from fabrication of
RADIOACTIVE WASTE - processing of liquid, from fabrication of Zircaloy-2-clad uranium
RADIOACTIVE WASTE - disposal of liquid, from fuel fabrication by evaporation using solar energy
URANIUM - fabrication of fuels of, processing and disposal of liquid wastes from

6 27:215 FLUIDIZED BED CALCINATION OF RADIOACTIVE WASTES USING IN-BED COMBUSTION HEATING. Thompson, T. K.; Lohse, G. E.; Wheeler, B. R. (Allied Chemical Corp., Idaho Falls, Idaho). Nucl. Technol.; 16: No. 3, 396-405(Nov 1972).

RADIOACTIVE WASTE - calcination at ICPP, in-bed combustion of hydrocarbon fuels for fluidized-bed
RADIOCHEMICAL PROCESSING PLANTS - radioactive waste processing at ICPP, in-bed combustion for fluidized-bed calcination for

7 27:216 INSTALLATION FOR CONTINUOUSLY INCINERATING RADIOACTIVE WASTES WITHOUT THE PRODUCTION OF LIQUID EFFLUENTS. Auchapt, Pierre; Bouzou, Georges (to Commissariat a l'Energie Atomique). French Patent 2,100,528. 24 Mar 1972. Filed 18 Feb 1970. (In French).

INCINERATORS - development for continuous processing of α-emitting wastes
RADIOACTIVE WASTE - processing of α-emitting, installation for continuous incineration for

8 27:297 (ORNL-TM--3613) COMPUTER CODE FOR ANALYZING ROUTINE ATMOSPHERIC RELEASES OF SHORT-LIVED RADIOACTIVE NUCLIDES. Reeves, M.; Fowler, P. G.; Cowser, K. E. (Oak Ridge National Lab., Tenn.). Oct 1972. 29p. Dep. NTIS.

COMPUTER PROGRAMS - for analysis of routine atmospheric releases of short-lived radioisotopes
RADIOISOTOPES - release of short-lived, from stacks, computer code for analysis of routine
STACK DISPOSAL - radioisotope release from, computer code for analysis of short-lived

9 27:314 ENVIRONMENTAL MONITORING AND DISPOSAL OF RADIOACTIVE WASTES FROM U. S. NAVAL NUCLEAR-POWERED SHIPS AND THEIR SUPPORT FACILITIES, 1971. Miles, M. E.; Sjoblom, G. L.; Burke, R. D. Radiat. Data Rep.; 13: No. 9, 469-78(Sep 1972).

ENVIRONMENT - contamination by disposal of radioactive wastes from US Navy nuclear ships, evaluation of
HARBORS - radioactivity by monitoring of, from radioactive waste disposal from US Navy nuclear ships
NAVAL VESSELS - radioactive waste disposal from US nuclear, evaluation of environmental contamination from
RADIOACTIVE WASTE - disposal from US Navy nuclear ships, evaluation of environmental contamination from

10 27:892 (ARH--2548) ENGINEERING STUDY: ALTERNATIVE EVAPORATIVE FACILITIES FOR THE B TANK FARM COMPLEX. McKnight, J. P. (Atlantic Richfield Hanford Co., Richland, Wash.). 22 Jun 1972. Contract AT(45-1)-2130. 21p. Dep. NTIS.

EVAPORATORS - performance for solidifying Hanford B tank farm wastes, engineering studies of
RADIOACTIVE WASTE - solidification of Hanford B tank farm, engineering studies of evaporative facilities for

11 27:893 (DEMO--72/11) STUDY OF LEACHING KINETICS OF IONS FROM ''WASTES'' INCORPORATED IN BITUMEN. Amarantos, S. G.; Petropoulos, J. H. (Democritus Nuclear Research Center, Athens (Greece)). Jul 1972. 31p. Dep. NTIS (U. S. Sales Only).

BITUMINOUS MATERIALS - effectiveness for incorporation of radioactive wastes for disposal, leaching studies on
CALCIUM - leaching by water from simulated radioactive wastes incorporated in bitumen, kinetics of
RADIOACTIVE WASTE - calcium and sodium leaching by water from simulated, incorporated in bitumen, kinetics of
SODIUM - leaching by water from simulated radioactive wastes incorporated in bitumen, kinetics of

12 27:894 (ORNL--4827) HYDRAULIC FRACTURING TEST AT WEST VALLEY, NEW YORK. de Laguna, Wallace (Oak Ridge National Lab., Tenn.). Oct 1972. Contract W-7405-eng-26. 69p. Dep. NTIS.

NEW YORK - shale fracturing tests at West Valley, mode for hydraulic
RADIOACTIVE WASTE - disposal at West Valley, N. Y., hydraulic fracturing tests for
SHALES - fracture testing of West Valley, N. Y., mode for hydraulic

13 27:895 (UCRL--73818) PLACEMENT AND SELF-PROCESSING OF NUCLEAR FUEL REPROCESSING WASTES IN DEEP UNDERGROUND CAVITIES. Lewis, A. E.; Cohen, J. J.; Braun, R. L.; Rothman, A. J. (California Univ., Livermore. Lawrence Livermore Lab.). Aug 1972. 21p. (CONF-721107--1). Dep. NTIS. From Symposium on the management of radioactive wastes from fuel reprocessing; Paris, France (27 Nov 1972).

RADIOACTIVE WASTE - disposal in deep underground rock cavities, self-boiling and solidification method for

14 27:896 TREATMENT AND DISPOSAL OF LOW-LEVEL RADIOACTIVE SLUDGES BY SOLAR EVAPORATION. Lee, Sang Hoon (Atomic Energy Research Inst., Seoul). J. Korean Nucl. Soc.; 4: No. 3, 194-202(Sep 1972).

EVAPORATORS - design and economics of solar energy-heated, for low-level radioactive waste sludge concentration
FURNACES - economics of solar, for evaporation of low-level radioactive waste sludge prior to disposal
RADIOACTIVE WASTE - evaporation of low-level sludge, for volume reduction prior to disposal, economics of solar energy for

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PLUTONIUM - separation from spent oxide fuels by pyrochemical and solvent extraction processes
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PLUTONIUM - determination in fast breeder oxide fuels by x-ray fluorescence methods, on-line
PLUTONIUM - reduction and reoxidation in Purex Process
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PLUTONIUM NITRATES - $Pu(NO_3)_4$--$UO_2(NO_3)_2$, denitration of solutions of, production of oxides by fluidized-bed
PLUTONIUM OXIDES PuO_2 - PuO_2--UO_3, preparation by fluidized-bed denitration of plutonium nitrate--uranyl nitrate solutions
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REACTOR FUEL ELEMENTS - reprocessing of stainless steel-clad oxide, head-end pyrochemical
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STAINLESS STEEL - solubility in liquid zinc
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URANYL NITRATES - $Pu(NO_3)_4$--$UO_2(NO_3)_2$, denitration of solutions of, production of oxides by fluidized-bed
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RADIOISOTOPES - fate and transport in rivers
RIVERS - radioactive waste disposal into, sources and fate of
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PIPES - encasement for buried metal, for radioactive wastes at NRTS, design of
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COOPER POWER REACTOR - radioactive waste disposal system for, revised and requested data on
COOPER POWER REACTOR - electrical systems for, revised and requested data on
COOPER POWER REACTOR - instrumentation for, revised and requested data on
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SHEARON HARRIS POWER REACTOR, UNIT 1 - power conversion system for, requested data on
SHEARON HARRIS POWER REACTOR, UNIT 1 - auxiliary systems for, requested data on
SHEARON HARRIS POWER REACTOR, UNIT 1 - engineered safeguards for, requested data on
SHEARON HARRIS POWER REACTOR, UNIT 2 - site for, requested data on
SHEARON HARRIS POWER REACTOR, UNIT 2 - safety analysis of, requested data on
SHEARON HARRIS POWER REACTOR, UNIT 2 - quality assurance program for, requested data on
SHEARON HARRIS POWER REACTOR, UNIT 2 - radioactive waste disposal system for, requested data on
SHEARON HARRIS POWER REACTOR, UNIT 2 - power conversion system for, requested data on
SHEARON HARRIS POWER REACTOR, UNIT 2 - auxiliary systems for, requested data on
SHEARON HARRIS POWER REACTOR, UNIT 2 - engineered safeguards for, requested data on
SHEARON HARRIS POWER REACTOR, UNIT 3 - site for, requested data on
SHEARON HARRIS POWER REACTOR, UNIT 3 - safety analysis of, requested data on
SHEARON HARRIS POWER REACTOR, UNIT 3 - quality assurance program for, requested data on
SHEARON HARRIS POWER REACTOR, UNIT 3 - radioactive waste disposal system for, requested data on
SHEARON HARRIS POWER REACTOR, UNIT 3 - power conversion system for, requested data on
SHEARON HARRIS POWER REACTOR, UNIT 3 - auxiliary systems for, requested data on
SHEARON HARRIS POWER REACTOR, UNIT 3 - engineered safeguards for, requested data on
SHEARON HARRIS POWER REACTOR, UNIT 4 - site for, requested data on
SHEARON HARRIS POWER REACTOR, UNIT 4 - safety analysis of, requested data on
SHEARON HARRIS POWER REACTOR, UNIT 4 - quality assurance program for, requested data on
SHEARON HARRIS POWER REACTOR, UNIT 4 - radioactive waste disposal system for, requested data on
SHEARON HARRIS POWER REACTOR, UNIT 4 - power conversion system for, requested data on
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FAST TEST REACTOR - operations aspects of, preliminary safety analysis of
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SOILS - radioisotope distribution in NRTS, from waste disposal, measurement of cerium-144, cesium-137, cobalt-60, and strontium-90
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REACTORS, PRESSURIZED-WATER - radioactive waste storage tank for power, standard for radiological consequences of failure of gaseous
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DUANE ARNOLD POWER REACTOR - radioactive waste disposal system for, revised and requested data on
DUANE ARNOLD POWER REACTOR - auxiliary systems for, revised and requested data on
DUANE ARNOLD POWER REACTOR - safety analysis of, revised and requested data on
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RADIOACTIVE WASTE - processing of high-level, by solidification and glazing
REACTOR FUEL ELEMENTS - reprocessing of thorium-containing graphite, head-end combustion processes in
REACTOR FUELS - reprocessing of thorium-containing, Chlorinex, KFA--TBP 23/25, and Thorex processes for
REACTOR FUELS - reprocessing of mixed oxide, molten salt dissolution in
REACTOR FUELS - reprocessing of mixed oxide, chlorination process for
THOREX PROCESS - flowsheet development for
THORIUM - separation from spent fuels, evaluation of processes for
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RADIOCHEMICAL PROCESSING PLANTS - development of Jupiter, for processing thorium-containing fuels
REACTOR FUEL ELEMENTS - reprocessing of thorium-containing graphite, combustion processes for head-end
REACTOR FUELS - reprocessing of thorium-containing, Thorex, KFA--TBP 23/25, and Chlorinex flowsheets for
REACTOR FUELS - reprocessing of mixed oxide, chlorination process for
REACTOR FUELS - burnup analysis of thorium-containing
REACTOR FUELS - burnup analysis of graphite, automatic device for
THOREX PROCESS - flowsheet development for
THOREX PROCESS - thorium determination in
THORIUM - determination in Thorex fuel solutions
URANIUM - determination in aqueous fission product solutions, in-line
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DONALD C. COOK POWER REACTOR, UNIT 1 - safety analysis of, revised and requested data on
DONALD C. COOK POWER REACTOR, UNIT 2 - engineered safeguards for, revised and requested data on
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