

RADIOACTIVE WASTE (M-3679, 23rd Ed.)

RADIOACTIVE WASTE PROCESSING AND DISPOSAL (EXCERPT)

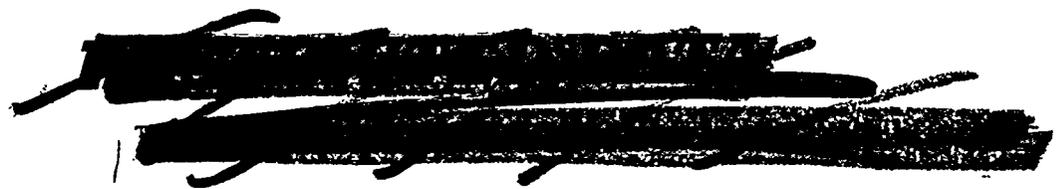
A Bibliography of Selected Classified Report Literature

Compiled by Theodore F. Davis

October 1959

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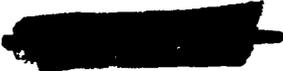
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AEC Technical Information Service Extension, Oak Ridge, Tennessee



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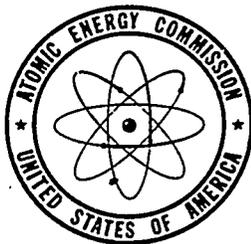
RADIOACTIVE WASTE PROCESSING AND DISPOSAL.

A Bibliography of Selected Classified Report
Literature

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Theodore F. Davis

~~Hugh B. Vorens~~ *J*



October 1959

Technical Information Service, Oak Ridge, Tennessee



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Conference and Visit Reports

Division of Engineering, AEC

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REPORT FOR THE FOURTH CONFERENCE HELD AT OAK RIDGE BY THE COMMITTEE ON WASTE DISPOSAL—AUGUST 23–25, 1948. Sept. 11, 1948. Changed from SECRET Jan. 28, 1958. 25p. (TID-100). OFFICIAL USE ONLY

The nature of waste disposal problems confronting Oak Ridge plants, and the scope of research and development work being conducted toward the solution of the problems are discussed. The present radiation levels released to the stream are included.

Oak Ridge National Laboratory Oak Ridge, Tennessee

37

PROPOSED SOLIDIFICATION CELL FOR HIGH-LEVEL RADIOACTIVE AQUEOUS WASTES. R. A. Elmgren, N. J. G. Bohlin, M. E. Walters, and J. H. Schrafft. p.273-282 of HOT LABORATORIES AND EQUIPMENT; SECOND INFORMATION MEETING HELD AT ORNL OCTOBER 7, 8, AND 9, 1952. July 31, 1953. 348p. (CF-52-10-230). SECRET

Details of the design, construction and operation of the cell are presented. A shielded container for storage or shipping of liquid wastes is described on p.297-303.

38

TRIP REPORT: VISIT TO ALCO PRODUCTS (SCHENECTADY), KAPL, COMBUSTION ENGINEERING (WINDSOR) AND APPR. J. T. Roberts. July 31, 1958. 3p. (CF-58-7-109). CONFIDENTIAL

Methods used by the Army and Navy in disposal of waste from power reactors are discussed.

REFERENCES

Stack Gas Problem Working Group, AEC

42

COMPILATION OF PAPERS PRESENTED AT THE MEETING OF THE STACK-GAS PROBLEM WORKING GROUP, JUNE 21 AND 22, 1949. [nd]. 86p. (WASH-7). SECRET

The following papers were presented: Particle Detection in Oak Ridge National Laboratory Pile Exhaust Air, C. D. Cagle; Plans for Attacking the Residual Atmospheric Contamination Problem at Oak Ridge National Laboratory. F. L. Culler;

Particulate Contamination of the Atmosphere by ORNL Operations from June 1948 to June 1949. Forrest Western.

Waste Processing Committee, AEC

43

MEETING OF AEC WASTE PROCESSING COMMITTEE AT LOS ALAMOS, N. M. October 1950. 82p. (TID-460). OFFICIAL USE ONLY

A number of problems related to radioactive waste disposal have been considered. Topics discussed include:

control of radioactive air contamination at ORNL.

Oak Ridge National Laboratory Oak Ridge, Tennessee

129

REPORT OF TECHNICAL DIVISION SECTION I FOR MONTH ENDING OCTOBER 20, 1948. F. L. Steahly. Oct. 20, 1948. Changed from SECRET Dec. 29, 1955. 30p. (CF-48-11-224) CONFIDENTIAL

Progress is reported on Redox studies; second U recovery cycle in laboratory and semiworks runs; Pu oxidation and distribution; waste treatment by ion exchange; decontamination of 25 and plant radiochemical wastes; Th recovery by solvent extraction; theoretical extraction studies; equipment testing and development; bibliography of chemical process development; section 23 and 25 reports; Redox second-cycle U flowsheet; flowsheet for Al decontamination from 25 wastes by ion exchange.

130

DISSOLVER OFF-GAS TREATMENT-ARCO. Quarterly Report for May through July, 1951. E. P. Reichardt. Sept. 27, 1951. Changed from SECRET Dec. 15, 1955. 8p. (CF-51-9-148) CONFIDENTIAL

In order to remove Kr and Xe from the off-gases from dissolvers, other gases which interfere with their adsorption must be removed. A description of the Puregas

Catalytic Unit for the removal of O_2 and N_2O from the off-gas mixture is given. The unit has satisfactorily reduced the O_2 and N_2O from the synthetic gas mixtures to trace quantities. In determining the total throughput of the bed, both N_2 and super-heated steam worked satisfactorily as a diluent. It was found, however, that the efficiency of reduction of the O_2 and N_2O was low until the bed reached an operating temperature of $\sim 325^\circ C$. It was also found experimentally that explosions were encountered when the H_2 in the off-gas equalled or exceeded 20% by volume.

131

PROGRESS REPORT ON THE PILOT PLANT DEVELOPMENT OF THE RADIOACTIVE GAS SEPARATIONS PROCESS. W. E. Watkins. Aug. 20, 1952. 26p. (CF-52-8-137) SECRET

A system for the recovery of radioactive I_2 , Xe, and Kr from dissolver off-gas has been installed in the ORNL Pilot Plant and successfully operated. It has been demonstrated that the gas from full Hanford-level dissolutions can be processed to yield HNO_3 acceptable for reuse and that radioactive I_2 , Xe and Kr can be satisfactorily removed from the off-gas leaving a mixture of N_2 and O_2 acceptable for release to the atmosphere. Total recovery of NO and NO_2 from the dissolver off-gas was shown. Of the total NO and NO_2 evolved during dissolution the acid tower recovered about 90%, whereas the remaining 10% was adsorbed in the gas holder. The results of preliminary experiments show that an over-all volume-reduction factor for Xe and Kr of 400 can be obtained, but factors as high as 2000 appear available by improvements in the adsorption-desorption cycle. The volumetric requirements of the gas holder have been clearly defined. Results from 14 dissolvings have shown that the volume of the holder need only be about 10% of the total volume of off-gas evolved during dissolution. The residual gas that remained in the holder after all extractable components (such as No and NO_2) had been adsorbed represented only 4 to 5% of the total dissolver off-gas. The use of the thermal decomposition unit or pure gas unit for N_2O removal from dissolver off-gas has been abandoned since no significant quantity of N_2O was found in any of the off-gas from 30 to 40 Purax dissolvings.

132

WASTE STORAGE STUDIES. p. 22-29 of PROGRESS REPORT ON 25 PROCESS ASSISTANCE FOR PERIOD ENDING OCTOBER 29, 1954. F. R. Bruce and E. L. Nicholson. Nov. 15, 1954. Changed from SECRET May 27, 1957. 36p. (CF-54-11-67) CONFIDENTIAL

The volumes and compositions of waste from the caustic dissolution and acid dissolution flowsheets have been compared from the standpoints of maximum volume reduction obtainable and of storage problems likely to be encountered. Data based on 25 Kg of U^{235} processed per day for both acid and caustic dissolutions give volume of waste produced, volume after neutralization and volume after concentration. Freezing points of the two wastes vs. concentration are also given.

133

AN EVALUATION OF MATERIALS FOR LINING EARTHEN BASINS FOR CONTAINING RADIOACTIVE SOLUTIONS. G. A. West, F. L. Rodgers. p. 27-8 of UNIT OPERATIONS STATUS REPORT FOR MARCH 1955. (Unit Operations Sect., Chemical Technology Div.). W. K. Elster, ed. 60p. (CF-55-3-190) CONFIDENTIAL

Permeometer tests to determine the leakage of simulated waste through asphalt, soil, ball clay, and shale are summarized. The tests indicate that the leakage through a 1/2" asphalt layer is negligible. Laboratory tests of 1/16" asphalt and permeometer tests of 1/2" thick asphalt show that tap water at $77^\circ C$ has a progressive deteriorating effect. Water penetrated the 1/16" asphalt in 30 days and the top layer of the 1/2" asphalt deteriorated. This suggests that leakage may occur after long time exposures.

REFERENCES

134

AN EVALUATION OF MATERIALS FOR LINING EARTHEN BASINS FOR CONTAINING RADIOACTIVE WASTE. G. A. West. p. 66-7 of UNIT OPERATIONS SECTION, CHEMICAL TECHNOLOGY DIVISION MONTHLY PROGRESS REPORT FOR JUNE 1955. W. K. Elster. 94p. (CF-55-6-180) CONFIDENTIAL

Results of tests on four tars, Barrett Roofing Pitch, pipeline enamel, millwrap enamel, and waterworks enamel show that they do not have the resistance of asphalts against a simulated waste (1.6M Al(NO₃)₃, 8.2 N acid) and against tap water at room temperatures.

135

RADIOCHEMICAL WASTE DISPOSAL. p. 78-86 OF TECHNICAL DIVISION REPORT FOR QUARTER ENDING SEPTEMBER 1, 1948. Stuart McLain. Sept. 1, 1948. Changed from SECRET Jan. 24, 1956. 118p. (ORNL-140) CONFIDENTIAL.

Preliminary work is reported on a program to reduce the amounts of activities in the plant wastes. Design studies are reported for an evaporator to concentrate the most active non-fission metal liquid wastes to increase storage capacity. A survey of the best methods of eliminating or removing particles of UO₂ from the air discharged from the pile is presented.

136

RADIOCHEMICAL WASTE PROCESSING EVALUATION OF ANTIFOAM AGENTS FOR PLANT WASTE EVAPORATOR. p. 29-31 of CHEMICAL TECHNOLOGY DIVISION PROGRESS REPORT FOR MONTH OF DECEMBER 1949. W. K. Elster. March 22, 1950. 33p. (ORNL-640) SECRET.

Tabulated results of tests to evaluate numerous anti-foam agents show that the following agents gave foam reduction factors of 3 or 4: "Alrowax-140," "DC-200," "Nonisol-300," "Oleyl 0-8," "OPE-1," and "Span 20." The antifoam agents were evaluated using two solutions from the representative evaporator runs, one which foamed in the plant at the start of evaporation and one which did not foam. An ionic analysis of the two plant waste solutions is given in tabular form.

137

RADIOCHEMICAL WASTE PROCESSING. p. 17-19 of PILOT PLANTS SECTION REPORT FOR JULY-AUGUST, 1950; CHEMICAL TECHNOLOGY DIVISION. D. G. Reid. Oct. 10, 1950. 28p. (ORNL-850) SECRET

A brief summary of operation and development of the radioactive waste evaporator is presented. Details of the planned vapor scrubber to be installed in the waste evaporator, including a sketch drawing, are given.

138

RADIOACTIVE GAS SEPARATION. p. 55-8 OF PUREX PILOT PLANT QUARTERLY REPORT FOR MAY, JUNE, AND JULY, 1951; CHEMICAL TECHNOLOGY DIVISION. D. O. Darby. Nov. 14, 1951. 74p. (ORNL-1115(DeL)) CONFIDENTIAL

Equipment for the radioactive gas separation process is described and a schematic diagram of the system is shown.

139

WASTE TREATMENT STUDIES. p. 13 of CHEMICAL TECHNOLOGY DIVISION SEMIANNUAL PROGRESS REPORT FOR PERIOD ENDING MARCH 31, 1954. June 29, 1954. 28p. (ORNL-1708(DeL.)) SECRET

It is now planned to store the radioactive wastes from Hope type processes and the ORNL 25 recovery process in open basins in the earth rather than in the carbon steel, stainless steel, or concrete tanks now used for similar wastes. All the commercially available asphalts and tars that have been considered as lining material for these basins have deteriorated when they were exposed to activities considerably below those anticipated in the waste solutions. Two special modified asphalts with softening points of 278 and 256°F showed a 30% increase in volume when γ irradiated to 5×10^8 r (equivalent to 20 to 100 yr of exposure, through 2 ft of earth shielding, to 500-c/gal waste) and were severely attacked when immersed in boiling synthetic Hope type process waste (1.6M Al(NO₃)₃, 0.2M HNO₃, 0.02M H₂SO₄) for 3 hr. Four modified tars with softening points of 237, 199, 192, and 158°F were not affected by exposure to 5×10^8 r of γ radiation or by immersion for two days in synthetic Hope process waste solution at 150°F. (The applicable portion of this report appears in its entirety.)

140

CERAMIC ASPECTS OF WASTE DISPOSAL. p. 122-127 OF METALLURGY DIVISION SEMIANNUAL REPORT FOR THE PERIOD ENDING APRIL 10, 1955. Nov. 7, 1955. 144p. (ORNL-1911(DeL.)) SECRET

Experiments to investigate the possibility of incorporating radioactive isotopes contained in Hope solution into a ceramic body are reported. The method studied involves mixing the liquid waste with clay, limestone, soda ash, or other material to form a gel or slurry which is subsequently dried and sintered to produce a mass in which the isotope may be fixed and from which the isotope cannot be leached. No results of the experiments are given. Experiments to test the ability of concrete to contain Hope solution are also reported in progress. Results of experiments to indicate the power generation required to bring a clay-flux mixture containing fission product waste to 900°C are shown in graphical form. The time-temperature plot in the results indicate that ~100W would have brought the temperature up to 900°C in ~29 days. Procedures, computations, and results of viscosity studies of selected clays for use in lining waste disposal pits show that the Hope solution increases the viscosity of Panther Creek bentonite, whereas W-8 solution decreases it. Both solutions lower the viscosity of the Wyoming bentonite very considerably, the W-8 even more effectively than the Hope. The viscosity of the Gleason ball clay is increased quite appreciably by both solutions. Mixtures of 10, 25, and 30 g of Gleason ball clay in 100 cc of distilled water show a consistent increase in apparent viscosity with increasing clay content.

141

WASTE DECONTAMINATION. p. 11-12 of CHEMICAL TECHNOLOGY DIVISION SEMIANNUAL PROGRESS REPORT FOR PERIOD ENDING MARCH 31, 1956. June 22, 1956. 57p. (ORNL-2079) CONFIDENTIAL

The present philosophy on waste treatment for the Hope project is to remove the long-lived radioactive fission products Cs and Sr and to store the remaining waste in inexpensive pits. A schematic diagram of the diban-ion-exchange process for removing long-lived fission products from Al-containing waste is shown in which synthetic waste was decontaminated by a factor of 220. In studies on removal of Al(NO₃)₃ from wastes containing fission products by crystallization, tracer Cs and Sr were separated from Al by factors of greater than 60 and 130, respectively, in one crystallization. In a second crystallization the DF, S were greater than 60 and 10, respectively. A schematic diagram for the crystallization method is included.

142

WASTE DISPOSAL. R. L. Hammer and M. P. Hayden. p. 191-198 of METALLURGY DIV. SEMI-ANNUAL PROGRESS REPORT FOR PERIOD ENDING APRIL 10, 1956. 237p. (ORNL-2080) SECRET.

Progress is reported in fission product fixation in ceramic materials, and fission product removal from solutions by clays.

143

WASTE DISPOSAL. R. L. Hammer. p. 210-12 of METALLURGY DIVISION SEMI-ANNUAL PROGRESS REPORT FOR PERIOD ENDING OCTOBER 10, 1956. Dec. 19, 1956. 252p. (ORNL-2217) SECRET.

Continued studies to investigate the feasibility of disposing of radioactive wastes in ceramic materials are reported. Chemical analysis of samples taken from various locations within a sintered cake revealed inhomogeneity in respect to composition. As a result of the inhomogeneity characteristics found in the cake, experiments are reported to determine the thermal conductivity of the clay-flux mix. Experiments to determine the fixation of radioisotopes in clay-flux mixes and in Hope solution residue are reported. Counts were made on leach solutions after 11 months' contact with clay-flux mix which had been tagged with Sr^{90} , Ce^{144} , Ru^{106} , Rh^{106} , Cs^{137} , Ba^{137} , and mixed fission products and fired at temperatures from 850 to 1750°F. With the exception of the Sr^{90} -tagged mix fired at 850°F and the mixes containing Cs^{137} and Ba^{137} there was no appreciable radioactivity in the leach solutions.

144

WASTE TREATMENT AND DISPOSAL. R. J. McNamee, F. L. Rogers, S. G. Kent, I. R. Higgins. p. 40-2 of CHEMICAL TECHNOLOGY DIVISION MONTHLY PROGRESS REPORT FOR MARCH 1957. Aug. 12, 1957. 96p. (ORNL-2307) CONFIDENTIAL.

A proposed new method for permanent radioactive waste disposal is briefly reported. The method consists of mixing solids resulting from evaporating waste solutions with asphalt, thereby forming a mass from which it is hoped activity cannot be leached. A mixed (anion- and cation-exchange) resin bed has been proposed for treating the water of water-cooled reactors for protection against corrosion. A schematic diagram of the unit is included. The adequacy of the resin method is discussed.

145

WASTE TREATMENT STUDIES. R. J. McNamee, A. R. Irvine. p. 48 of CHEMICAL TECHNOLOGY DIVISION MONTHLY PROGRESS REPORT FOR APRIL, 1957. Aug. 12, 1957. 72p. (ORNL-2324) CONFIDENTIAL.

Studies on disposal of solid wastes by mixing with asphalt, as previously presented in (ORNL-2307), are reported. Continued tests on leaching of fission products from the asphalt mixture revealed that after an initial build up of activity in the leach water equal to 0.2 and

0.4% of the β and γ activity, respectively, the activity leveled off at an \sim constant value, indicating that little further leaching is taking place. The results of tests for 4 consecutive weeks are given in tabulated form.

146

WASTE STUDIES. I. R. Higgins. p. 32-3 of CHEMICAL TECHNOLOGY DIVISION MONTHLY PROGRESS REPORT FOR JUNE 1957. Nov. 6, 1957. 75p. (ORNL-2362). CONFIDENTIAL.

Results of waste treatment testing of high level solvent extraction raffinate by electrolytic destruction of nitrate, caustic regeneration, and plating of the Ru using an Fe cathode and stainless steel anode showed nitrate destruction of 60-80%. Ru was plated on the cathode. ORNL wastes run about 20g nitrate/liter, so costs would be \sim 1.5¢/gal for electricity (at 0.5¢/kwh).

147

WASTE STUDIES. p. 30-1 of CHEMICAL TECHNOLOGY DIVISION MONTHLY PROGRESS REPORT FOR AUGUST 1957. Jan. 9, 1958. 83p. (ORNL-2400). CONFIDENTIAL

A new process and flowsheet for ORNL waste is presented. At the present time these wastes are stored in ground surface pits. Naturally occurring Conasauga shale possesses sufficient ion exchange capacity to sorb all the fission products except Ru, but the Ru, together with the nitrate ions, which escapes directly to the ground water and may eventually contaminate the environment. In the new process, waste at a rate of about 5,000 gal/day would be sent to either a continuous ion exchange contactor or a shale-filled pit where Cs, Sr, and the rare earths would be removed. Data show that a 6 in. dia. column would be required. Based on recent determinations by the Health Physics Division, approximately 35 yd³ of shale per year would be required as an alternative. For the removal of nitrate and Ru, two types of electrolytic cells are being considered. An acid-base cell using anion- and cation-exchange membranes would require about 35 ft² each of electrode area and a 60-kw power supply. A nitrate reduction cell would be simpler in construction, not requiring membranes and compartments, but possessing a total of about 280 ft² of electrode area and a 500-kw power supply. The latter cell may be somewhat more efficient for Ru plating. On the other hand, if Ru removal only is desired without nitrate or caustic destruction or recovery, a much simplified electrolytic cell may be used. Considerably more laboratory work will be required before the final flowsheet conditions and equipment may be chosen. (The applicable portion of this report appears in its entirety in this abstract).

148

WASTE DISPOSAL. p. 198-201 of METALLURGY DIVISION ANNUAL PROGRESS REPORT FOR PERIOD ENDING OCTOBER 10, 1957. Dec. 13, 1957. 283p. (ORNL-2422). SECRET

Simulated Hope waste solution experiments to investigate the retention of fission products (without prior separation) in fired ceramic bodies are reported. Materials tested were ball clay, bentonites, phosphate slimes, shales, and attapulgites. Best results were obtained with shale and attapulgite.

149

WASTE DISPOSAL RESEARCH AND ENGINEERING. p. 19-21 of STATUS AND PROGRESS REPORT FOR DECEMBER 1957. Jan. 9, 1958. (ORNL-2458). CONFIDENTIAL.

Status and progress is reported on the following studies: disposal of high-level radioactive waste by sintering; disposal of radioactive wastes in geologic structures; and dispersal of radionuclides in the environment. Soil disposal of intermediate-level radioactive wastes is briefly discussed on p. 30.

Oak Ridge National Laboratory
Oak Ridge, Tennessee

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AN AERIAL SURVEY OF RADIOACTIVITY ASSOCIATED WITH ATOMIC ENERGY PLANTS. F. J. Davis, Lt. W. E. Harlan, P. A. Humphrey, Lt. R. L. Kane, and P. W. Reinhardt. April 13, 1949. 165p. (ORNL-341). SECRET

A group of laboratory instruments as airborne detectors of radioactivity are compared and, simultaneously, data relative to the diffusion rate of radioactive contamination emitted into the atmosphere from off-gas stacks of production runs are obtained. Research was conducted in the Oak Ridge and Hanford areas. Detection was accomplished at a maximum distance of seventeen miles from the plant. Very little information of a conclusive nature was gained concerning the diffusion. Further research with the nuclear instruments, using a stronger source, is recommended. To obtain conclusive information concerning the meteorological aspects of the project, a larger observational program will be needed.

Oak Ridge National Laboratory
Oak Ridge, Tennessee

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EVALUATION OF FULL SCALE SAVANNAH RIVER PROJECT EVAPORATOR. W. B. Watkins. Nov. 19, 1951. Changed from SECRET Nov. 29, 1955. 20p. (CF-51-11-113). CONFIDENTIAL

Data obtained from the operation of the full-scale Savannah River type evaporator now installed in the ORNL pilot plant are summarized. The program was initiated to obtain data that would guide current and future evaporator design in connection with the Savannah River Project. The principal variable considered was the effect of boil-up rate on the over-all decontamination factor obtainable from the evaporator plot to the condensate.

232

RESULTS OF EVALUATION OF THE SAVANNAH RIVER TYPE EVAPORATOR USING SYNTHETIC HIGH ACTIVITY WASTE. W. B. Watkins. Dec. 14, 1951. Changed from SECRET Nov. 18, 1955. 6p. (CF-51-12-118). CONFIDENTIAL

The results of this study show that the Savannah River high activity evaporator can be expected to give an over-all decontamination factor (evaporator liquid to condensate) of at least 3.5×10^5 when operated at its designed capacity of 87.8 lb/hr/ft² boil up rate and 3.1 ft/sec superficial velocity in the column. At boil up rates of 115 to 125 lb/hr/ft² and column superficial vapor velocities of 5 to 7 ft/sec, (which are 150 to 200% of designed capacity) the over-all decontamination factor is likely to be from 5×10^5 to 1×10^6 . While it was impossible to operate the ORNL evaporator at a boil up rate of 187 lb/hr/ft² which would simulate the low activity unit, it is believed that the over-all decontamination which could be expected from this unit would be no greater than the value of 1×10^6 which was obtained at a 115 to 125 lb/hr/ft² boil up rate. A heavy brown precipitate, believed to be MnO₂, which formed upon initial heating of the synthetic high activity feed solution, caused considerable difficulty in sampling and in cleaning the evaporator. Traces of the solids were found in the condensate samples obtained from tests at boil up rates over 115 lb/hr/ft² and may have contributed to a lowering of over-all decontamination factor since brief experiments showed that the precipitate carried from 50 to 90% of the activity in the feed solution.

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DISTILLATION OF PUREX WASTES. R. G. Mansfield. April 24, 1952. Changed from SECRET Nov. 10, 1955. 18p. (ORNL-1472). CONFIDENTIAL

Experiments on distillation of Purex waste containing fission products showed that the radioactivity in the distillate is due principally to volatilized Ru. This can be minimized by keeping the still-pot HNO₃ concentration below 9M.

University of Tennessee,
Knoxville, Tennessee

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GEOLOGIC CONDITIONS AT THE OAK RIDGE NATIONAL LABORATORY (X-10) AREA RELEVANT TO THE DISPOSAL OF RADIOACTIVE WASTE. Paris B. Stockdale. Aug. 1, 1951. 88p. (ORO-58). OFFICIAL USE ONLY

An investigation of the geologic conditions at the X-10 site of ORNL relevant to the disposal of radioactive waste is reported. Fifty-one wells were drilled and the ground water tested for radioactive wastes. It was found that the ground water followed the topography and surface drainage and that the surface and ground water ultimately emptied into the intended liquid-waste-disposal system of White Oak Creek and Lake, thus assuring no radiation hazard to an area not already under control or unknown. Recommendation is made to abandon the Bethel Valley burial ground and to establish burial sites in the Conasauga shale belt south-east of Haw Ridge.

Oak Ridge National Laboratory
Oak Ridge, Tennessee

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FEASIBILITY REPORT ON RECOVERY OF FISSION Zr AND Cb FROM REDOX PROCESS WASTE SOLUTIONS

USING SILICA GEL. W. H. Baldwin, J. H. Gross, C. E. Higgins, H. W. Kohn, J. M. Ruth, A. W. Smith, and R. E. Wacker. Sept. 28, 1949. 35p. (CF-49-9-178). SECRET

A series of runs with a Cm² cross-section silica gel column has shown that a process for the separation of Zr-Nb fission activity from 1AW Redox waste solution is feasible. The feed used in these test runs was the 1AW waste from the Redox Pilot Plant Operations at ORNL. A proposed flowsheet for operation with Hanford Redox waste is given. The indicated overall yield is 80 to 85%. This should amount to about 0.40, 0.31, and 0.26 megacuries of activity after 10, 25, and 50 days cooling, respectively. The largest proportion of the indicated 15 to 20% loss occurs in the adsorption step, and appears to be due to a slow attainment of equilibrium at the 30°C temperature used in the column test runs. Batch studies have indicated that equilibrium is reached much faster at higher temperatures; so it appears that the yield may be raised to around 95% by operating the adsorption step at higher temperatures or by operating at slower flow rates per unit cross-sectional area. Six successive runs were made with one column with no indication of decreased silica gel adsorption efficiency.

Oak Ridge Gaseous Diffusion Plant Oak Ridge, Tennessee

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CORROSION OF MILD STEEL IN SULFURIC ACID SOLUTIONS OF SYNTHETIC HANFORD WASTE. H. A. Bernhardt, W. Davis, Jr., and J. R. Flanary. May 10, 1949. Changed from SECRET Feb. 8, 1956. 17p. (K-396; KLO-98). CONFIDENTIAL

The feasibility of dissolving Hanford waste (sludge or composite) in acid prior to its removal from the storage tanks would be determined primarily by the rate of corrosion of the mild steel, from which these tanks are constructed, by acid solutions. HNO_3 solutions, less than 5 M H_2SO_4 solutions (with and without the two inhibitors n-tributylamine and Enthone No. 9), and $\text{HF}-\text{H}_2\text{SO}_4$ solutions containing precipitated UF_4 all corrode mild steel at excessive rates. Only when synthetic waste is dissolved in a large excess of concentrated H_2SO_4 does the corrosion rate of mild steel drop to a value of 0.005 in./3 months, at 25 or 40°C, a value estimated to be the maximum allowable corrosion rate. The long time required to add enough H_2SO_4 to produce a solution of low corrosivity and the evolution of large quantities of CO_2 introduce problems other than those of corrosion.

Oak Ridge National Laboratory Oak Ridge, Tennessee

314
IDAHO CHEMICAL PROCESSING PLANT LIQUID WASTE FACILITIES: DESIGN REPORT. F. N. Browder. Aug. 20, 1954. Changed from SECRET June 14, 1955. 272p. (ORNL-1687). CONFIDENTIAL

The liquid waste facilities of the Idaho Chemical Processing Plant are described in detail, the basis of the design is outlined, and early performance data from operation of the completed plant are presented. Only the facilities for liquid waste handling, exclusive of sanitary waste, are covered.

Oak Ridge National Laboratory Oak Ridge, Tennessee

338
COMPARISON OF THE THOREX AND REDOX AQUEOUS WASTE SYSTEMS. A. T. Gresky and R. P. Wischow. June 17, 1954. Changed from SECRET Feb. 15, 1957. 6p. (CF-54-7-254). CONFIDENTIAL

The results of the investigation indicated that the Thorex hot waste volumes should be at least a factor of 5 lower than Redox wastes, use of stainless steel in lieu of mild steel tanks for storage of the highly acid-deficient AP concentrate, and anticipated Al wastes from a second Th cycle, would decrease the cost savings factor from about 5 to 2.5, and storage of the concentrated Thorex AP stream may become mandatory if the Pa^{233} is to be recovered as U^{233} after cooling; i.e., in the eventuality that short-cooled Th is to be processed without the inclusion of a Pa recovery step.

339
HEAT TRANSFER IN WASTE BASINS. S. H. Jury. Aug. 11, 1955. Changed from SECRET Apr. 19, 1956. 17p. (CF-55-8-76). CONFIDENTIAL

Estimates are made of the heat transfer in proposed out-of-door radioactive waste basins. Calculations are outlined, and complete numerical results are included in appendices.

340
CALCULATIONS CONCERNING ALTERNATE WASTE DISPOSAL METHODS. p.143-149 of A CHEMICAL REPROCESSING PLANT FOR A NUCLEAR POWER ECONOMY. (PROJECT HOPE). R. A. Charpie, J. Halperin, R. J. Klotzbach, J. R. McWhorter, F. Nelson, E. L. Nicholson, C. H. Odom, R. W. Stoughton, E. P. Wigner, and M. R. Zeitlin. Feb. 5, 1954. Decl. June 19, 1959. 166p. (ORNL-1638). UNCLASSIFIED

Calculations are summarized for the computing of the total amount of shadow shield required for personnel protection when the storage scheme is an open tank or pit. Preliminary estimates of alternate liquid waste storage tank installations have been prepared to obtain an order of magnitude of costs. The estimates cover only the purchase, erection, and earth or concrete shielding for a tank farm consisting of three 500,000 gal capacity tanks. The tanks considered are as follows: (1) spherical of mild steel construction internally braced to withstand 10 ft of earth covering, (2) cylindrical of mild steel construction internally braced to withstand 10 ft of earth covering, (3) cylindrical of mild steel construction built to withstand internal pressure of liquid only, and (4) cylindrical of stainless steel construction (347) to withstand internal pressure of liquid only. The total costs of the four systems as outlined above are as follows and are given in detail in tabulated form: underground spherical, \$528,000; underground cylindrical, \$279,000; above ground cylindrical mild steel, \$309,000; above ground cylindrical stainless steel, \$615,000.

Oak Ridge National Laboratory Oak Ridge, Tennessee

349
SHIPPING COSTS OF SAVANNAH ENRICHED FUEL VIA COMMERCIAL CARRIER. R. J. Klotzbach. May 20, 1954. 10p. (CF-54-5-188(Del.)). SECRET

The cost of shipping Savannah River irradiated enriched fuel via commercial carrier to ORNL and to the ICPP has

been estimated for both Railway Express and Motor Freight shipment. The total annual shipping costs for each type of carrier to both destinations at various process rates is shown.

REFERENCES

Massachusetts Institute of Technology.
Engineering Practice School,
Oak Ridge, Tennessee

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REMOVAL OF RADIOACTIVE LIQUID PARTICLES FROM DISSOLVER OFF-GASES. D. W. Bartholomew, P. E. Tausche, and C. C. Williams, III. Aug. 15, 1949. Changed from SECRET Jan. 3, 1956. 30p. (K-484). CONFIDENTIAL

Experimental work was done in a by-pass line already available in the dissolver off-gas line from the Hot Pilot Plant of the Oak Ridge National Laboratory. The tests were made while Hanford slugs, averaging 120 days out of the pile, were being dissolved. The slugs were dissolved in two separate steps, the first step being the dissolution of the Al jackets and the second step the dissolution of the U metal. On the basis of results obtained during the U dissolving operation, it was concluded that the major portion of the liquid particles were so large that they were removed as line condensate before the test apparatus. The remaining particles were so small that they passed through the cyclone separators and were collected on the filter paper used as a backing. In the case of the jacket dissolving operation, considerably more of the particles appeared to range between 5 and 20 μ than in the dissolving run. This conclusion is based on the cyclone performance during the jacket dissolving run. In all cases, the major γ activity of the line condensate was due to the presence of Nb and Zr salts in ~ equal proportions.

Oak Ridge National Laboratory
Oak Ridge, Tennessee

364

A STUDY OF THE CONTRIBUTION OF THE RALA PROCESS TO ATMOSPHERIC CONTAMINATION AT ORNL. R. L. Bradshaw and W. D. Cottrell. Nov. 1, 1954. 35p. (CF-54-11-186). SECRET

Data on the particulate contamination of the atmosphere at the Oak Ridge National Laboratory were correlated with laboratory processes and reactor operations for the period from March 1949 to June 1954. A significant correlation was found to exist between one of the chemical separation processes, RaLa, and peaks of particulate activity on the Laboratory area. The RaLa process was monitored during two complete cycles of operation. The contribution of the process to the general atmospheric contamination of the Laboratory area was determined. The activity was identified and its release to the atmosphere was investigated. It was concluded that the major portion of the contaminated RaLa effluent released to the stack does not fall out or diffuse in the immediate environs, and hence, the stack is not the primary means by which RaLa contributes to atmospheric contamination. A serious offender was found to be an underground liquid waste storage tank in the south tank farm. This tank is used to store RaLa wastes and is vented to the atmosphere. Atmospheric contamination in the immediate vicinity of this vent was observed to reach values as high as 2×10^{-5} $\mu\text{c}/\text{cc}$, β activity, averaged over a 24-hr period.

Oak Ridge Gaseous Diffusion Plant
Oak Ridge, Tennessee

395

PILE COOLING AIR FILTRATION WITH BARRIER MATERIALS. D. W. Magnuson. October 31, 1949. 20p. (K-516). SECRET

The efficiencies of barrier, and CWS #6 filters were less than 70% as determined by the activities found when pile cooling air was passed through these filters for extended periods. The hypothesis that decay in flight of radioactive gases between filters which may partially or completely explain the low filter efficiencies is substantiated. Plugging of the first filter in the series was observed and was due to the normal atmospheric dust collected on the filter. Barrier plugging was much less than CWS #6 filter plugging. In the last run of 55 days, the total activity in the pile cooling air, calculated from the filter activities, was 800 mc of β activity and 500 mc of γ activity (1 Mev equivalent).

396

THE CONVERSION OF FLUORINE CONTAINED IN WASTE GAS TO CHLORINE TRIFLUORIDE BY REACTION WITH CHLORINE. R. L. Jarry and W. Davis, Jr. Jan. 7, 1953. 11p. (K-977). SECRET

In order to reduce F_2 concentration in waste gas, a ClF_3 unit was constructed and placed in operation in the feed plant. Data have been obtained for conversion of F_2 in the stack gas to ClF_3 at concentrations of F_2 from 25 to 55%. The reaction temperature was 540°F, the contact time in the reactor 2 minutes, and the mole ratio of F_2 to Cl_2 kept at 3:1. Mass spectrometer scans of the product for analytical purposes showed results essentially the same as those for standard samples of ClF_3 . Some recommendations are included for setting up a large-scale unit.

Oak Ridge National Laboratory
Oak Ridge, Tennessee

397

DESIGN OF AN ADSORBER FOR REMOVAL OF KRYPTON AND XENON FROM DISSOLVER OFF GAS STREAM [FROM CPP 601]. J. M. Holmes. June 20, 1951. 29p. (CF-51-6-70). SECRET

The calculation for the design of an adsorber for the removal of Kr and Xe from the dissolver off gas stream is given. It appears that the three-month charcoal bed would be the most economical. It would be possible to use vacuum as insulation but the mechanical design problems and the maintenance problem in a hot area indicate that the use of insulation would be more economical and would simplify operation.

398

OFF-GAS PROCESSING STUDIES CT-32. E. P. Reichardt. Oct. 8, 1951. 11p. (CF-51-10-128). SECRET

A resume of the Idaho dissolver off-gas treatment is presented. Equipment and operation data are given. A summary of 4 test runs, using actual off-gas mixtures indicate that the process will work satisfactory. Recommendations for improving the system are included.

399

REVISED CALCULATIONS OF OFF-GAS ACTIVITY LEVELS. H. Weeren. Oct. 22, 1951. 7p. (CF-51-10-147). SECRET

Preliminary calculations have been made to determine the intensity of radiation that will be given off by the I_2 unit, the charcoal bed, and the gases coming from the charcoal bed during regeneration. These calculations have been made by assuming the worst possible conditions.

400

DISSOLVER OFF GAS RATE OF EVOLUTION AND COMPRESSOR PERFORMANCE IN IDAHO CHEMICAL PROCESSING PLANT. William L. Carter. Nov. 8, 1951. 13p. (CF-51-11-34). SECRET

In order to choose compressors of adequate capacity for removing the gases produced in the chemical reaction of dissolving fuel elements, a set of experiments were performed to determine the rate at which gases would be produced. Data presented show that the maximum rate of gas evolution will occur during MTR dissolving and will be on the order of 13 SCFM. Calculations to determine the number of MTR dissolving that can be started at arbitrarily specified time intervals without exceeding a compressor capacity of 20 SCFM are given.

401

DESIGN OF THE DISSOLVER OFF-GAS SYSTEM FOR THE IDAHO CHEMICAL PROCESSING PLANT. John M. Holmes. Nov. 5, 1952. Changed from SECRET Apr. 4, 1956. 41p. (CF-52-11-39). CONFIDENTIAL

Pertinent calculations for the WN System adsorption units have been carried out on heat transfer, heat exchangers, and adsorption beds. Static adsorption and dynamic adsorption data are presented for Kr, Xe, Co, and N_2 on charcoal.

402

DESIGN OF A UNIT FOR THE REMOVAL OF IODINE FROM THE DISSOLVER OFF-GAS STREAM IN THE IDAHO CHEMICAL PROCESSING PLANT. William L. Carter. Nov. 5, 1952. 33p. (CF-52-11-48). SECRET

Iodine may be removed from the dissolver off-gas stream by passing the gas stream at an elevated temperature through a bed of ceramic porcelainized berl saddles which have been coated with $AgNO_3$. Any I_2 present will react quantitatively with the $AgNO_3$. The optimum operating temperature is 375°F. Design specifications and operating procedures for the I_2 removal unit are included.

403

PILOT PLANT RADIOACTIVE GAS SEPARATION PROCESS EQUIPMENT AND OPERATING INSTRUCTIONS. W. G. Watkins, W. L. Poe, and D. R. Lindow. Nov. 12, 1952. 45p. (CF-52-11-82). SECRET

The Radioactive Gas Separation Process (RAGS) to remove radioactive elements, particularly Kr^{85} from the gas evolved during the dissolution of irradiated fuel units, was installed in the ORNL pilot plant to confirm the feasibility of the laboratory flowsheet and to secure sufficient data to support an economic study of a plant scale installation. The process was operated in distinct phases; (1) the dissolver off-gas was continuously processed through the entrainment separator, condenser, I^{131} absorption tower, filter, acid tower and discharged into the gas holder; (2) the gas was continuously processed through the caustic scrub tower, Al_2O_3 dryer, cold trap and silica gel adsorption unit; (3) the Kr and Xe were desorbed from the silica gel bed and stored. The off-gases from 24 runs were processed through the equipment. The adsorption of Kr on silica gel at $-183^\circ C$ was demonstrated in six successive runs. The equipment used in the pilot plant and its operation is described. This manual supplements (ORNL-1410)(Ref. No. 406) entitled Summary of the Pilot Plant Development of the Radioactive Gas Separation Process, which describes the results of pilot plant experimental program.

404

RECOVERY OF NITROGEN OXIDES AND RARE GAS FISSION PRODUCTS FROM THE NITRIC ACID DISSOLUTION OF IRRADIATED URANIUM. A. T. Grealy. Apr. 30, 1952. 90p. (ORNL-1208). SECRET

Data obtained in laboratory studies of the fumeless dissolution of irradiated U are presented. Procedures for the removal or recovery of the nitrogen oxide products formed in the reaction of HNO_3 with U or Al and the collection and concentration of fission product gases composed principally of I_2 , Xe, and Kr are proposed.

405

REMOVAL OF NITROUS OXIDE AND KRYPTON FROM URANIUM DISSOLVER OFF-GAS. T. S. McMillan and W. L. Johnson. Dec. 20, 1952. 24p. (ORNL-1300). SECRET

Process conditions are presented whereby a flowing gas mixture, similar to that liberated during dissolution of irradiated U may be stripped of N_2O by thermal decomposition and of Kr by adsorption on either silica gel or carbon.

406

SUMMARY OF THE ORNL PILOT PLANT DEVELOPMENT OF THE RADIOACTIVE GAS SEPARATION PROCESS. W. B. Watkins. Mar. 17, 1953. 74p. (ORNL-1410). SECRET

This report describes the pilot-plant investigation of a process for the recovery of nitrogen oxides and the removal of Kr^{85} from the gas evolved during the nitric HNO_3 dissolution of irradiated U. An economic study of a plant-scale process is included.