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ORNL-TM-2910

RADIOISOTOPE PROGRAM (8000) PROGRESS REPORT FOR FEBRUARY 1970

A.F. Rupp

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Contract No. W-7405-eng-26

Isotopes Development Center

RADIOISOTOPE PROGRAM (8000) PROGRESS REPORT FOR FEBRUARY 1970

A. F. Rupp

MARCH 1970

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RADIOISOTOPE PROGRAM (8000) PROGRESS REPORT FOR FEBRUARY 1970

A. F. Rupp

RADIOISOTOPE PRODUCTION AND MATERIALS DEVELOPMENT

NEUTRON PRODUCTS - 08-01-01

A. Biomedical Radioisotopes

1. Thulium-171

Thulium-171 is a 1.9-year pure beta emitter with a maximum beta energy of 0.097 MeV; it is potentially useful as a tracer in biology and biomedical experiments. It can be made in relatively high isotopic abundance, free of the more energetic thulium-170, by irradiating enriched erbium-170.

A cooperative study involving the ORNL Biology and Health Physics Divisions has been initiated to study the relation of air pollutants to lung carcinomas. Carcinogens enter the respiratory tract either as particulates or bound to particulates. A study of alteration of deposition of particles in the lung and their elimination from the lung following respiratory infection is therefore of great importance in understanding the possible role of respiratory infection in the development of lung cancer. A radioactive dust is used in the first phase of these experiments to study particulate deposition and removal rates; a carcinogen will be used in later experiments. This study will be made using inbred mice with and without induced viral and/or bacterial pneumonitis. The radioactive dust is introduced into the animals by intravenous or intratracheal injection and by inhalation. The effect of infection on distribution in the lung will be studied by autoradiographic techniques. The elimination of particles from the lung can be followed by counting the emitted radiation from the dissected lungs and major respiratory organs in a gamma spectrometer.

For the initial stages of this project, a highly insoluble Cr_2O_3 dust with known particle size distribution was activated to form $^{51}Cr_2O_3$, which was made into a gelatin-stabilized suspension for intravenous and intratracheal administration. These studies will be correlated with previous ones that used the inactive form of the same dust. Similar studies with an $^{59}Fe_2O_3$ preparation will be made. Future work will entail tagging thulium-171 on a suitable aerosol carrier of known particle size distribution as a tracer to study long-term effects.

2. Phosphorus-33

The purpose of this project is to develop methods of preparing hundred-millicurie quantities of carrier-free phosphorus-33 containing $\leq 5\%$ phosphorus-32. This radioisotope of phosphorus has a lower energy beta and a longer half-life than phosphorus-32, making it useful for autoradiography work in biochemistry, agriculture, and metallurgy.

Two methods of preparing phosphorus-33 are being evaluated. One method is the irradiation of sulfur-33 enriched to greater than 92% sulfur-33 in a fast neutron flux to prepare the radioisotope by the $^{33}S(n,p)^{33}P$ reaction. The other method of preparation is the fast neutron irradiation of chlorine-36 to prepare phosphorus-33 by the $^{36}Cl(n,\alpha)^{33}P$ reaction. Highly enriched targets are necessary to reduce the phosphorus-32 content of the phosphorus-33 produced from either of the targets. The progress on each method of production is discussed below.

a. Phosphorus-33 Preparation from Enriched Sulfur-33

The sulfur-33 target material recovered from previous irradiations (potassium sulfate, anhydrous) used for phosphorus-33 preparation was purified for further irradiation. Additional target material was added to increase the amount from 393 to 873 mg of anhydrous $K_2{}^{33}SO_4$ and to obtain an increase in yield of phosphorus-33 from 230-250 mCi to approximately 550 mCi per irradiation. Irradiation of the larger target will begin February 25, 1970, in a fast flux of 2.6×10^{14} n/cm^2.sec.

b. Phosphorus-33 Preparation from Enriched Chlorine

The purpose of this work is to prepare highly enriched chlorine-36 for target material for future phosphorus-33 production by the $^{36}Cl(n,\alpha)^{33}P$ method and to supplement current phosphorus-33 needs now normally supplied by the $^{33}S(n,p)^{33}P$ method.

The chlorine being enriched in chlorine-36 was previously irradiated for approximately 10 years in an unknown neutron flux. This material was then irradiated for 331 days in a thermal neutron flux of 7×10^{14} n/cm^2.sec and a fast neutron flux (>1 MeV) of 2.4×10^{14} n/cm^2.sec. The isotopic mass compositions of the chlorine before and after the 331 days of irradiation are shown in Table 1.

Table 1. Isotopic Mass Composition of Chlorine Before and After Irradiation for 331 Days in a Flux of 7×10^{14} n/cm^2.sec

Isotope	Natural Abundance, at. %	Initial Target Material,[a] at. %	Final Target Material, at. %
Chlorine-35	75.53	28.04	6.47
Chlorine-36[b]	-	47.02	67.94
Chlorine-37	24.47	24.94	25.59

[a]After irradiation at unknown flux thought to be ~10^{13} to 10^{14} n/cm^2.sec.
[b]Half-life is 3×10^5 years.

The change in isotopic abundance of the chlorine-35 from 28.0 to 6.47% was used to calculate the $^{35}Cl(n,\gamma)^{36}Cl$ reactor cross section. Our value of the cross section is 40.8 barns, which is in good agreement with a recent value of 41.8 ± 1.2 barns.[1] The cross section of the $^{36}Cl(n,\gamma)^{37}Cl$ reaction

[1]G. H. E. Sims and D. G. Juhnke, The Thermal Neutron Capture Cross Section and Resonance Capture Integral of ^{35}Cl for (n,γ) and (n,p) Reactions, J. Inorg. Nucl. Chem. 12: 3721-6 (1969).

was calculated from the change in the chlorine-37 atom percent during the 331-day irradiation at 7 x 10^{14} n/cm^2.sec to be 1.0 barn, which is much lower than a previously reported value of 90 ± 25 barns.[2]

These results indicate that very high specific activity chlorine-36 can be made with only small losses due to conversion to chlorine-37.

3. Potassium-43

Nothing to report.

4. Rubidium-84

Nothing to report.

5. Ruthenium-97

Ruthenium-97 has a half-life of 2.9 days and decays by electron capture and the emission of the principal gamma rays of 215 keV (91%) and 325 keV (11%) and higher energy gammas of less than 1% abundance to technetium-97 (2.6 x 10^6 y). The gamma transitions have low conversion coefficients, and their energy and abundance, coupled with the convenient half-life, make this radionuclide promising for scanning applications.

The purpose of this study is to develop the methods for preparing and processing ruthenium-97, derive the production parameters required to establish costs, and characterize the product. An adjunct to these objectives is the establishment of a cooperative program with the Upstate Medical Center of the State University of New York in which the material produced is supplied for evaluation of the biological applications.

Ruthenium-97 can be prepared by the $^{96}Ru(n,\gamma)^{97}Ru$ reaction. A 10-mg sample of enriched ruthenium-96 (94.88%) metal was irradiated for 9.3 days in a flux of 2 x 10^{15} n/cm^2.sec. The metal target was dissolved in a NaOCl solution containing excess NaOH to prevent volatilization of RuO_4. The product was then acidified with HCl to form a chloride salt solution and some of the product was shipped to the cooperating institution.

The specific activity of ruthenium-97 (calculated on target weight versus activity, as of the end of irradiation) was 57 mCi/mg. The only contaminant activities identified were 39.5-day ruthenium-103 (0.82 mCi/mg from ruthenium-102 in the target) and 4.3-day technetium-96 [0.69 mCi/mg from the $^{96}Ru(n,p)^{96}Tc$ reaction]. Calculations based on the yield lead to the value of 0.2 barn for the effective capture cross section (n,γ) of ruthenium-96 compared with the value of 0.27 barn in the Chart of the Nuclides. The amount of ruthenium-103 activity is consistent with that expected from the ruthenium-102 in the target and its 1.23-barn cross section.

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[2] A. W. Boyd, F. Brown, and M. Lounsbury, Mass Spectrometric Study of Natural and Neutron-Irradiated Chlorine, Can. J. Phys. 33: 35-42 (1955).

Preliminary studies at the Upstate Medical Center indicate that human serum albumin (HSA) can be labeled with ruthenium-97 with yields better than 80%. The labeling of HSA with ruthenium-97 was confirmed by distribution studies in mice and by electrophoresis in comparison with iodine-125-HSA. Furthermore, it was found that the label is strong enough to prepare micro- and macroaggregates with ruthenium-97-HSA. These preparations have biological behavior similar to that of iodine-labeled compounds.

In addition, chelates of diethylenetriaminepenta-acetic acid (DTPA) and ethylenediaminetetra-acetic acid (EDTA) with ruthenium-97 were prepared at the Upstate Medical Center. Preliminary studies in rabbits indicate that these chelates are not very stable in vivo and apparently break down in the blood stream. The ruthenium-97 EDTA was more stable than the DTPA preparation. Paper electrophoresis of these chelates showed several species - as many as eight with ruthenium-97 DTPA and three with ruthenium-97 EDTA. Better methods of preparing these chelates are under investigation.

These preliminary studies have been submitted in abstract form for presentation at the Miami meeting of the Society of Nuclear Medicine. Future process development will include the separation of the technetium-96 from the ruthenium-97; technetium-96 has high energy gamma rays and therefore substantially increases the radiation dose.

B. Research Radioisotopes

1. Lanthanum-137

Nothing to report.

C. Isotopic Power

1. Thulium-170

Nothing to report.

D. Miscellaneous

1. HFIR Target Testing

Twenty-six hydraulic tube rabbits and irradiation capsules for the removable beryllium reflector position were tested during February; three of these were nonradioactive rabbits and 17 contained reprocessed zinc and calcium oxides. Of these 20 rabbits, two were found to be defective. Six irradiation capsules were tested and two of these leaked.

After the irradiation capsules have passed the leak tests, they are checked in our gage to assure that their dimensions are correct and that they will fit the reactor holes. Before insertion in the reactor they are checked again in a similar gage by the HFIR operators. The HFIR operators recently began to reject our irradiation capsules because they would not pass their gage inspection. To resolve this problem, both gages were sent to the inspection department to verify their dimensions, and it

was found that the HFIR gage was basically more lenient than ours, but had accumulated some material on it. This accumulation in the gage was apparently preventing capsules from entering. After the HFIR gage was cleaned, it accepted irradiation cans properly, which solved the problem of rejection.

E. Supporting Technology

1. Radioisotope Characterization

Furnishing information in answer to inquiries about characteristics and analysis of radioisotopes and reviewing reports, papers, and books are important functions of the characterization program. The reference book Induced Radioactivity by M. Barbier (Wiley, 1969) was reviewed for Nuclear Science and Engineering. A request for suggestion of a source of electromagnetic radiation of dominant energy below 30 keV was answered by suggesting iodine-125, a 60-day emitter of approximately 130% K X-rays (averaging ~28 keV) plus 7% 35-keV gamma. Information on intensities of external gamma radiation from sources continues to be needed. A value for tantalum-182 was requested and could be answered in principle by reference to equations for the exposure rate in mR/hr at 1 meter from a 1-mCi point source;[3] however, a reference for the actual value, approximately 0.68, was more difficult to supply. Two publications gave erroneous numbers as compared to our estimates; one was the recently reviewed book, Induced Radioactivity, mentioned above. A value sufficient for this purpose was found in the Radiochemical Manual (Radiochemical Centre, Amersham, England, 1966).

2. Radioisotope Special Analysis and Quality Control

Progress continues in the survey on radioactivity standards being carried on for the National Research Council. A draft of the first part of the report, covering introductory material and the questionnaire responses, was presented to the advisory panel during the meeting February 5 and 6, 1970. Members' suggestions are being incorporated in a second draft. The panel received reports from producers and users of radioactivity standards and discussed possible solutions to the problems presented. A request for information and a copy of the questionnaire was received from the Japan Atomic Industrial Forum, in connection with a proposed survey in that country. Information was also sent in response to a request from the quality control committee of the College of American Pathologists.

Unusual chemical analyses are often required in connection with development or routine production of isotope products. A need for analyses for atmospheric gases in inactive cesium chloride was discussed with analytical specialists, and it was decided that vacuum fusion, with measurement of pressure of evolved gas, would be satisfactory. Although the amounts of gas found were small, generally less than 0.1 cm^3 at NTP per gram of CsCl, precision was adequate for the purpose.

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[3] Handbook of Radioactive Nuclides, ed. by Y. Wang, Chemical Rubber Company, Cleveland, 1969.

CYCLOTRON PRODUCTS - 08-01-02

A. Biomedical Radioisotopes

1. Rubidium-83

Nothing to report.

2. Rubidium-84

The objectives of this program are to develop the technology for rubidium-84 production in acceptable purity and quantity to meet the needs of cooperative participants and to establish cooperative programs with medical investigators interested in evaluating this radionuclide as a diagnostic agent in nuclear medicine. The positron emission (intensity = 21%) and accompanying 511-keV annihilation radiations of rubidum-84 have aroused interest in using this radionuclide for myocardial scanning with a positron camera.

Rubidium-84 is routinely produced by the ORNL Isotopes Division by proton bombardment of natural krypton targets, but this product is unsuitable for positron camera scanning due to appreciable amounts of rubidium-83 produced simultaneously. The activity of the rubidium product obtained from proton bombardment of natural krypton in the ORNL 86-Inch Cyclotron is comprised of 9.7% rubidium-83 (83 d), 68% rubidium-84 (33 d), and 22% rubidium-86 (18.7 d) at end of bombardment. The longer half-life of rubidium-83 reduces the $^{84}Rb/^{83}Rb$ ratio even further with time. Appreciable amounts of rubidium-83 are undesirable for positron scintigraphy because its principal gamma emissions [521 keV (46%), 530 keV (31%), and 553 keV (16%)] are similar enough in energy to the 511-keV annihilation radiation to prevent energy resolution by the positron camera, thereby leading to increased background and loss of spatial resolution.

Attempts to produce 8- to 10-mCi amounts of high radionuclidic purity rubidium-84 by reactor irradiation of a 190-mg SrO target (75% isotopically enriched strontium-84) for 46 days in a fast neutron flux of 2.6×10^{14} $n/cm^2.sec$ resulted in a product which contained much less rubidium-84 and much greater amounts of rubidium-83 and rubidium-86 contaminants than was anticipated.[4,5] Extrapolation of cyclotron yields for the production of rubidium-84 from natural krypton targets and the availability of a highly isotopically enriched by-product[6] krypton-84 target was used to evaluate the feasibility of producing an acceptable rubidium-84 product for positron scintigraphy. On this basis a new contamination-free gas target holder was fabricated and the irradiation performed.

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[4]A. F. Rupp, Radioisotope Program (8000) Progress Report for December 1969, USAEC Rpt. ORNL-TM-2828, Oak Ridge National Laboratory, p. 3.
[5]H. B. Hupf and H. A. O'Brien, Jr., The Reactor Production of ^{84}Rb from Strontium, Intern. J. Appl. Radiation Isotopes, 18: 417-20 (1967).
[6]A. F. Rupp, Radioisotope Program (8000) Progress Report for October 1 through November 30, 1969, USAEC Rpt. ORNL-TM-2807, Oak Ridge National Laboratory, p. 14.

An aluminum gas-type target holder was used to contain the enriched krypton-84 target during irradiation. After the irradiation, the krypton-84 target material was pumped out for recovery and the target holder was opened for processing. The rubidium-84 product was recovered by leaching the target with hot sterile water. In order to ensure maximum product recovery, the rubidium-84 product was leached from the target holder with separate 5-ml water portions which were subsequently monitored with a survey meter. It was determined that each hot water leach removed approximately 60% of the water-leachable activity and that five leaches were needed to remove 99% of the leachable activity. Survey meter readings on the target holder before and after the leaching procedure indicated that greater than 95% of the total activity had been removed by the procedure. An attempt will be made to leach the remaining activity with dilute HNO_3 to avoid buildup of the longer lived rubidium-83 which might be removed during subsequent experiments.

The rubidium-84 was converted to the chloride by evaporating to dryness from dilute HCl solution. The final product was taken up with 0.1 N HCl solution (dilute HCl prepared with sterile water), filtered through a 0.45-μ Millipore filter, and shipped to a cooperative participant for evaluation. The iron contamination of the product was determined using a chemical spot test to be less than 5 μg/ml.

Gamma-ray pulse height analysis (PHA) using a calibrated Ge(Li) detector and computer analysis of the PHA spectrum identified photopeaks attributable only to long-lived rubidium isotopes (i.e., ^{83}Rb, ^{84}Rb, or ^{86}Rb). Analysis of the areas under the photopeaks indicated a radionuclidic composition of the product at end of bombardment of 1.35% rubidium-83, 95.8% rubidium-84, and 2.8% rubidium-86. The $^{84}Rb/^{83}Rb$ ratio of this product, the pertinent factor for determining improved positron camera performance, was an order of magnitude larger than the product obtained from proton irradiation of a natural krypton target.

On the basis of the recovered rubidium-84, a cyclotron production rate and cyclotron yield of 120 mCi/mA-hr and 3.75 mCi/hr, respectively, were determined for this experimental configuration. Further development work aimed at gaining additional experience on target recovery (estimated in-house direct cost of target material ~$7000), a cyclotron production rate and yield, and improved processing procedures to avoid possible sources of pyrogen contamination are planned if this material is found acceptable by our cooperative participant for positron camera scintigraphy.

3. Indium-111

The objectives of this program are twofold: (1) to elucidate, define, and optimize pertinent production parameters (cyclotron target material, cyclotron beam energy and current conditions, adverse side reactions leading to undesirable isotopic impurities, target processing and product purification procedures, and target recovery, if necessary) and (2) to make sufficient material available to cooperative participants for their evaluation of indium-111 as a diagnostic radionuclide in medicine. Such applications would include spinal-cerebral cisternography, aerosol lung studies, delayed

brain scanning, visualization of the lymphatic system, metabolic studies of indium-labeled macroaggregates and colloids, and possible applications for tumor localizations. Indium-111 has both a favorable decay scheme [gamma emissions of 173 keV (89%) and 247 keV (94%)] ideally suited for external detection and an optimal half-life (2.81 days) for labeling and distribution studies which must be carried out over 24 hr or longer. The use of indium in such lengthy studies is currently impossible with the indium-113m (1.73 hr) isotope now in general use.

The use of this product in humans makes product characterization and quality control very important and necessary items. In an attempt to completely characterize the chemical composition of the initial indium-111 products being produced by the HBr-isopropyl ether extraction procedure currently being investigated, an aliquot of the purified product was analyzed by spark-source mass spectrometry. The analysis of the product prepared during January[7] indicated that zinc (less than 5 μg/ml) and cadmium (less than 20 μg/ml) contamination was no longer a problem. Iron contamination (approximately 40 μg/ml) of the product was still too high to be considered acceptable for this product.

In order to reduce the iron impurities an 8 M HCl "strip" solution was used in the February development run to back-extract the carrier-free indium-111 product from the organic phase. Preliminary tracer experiments using carrier-free indium-113 (III) had shown that: 1) a distribution ratio of $\leq 10^{-2}$ could be expected for indium (III) in the HCl-isopropyl ether system under the desired conditions (8 M HCl) and 2) an efficient separation of indium from macro amounts of iron (III), which remains in the organic phase, can be achieved without appreciable product losses. A chemical process yield of approximately 94% was obtained (down from approximately 98% obtained using a 0.1 M HCl "strip" and fewer "wash" steps), but the iron (III) contamination of the product, as determined by chemical spot tests, was reduced to less than 5 μg/ml, which is considered acceptable.

Additional operating experience pertaining to the thick-target production rate and cyclotron yields for the $^{111}Cd(p,n)^{111}In$ reaction was obtained. A thick-target production rate of approximately 510 mCi/mA-hr would seem routinely possible by using a 0.060- by 1.25-in. capsule void versus a rate of approximately 420 mCi/mA-hr for the 0.100- by 0.625-in. void capsule used in earlier experiments. This improvement has led to an approximately 20% increase in cyclotron yield. Four shipments of indium-111, totaling approximately 110 mCi, were supplied to our cooperative participants for their application and evaluation.

Additional data were obtained confirming the amount of indium-114m (50 d) produced simultaneously by the $^{114}Cd(p,n)^{114m}In$ reaction on isotopic impurities in the target material.[8] An indium-114m radionuclidic impurity level of 0.011% at the end of bombardment has been determined as characteristic of the 96.5% isotopically enriched cadmium-111 target material in current use.

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[7]A. F. Rupp, Radioisotope Program (8000) Progress Report for January 1970, USAEC Rpt. ORNL-TM-2876, Oak Ridge National Laboratory, p. 7.
[8]A. F. Rupp, ORNL-TM-2828, p. 9.

Fabrication has begun on an alternative target design for the production of indium-111 by the $^{112}Cd(p,2n)^{111}In$ reaction. Testing of this new design, which employs cadmium metal rather than the oxide, is planned. Initial test models will use naturally abundant cadmium to avoid committing a large inventory of isotopically enriched cadmium-112 until the design has proved feasible.

Another development run to produce indium-111 by the $^{111}Cd(p,n)^{111}In$ reaction has been scheduled for early March to gain additional operating experience on the HBr-isopropyl ether extraction process, to check the reproducibility of the pertinent production parameters (chemical yield, production rate and yield, chemical and radionuclidic purity of the product, target recovery, and iron decontamination), and to provide sufficient material to cooperative participants for their evaluation. New target design, fabrication, and testing will be carried out parallel to the $^{111}Cd(p,n)^{111}In$ production so that a reliable source of indium-111 can be maintained for our cooperative participants.

4. Gallium-67

The objectives of this program are to determine the optimal target configuration of gallium-67 (78 hr) production in acceptable purity and quantity and to provide gallium-67 for clinical applications research and development. Interest in this isotope has been spurred by evidence, obtained by the Medical Division of Oak Ridge Associated Universities (ORAU), of a high uptake of carrier-free gallium-67 by lymphoid tumors in both animals and humans. This interesting discovery was made during the evaluation of gallium-67, produced in the ORNL 86-Inch Cyclotron by the $^{68}Zn(p,2n)^{67}Ga$ reaction, as a bone-scanning agent for a patient with Hodgkin's disease.

Two gallium-67 preparations were made and shipped to ORAU, and five smaller orders for gallium-67 were filled for clinicians interested in evaluating the potential of this radionuclide for their applications.

Two new targets of natural zinc electrodeposited on copper have been fabricated similar to earlier models[9] and will be tested to determine whether these later models can stand higher cyclotron beam current without melting.

5. Neodymium-140

Nothing to report.

FISSION PRODUCTS - 08-01-03

A. Krypton-85 Enrichment

One liter of product at 12.5% krypton-85 (approximately 170 Ci) was removed from column D (see Fig. 1). The depleted ends were removed and replaced

[9] Ibid., p. 8.

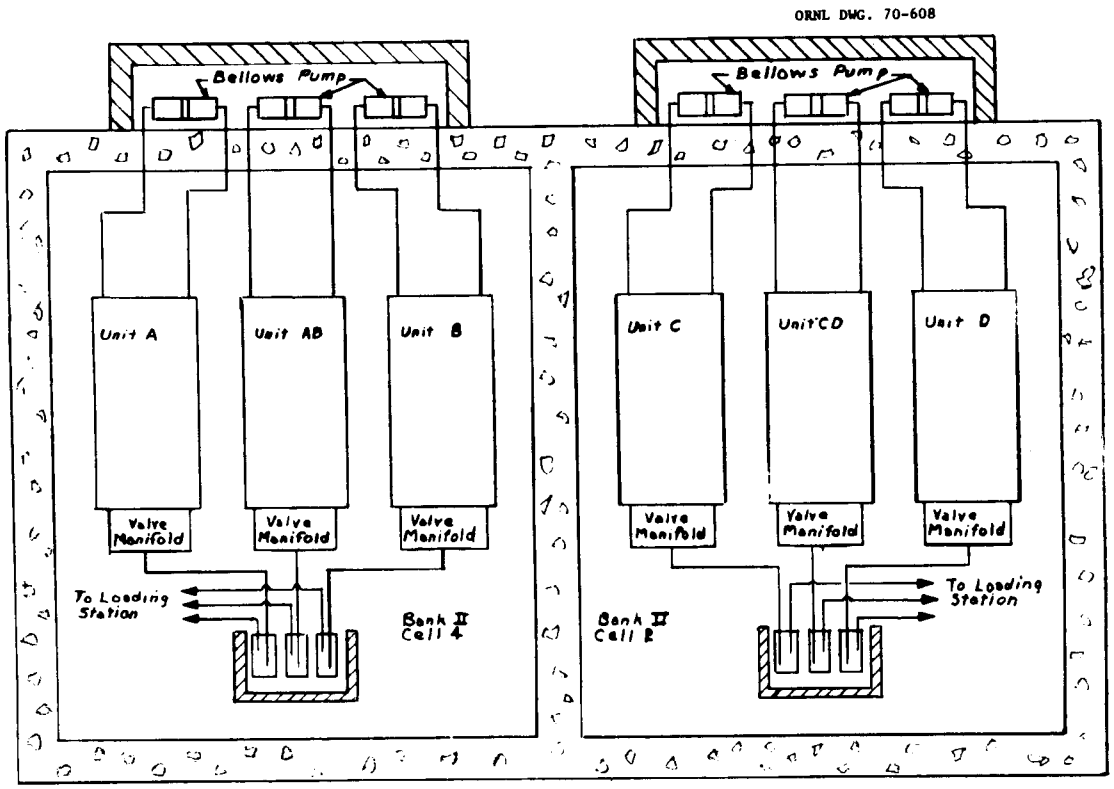

Fig. 1. Schematic Arrangement of Krypton-85 Columns.

with normal 3.5% feed material. The total activity contained in each unit, the elapsed operating time on the current runs, and the average count rate at each end of the center product section are shown below:

Unit	Activity in Unit, Ci	Time Since Last Product Removal, days	Count Rate in Product Section, counts/min	
			January 1970	February 1970
A	515	95	5,200	5,400
AB	1,071	65	7,350	8,200
B	843	65	6,400	7,500
C	1,044	65	4,200	2,600
CD	1,906	119	11,250	13,550
D	1,186	14	9,200	5,150

Column C was not building up its gradient in the expected manner, and investigation showed the cam inside the bellows transport pump had sheared loose from the drive shaft, stopping the intercolumn transport of gas. The pump is now being replaced with a standby unit. While gas was being removed from the ends of the system and from the bellows pump, the existing gradient was lost, which is shown by the decrease in the product section count rate.

The cooling water system had accumulated a great deal of dirt since the last maintenance shutdown; so the entire system was turned off for one day on February 19 and the dirty water drained, the cooling tower washed

down, the thermal diffusion units flushed clean, and the water filters cleaned. The system was put back in service with no further difficulties.

The automatic bleed line which froze shut in January during the cold weather has been steam traced. This should prevent similar trouble in the future.

B. Cesium-137 Process and Product Technology

1. Fission Product Process Study

The purpose of this project is the study and development of economical processes for recovery, purification, and conversion of fission products (cesium-137 in particular) obtained from reactor fuel reprocessing operations.

The current objectives are to improve the accuracy of the differential thermal analysis (DTA) unit to be used in the study of the CsCl-KCl system and to continue the process economics study. The study on the equilibrium phase behavior of the CsCl-KCl system was initiated to investigate the possible absence of the solid phase transition of CsCl containing a small amount of KCl.[10]

Considerable improvements have been made in the accuracy of the DTA thermograms for both CsCl and benzoic acid (calibration standard) through an investigation of selected parameters that are known to exert influences on the characteristics of thermograms. These parameters consist of both sample and instrumental types, including (a) the amount of sample and method of compaction, (b) thermal characteristics of sample, (c) location of the thermocouple, (d) the heating rate, and (e) the furnace atmosphere. Of these, quantitative data were obtained only for the effect of heating rate on thermogram characteristics.

A preliminary estimate has been made for the process temperature factor (f_t) to be used in the equation presented in IDC Newsletter for July 1969, (ORNL CF-69-8-13):

$$F = Cf_c P^n N(I/\bar{I})$$

where

f_c = complexity factor = 2×10^m

$m = f_t$ (process temperature factor) + f_p (process pressure factor) + f_a (process alloy factor)

Other terms are defined in ORNL CF-69-8-13. This a continuation of the work to develop a generalized semiempirical method for evaluating various conceptual process designs for recovery of fission products.

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[10] A. F. Rupp, ORNL-2828, p. 13.

Process	Maximum Operating Temperature,[a] °C	f_t
1 (HCl as eluant)	350	0.06
2 (HNO_3 as eluant)	350 - 600	0.06 - 0.11
3 [$(NH_4)_2CO_3$-NH_4OH as eluant]	400	0.07

[a]This refers to the highest temperature encountered in the overall process. Except for Process 2, it is the temperature in the drying (or calcining) step.

The estimate of f_t for Process 2 depends on the thermal stability of $^{137}CsNO_3$ as well as the kinetics of reaction of $CsNO_3$ with HCl. Information on such characteristics for $CsNO_3$ is not available at present.

Studies on the equilibrium phase behavior of CsCl-KCl system will continue. An emphasis will be placed on the development of an efficient solid mixing technique for the CsCl-KCl mixtures. Preliminary DTA thermograms will be obtained for a few selected CsCl-KCl mixtures. The process economics study will be continued as time permits.

2. Cesium Chloride Slab Sources

Nothing to report.

RADIATION SOURCE DEVELOPMENT - 08-01-04

A. Strontium-90 Silicate Sources

1. Matrix Studies

Nothing to report.

2. Encapsulation

Nothing to report.

3. Characterization

A program to systematically determine pertinent data on strontium silicate was initiated with leach rate tests in which inactive strontium silicate minibeads were contacted with commonly used chemical reagents. The tests were carried out by stirring approximately 0.750 g of minibeads in 40 ml of the reagent for 2 hr, with the solution maintained at 25, 50, and 100°C. The quantity of material leached from the minibeads was determined by weighing the sample before and after exposure.

The experimental results are summarized in Table 2. Nitric acid at a concentration of 8 N gave complete reaction of the strontium silicate

Table 2. Results of Two-Hour Leach Rate Tests on Strontium Silicate Minibeads

Reagent	Concentration, N	Temperature, °C	Original Wt of Minibeads, mg	Weight Lost to Solution, mg	Percent Loss
Sulfuric acid	36.0	100	a	-	-
		50	665.5	6.2	0.93
		25	620	20.0	3.23
	18.0	100	a	-	-
		50	646	12.5	1.93
		25	684.0	15.1	2.21
	0.1	100	a	-	-
		50	686.7	33.2	4.83
		25	663.0	32.1	4.84
Nitric acid	16.0	100	664.4	Complete reaction	-
		50	731.0	172.0	23.5
		25	633.0	18.9	2.99
	8.0	100	775.4	766.0	98.79
		50	755.6	730.0	96.61
		25	750.2	378.0	50.39
	0.1	100	750.2	47.4	6.32
		50	764.9	41.0	5.36
		25	762.5	33.5	4.39
Ammonium hydroxide	7.5	100	761.6	26.5	3.48
		50	723.3	52.2	7.22
		25	741.4	5.1	0.69
	3.8	100	761.5	20.0	2.63
		50	763.3	4.7	0.62
		25	774.0	7.1	0.92
	0.1	100	791.7	2.5	0.32
		50	766.8	2.5	0.33
		25	761.5	7.5	0.98
Phosphoric acid	48.0	100	740.1	3.2	0.43
		50	758.0	41.0	5.41
		25	669.4	76.8	11.47
	24.0	100	741.0	43.0	5.80
		50	756.7	34.9	4.61
		25	736.8	12.3	1.67
	0.1	100	743.0	5.8	0.78
		50	760.0	11.1	1.46
		25	739.3	5.5	0.74
Acetic acid	17.5	100	743.8	1.0	0.13
		50	738.1	32.5	4.40
		25	741.4	2.9	0.39
	8.25	100	735.1	26.2	3.56
		50	741.1	21.2	2.86
		25	747.9	21.6	2.89
	0.1	100	752.0	20.2	2.69
		50	759.0	18.3	2.41
		25	745.5	22.6	3.03
Distilled water		100	754.5	41.4	5.49
		50	766.7	5.6	0.73
		25	769.4	18.3	2.38
Sea water		100	761.8	2.3	0.30
		50	768.7	0.9	0.12
		25	768.0	0.6	0.08
Sodium hydroxide	19.0	100	740.0	288.7	39.01
		50	735.5	5.7	0.77
	0.1	100	747.3	22.8	3.05
		50	751.5	5.6	0.75
		25	756.0	1.6	0.21
Oleic acid in ethyl alcohol	2.9	100	763.6	2.5	0.33
		50	766.2	9.7	1.27
		25	762.4	5.3	0.70
	1.5	100	764.5	5.0	0.65
		50	766.7	1.7	0.22
		25	768.1	1.6	0.21
	0.1	100	768.1	5.6	0.73
		50	766.4	0.8	0.10
		25	784.0	4.4	0.56

[a]Spilled sample.

in 1-1/2 hr at 100°C. Sodium hydroxide solution at concentrations of 19 N dissolved approximately 50% of the minibeads at 100°C. The attack of all other solutions on $SrSiO_3$ at all temperatures tested was imperceptible.

The main objective of these tests was to establish proper techniques for studying the leach rate of strontium-90 silicate minibeads in the hot cells.

Considerable interest has been expressed in the use of strontium-90 silicate minibeads as sources of beta radiation in a fluid bed reactor. Since the source particles are maintained in rapid motion, source attrition is a potential problem. Experiments are being conducted to study attrition by measuring the weight loss of a 30-g sample of inactive strontium silicate minibeads when fluidized with toluene in a 1-in.-ID glass tube with a bed expansion of 300%. A plot of the data is shown in Fig. 2. The rapid rate of

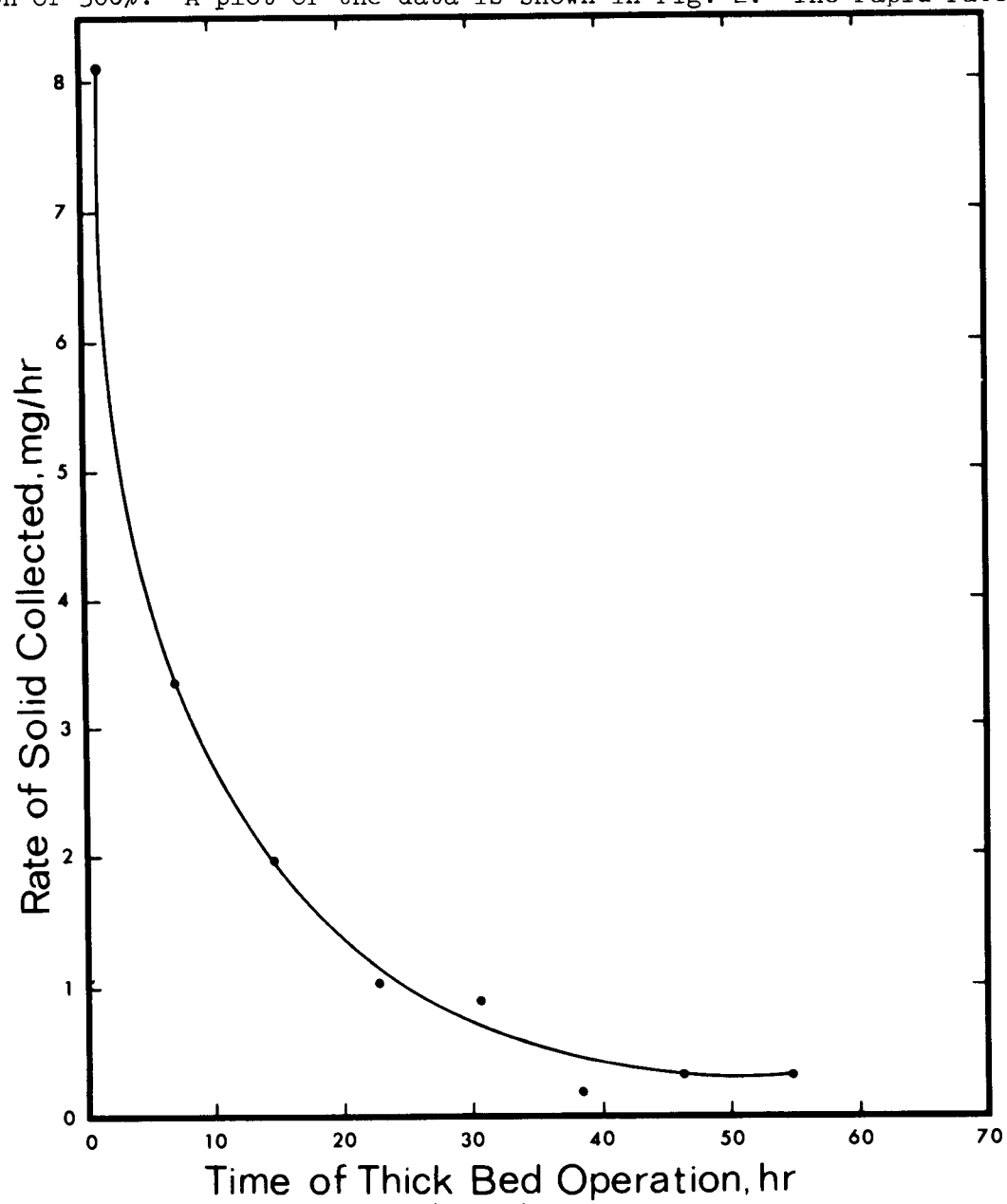

Fig. 2. Minibead Weight Loss (mg/hr) Versus Accumulated Time of Bed Operation.

sphere loss early in the experiment was due to removal of the smaller particles adhering to the original spheres. The level portion represents the rate at which attrition appears to be taking place but additional tests will be necessary. Additional experiments will be conducted in which graphite-coated minibeads will be tested.

A 5-g sample of strontium silicate minibeads was coated with pyrolytic carbon in an effort to increase the resistance of the minibeads to chemical attack and to investigate the possibility of using graphitic carbon as a matrix material. A thickness of approximately 0.00075 in. of carbon was deposited in 10 min at a temperature of 1050°C, propane flow rate of approximately 670 cm^3 per minute, and approximately 2170 cm^3 of helium per minute. The reaction vessel was constructed of graphite.

B. Fabrication of Carbon-14 and Technetium-99 Radiographic Sources

Nothing to report.

C. Alpha Source Fabrication Development

Nothing to report.

RADIOISOTOPE SOURCE SAFETY TESTING - 08-01-04

A. Special Form Prototype Testing

Nothing to report.

B. Testing Sources with Manufactured Leaks

Nothing to report.

C. Surveillance of Regulations and Cask Evaluations

Large quantity radioactive material packages are evaluated prior to application for AEC certification and petition for USA DOT approval. A special permit for the SNAP-7E thermoelectric generator was received from the Hazardous Materials Regulations Board of the Department of Transportation.

Two prototype casks for transporting curium and other transplutonium nuclides were completed by the shops and are being loaded with lithium hydroxide monohydrate, which will serve as neutron shielding as well as thermal accident protection.

The submission of the evaluation of the lightweight drum package to the AEC Oak Ridge Operations Office is being held up pending the outcome of their review of two existing package designs under consideration. Questions have been raised as to the effectiveness of a hydrated salt during normal conditions of transport and the hypothetical accident environment. Extensive additional evaluations of the two package designs were submitted to answer questions and to demonstrate package integrity. The review of regulations continues on a daily basis.

D. Source Leak Test Guide

Nothing to report.

E. Sealed Source Testing

Nothing to report.

F. Unsealed Source Testing

Testing of commerical tritium foils in argon at elevated temperatures was continued. Prior to testing, the foils were purged with dry air at 25°C for 16 hr to minimize differences due to humidity. In each test a section of foil was placed in a quartz tube which had been preheated to the desired temperature and an argon flow was used to sweep the released tritium into the recovery train to obtain a 2-hr sample. The results of these tests are given in Table 3.

Table 3. Tritium Loss at Elevated Temperatures[a]

Temperature, °C	Tritium Loss, % of initial activity		
	4 Ci/in.^{2} [b]	8 Ci/in.^{2} [b]	15 Ci/in.^{2} [b]
100	0.00096	0.00049	0.000035
150	0.0053	0.0022	0.00023
200	0.068	0.025	0.0038
300	1.1	0.39	0.078
400	5.3	2.31	0.62
500	c	c	c
600	70.0	48.7	c

[a]Over a 2-hr period in flowing argon.
[b]Nominal activity of foil.
[c]Results not yet available.

TECHNOLOGY UTILIZATION - 08-01-05

A. Information Center

The Isotopes Information Center now has approximately 17,000 accessioned items - 500 more than last month. During February, 258 IIC reports and reprints were sent out in answer to queries, and 39 requests that required specific answers were filled. A list of reviews in process (for answering of future queries) is as follows:

Title	Author(s)	Status, % Complete
Cement-Polymer Combinations	V. I. Solomatov, translated from Russian and edited by Martha Gerrard, P. S. Baker, and R. E. Greene	100 (draft)
DID Research and Developments - 1969	Isotopes Information Center Staff	90
Indium-113m: Preparation and Uses	Martha Gerrard	10
Flow Measurement	R. H. Lafferty, Jr.	5
Iodine-125	P. S. Baker and Martha Gerrard	25
Isotopic Methods of Examination and Authentication in Art and Archaeology	F. J. Miller, E. V. Sayre, and B. Keisch	95
Oceanography[a]	Isotopes Information Center Staff	90
Patent Review - Process Radiation Development	R. E. Greene, P. S. Baker and Helen Warren	80
Potato Sprout Inhibition by Radiation	F. E. McKinney	30
Radiation vs Glass[b]	D. N. Hess	20
Radioisotopes in the Pharmaceutical Industry	P. S. Baker and Martha Gerrard	5
Radioisotopes in the Steel Industry[c]	Republic Steel Corporation	10
Radioisotopes in X-Ray Fluorescence Analysis	R. H. Lafferty, Jr.	5
Self-Diffusion in Liquids	F. J. Miller	75
Strontium-90	Roberta Shor, R. H. Lafferty, Jr., and P. S. Baker	85
Technetium-99m: Preparation and Uses	Martha Gerrard and P. S. Baker	30
Radioisotopes in the Textile Industry	F. J. Miller	90
Use of Radioisotopes in Sedimentology	V. Romanovsky, translated and edited by Martha Gerrard	10
Wheat Disinfestation - A Worldwide Review	F. E. McKinney	5
Wood Plastics	R. E. Greene	75
Yttrium-90	Martha Gerrard	10
Permuted Isotopes and Radiation Technology Indexes	Helen Raaen	7

[a]Draft completed; being held at DID.
[b]Indefinite.
[c]Will be in sections; first section now in hand.

B. Isotopes and Radiation Technology

Galley proofing for Isotopes and Radiation Technology 7(4) was started and manuscript for 8(1) was about half completed.

C. Publications

P. S. Baker, A. F. Rupp, and Associates, Isotopes and Radiation Technology 7(3) (Spring 1970).

RADIOISOTOPE APPLICATIONS DEVELOPMENT

BASIC TECHNOLOGY DEVELOPMENT - 08-03-01

A. Liquid Scintillation Development

1. Alpha Radiation Damage in Phosphors

Nothing to report.

2. Cerenkov Counting

Nothing to report.

SYSTEMS ENGINEERING APPLICATIONS - 08-03-02

A. Oceanographic Systems Study

A field experiment was conducted at Oceanside, California. Operations began February 1, 1970, and were completed February 13, 1970. Two line injections and four point injections of gold-198-199 tagged sand were made (see Fig. 3). The first two injections were line sources: one inside the harbor and one outside the harbor near the breakwater. Five days later two point injections were made near the harbor mouth. Two days later two point injections to evaluate the core injection device were made near the beach inside the harbor. All injections indicated sand transport, and preliminary evaluation of the data indicates a successful experiment. The core injection device performed adequately and only slight modifications will be required. Stationary detectors were placed over the core injections and showed the decrease in activity as a function of time. Although time did not allow a detailed study of the core injections, sufficient information was obtained to prove the technique applicable to the problems of determining volume transport.

The electromechanical cable and the towing system (winch, termination block, etc.) worked well for approximately 10 days. An instrument signal problem then developed and it is not clear at this time what caused this problem. The backup detector system was used for the remainder of the experiment. We will determine if the problem involves the electromechanical cable as soon as it is returned to ORNL.

B. National Security

Nothing to report.

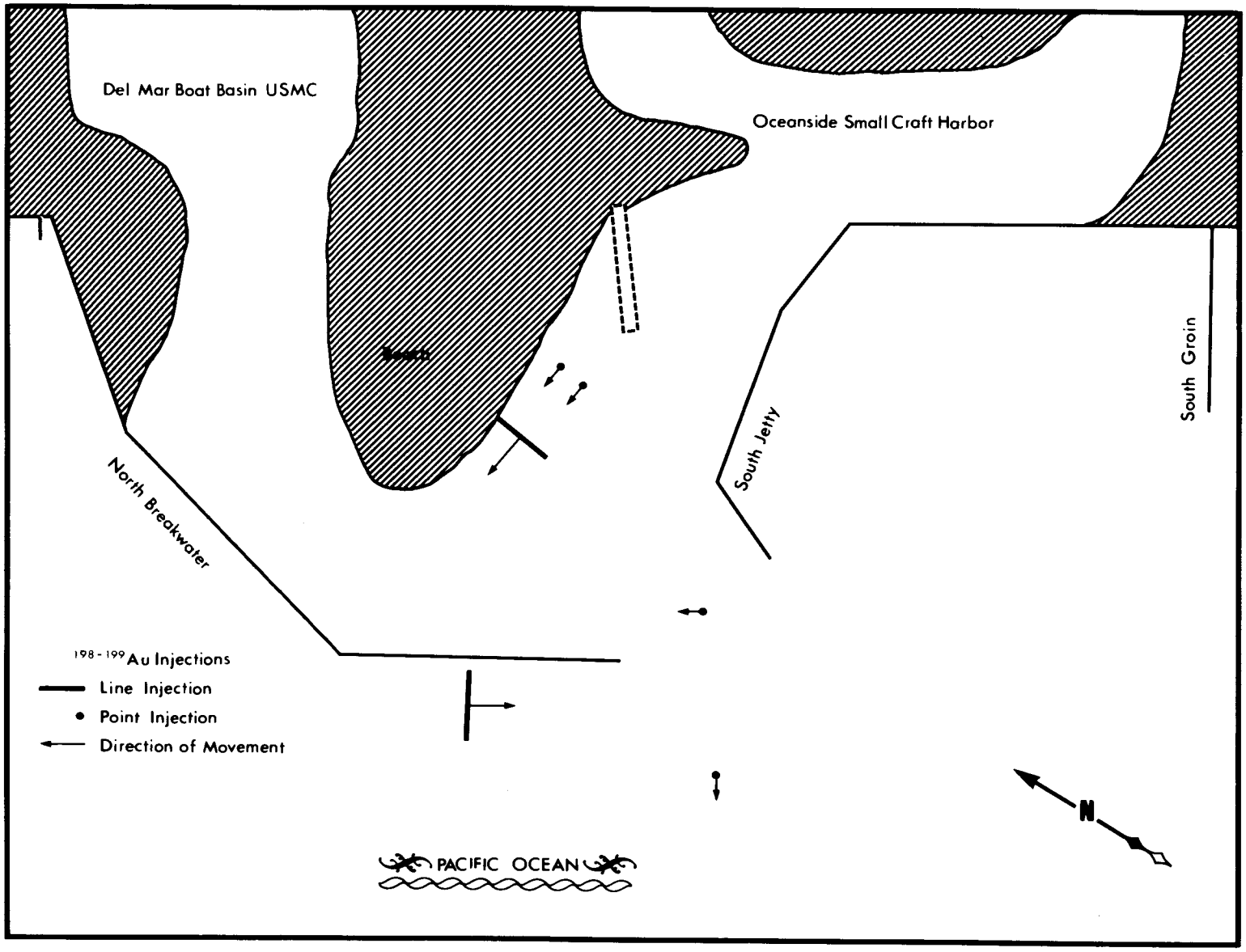

Fig. 3. Location of Tagged Sand Injections in February 1970 Experiment.

RADIATION PROCESSING APPLICATIONS DEVELOPMENT

ORGANIC SYNTHESIS WITH ISOTOPIC SOURCES - 08-04-01

A. Allyl Alcohol

Nothing to report.

B. Sucrose

Nothing to report.

C. Straight Chain Fatty Acids

A major part of February's efforts was spent writing reports on past work in radiolytic syntheses and radiolytic treatment of waste effluents. The remainder of the time was spent on the gamma radiolytic oxidation of 2-hexene. Experiments on 2-hexene under 1500-psig oxygen were repeated. Two-milliliter samples irradiated at 6.8×10^6 R/hr for 3 hr received a total absorbed dose of 20.2 Mrads. At standard temperature and pressure, gamma radiolytic oxidation of 2-hexene normally causes attack at the position alpha to the double bond; however, under high oxygen pressure 55.5% of the products were derived from oxidative fissure at the double bond yielding two and four carbon chain molecules.

The products have been tentatively identified by comparison of retention time on gas-liquid chromatography with authentic samples. The products, their abundance (percent by weight), and G value (number of molecules of product formed per 100 eV absorbed) are given below.

Products	Percent by Weight	G Value
Formaldehyde	10.1	161
Water	2.5	66
Methanol	9.0	134
Acetaldehyde	3.7	40
Ethanol	11.5	98
Formic Acid	4.5	47
Unknown A	2.1	-
Unknown B	1.1	-
Acetic Acid	8.8	70
2-Hexene (cis-trans)	14.3	-
n-Butyraldehyde	17.7	117
n-Butanol	0.7	5
n-Butyric Acid	13.1	71
Unknown C	0.9	-

The fraction which has been assigned as butyric acid has been confirmed by mass spectroscopy and infrared spectroscopy. The fraction designated as acetic acid has an acid function indicated by infrared analysis.

RADIOISOTOPE SALES

An inquiry was received from Radiochemical Centre for 30,000-50,000 Ci of strontium-90. McDonnell Douglas Astronautics Company advised that they were increasing their order for promethium-147 to 150,000 Ci. A request for quotation was received from Centre d'Etude de L'Energie Nucleaire, Belgium, for up to 10,000 Ci of strontium-90.

Shipments made during February included 116,129 Ci of cesium-137 (replacement) to Brookhaven National Laboratory; 10,001 Ci of tritium to Radium-Chemie; 180,260 Ci of strontium-90 to CEA, France, and 15,000 Ci of tritium to Radiochemical Centre.

A list of outstanding large orders is shown below:

Isotope	Amount, Ci	Customer
Cesium-137	360,000	Brookhaven National Laboratory[a]
Cesium-137	~180,000	American Nuclear Corporation[a]
Cesium-137	~207,000	U. S. Army, Natick Laboratories[a]
Cesium-137	205,000	Industrial Reactor Laboratories, Inc.
Cesium-137	120,000	Radiation Machinery Corporation
Cesium-137	~194,000	Atomic Energy of Canada Limited
Cesium-137	40,000	Radiation Resources, Inc.
Cesium-137	20,000	Radiochemical Centre
Cesium-137	200,001	CEA, France
Cesium-137	200,001	Radiation Resources, Inc.
Cesium-137	~5,000	Miscellaneous orders
Strontium-90	120,000	CEA, France
Strontium-90	~532,000	Isotopes, Inc.[a]
Strontium-90	~105,000	Isotopes, Inc.[a]
Strontium-90	~41,000	Brookhaven National Laboratory
Strontium-90	556,982	U. S. Navy
Strontium-90	~2,000	Miscellaneous orders
Promethium-147	5,100	Reactor Centrum Nederland
Promethium-147	5,000	Mitsubishi Corporation (Japan)
Promethium-147	150,001	McDonnell Douglas Astronautics Company[b]
Carbon-14	1,445 mCi	Catholic University
Tritium	75,000	Radiochemical Centre

[a]Activity has been billed in previous fiscal years. Material is now being fabricated into sources or has already been fabricated into sources and is stored for future requests for shipments.
[b]Material is to be shipped from Richland, Washington.

Tables 4 and 5 give the ORNL and non-ORNL cyclotron customers and runs for February 1970.

The radioisotope sales proceeds and shipments for the first seven months of fiscal years 1969 and 1970 are given in Table 6.

Table 4. Cyclotron Irradiations and Runs for ORNL Research Programs During February 1970

Customer	Product	Target	Time, hr:min Beam	Misc	Total	Total Charges
ORAU	Gallium-67	Zinc	1:00	1:00	2:00	$ 225.10
	Gallium-67	Zinc	1:00	1:15	2:15	248.85
	Gallium-67	Zinc	2:00	1:15	3:15	343.85
Physics	Cobalt-61	Nickel-64	2:00	1:15	3:15	318.75
	Cobalt-61	Nickel-64	2:00	:50	2:50	279.17
	Cobalt-61	Nickel-64	2:00	:50	2:50	279.17
Isotopes	Rubidium-84	Krypton-84	8:00	1:15	9:15	878.75
	Indium-111	Cadmium-111	2:00	1:15	3:15	318.75
	Indium-111	Cadmium	:30	1:15	1:45	176.25
	Promethium-143	Samarium-144	6:18	1:15	7:33	717.24
Stock	Gallium-67	Zinc	2:54		2:54	
Total			29:42	11:25	41:07	$3785.88

Table 5. Cyclotron Irradiations and Runs for Non-ORNL Customers During February 1970

Customer	Product	Target	Time, hr:min Beam	Misc	Total	Total Charges
New England Nuclear Corporation	Yttrium-87	Strontium-87	2:00	1:15	3:15	$ 540.10
	Rhodium-99	Ruthenium-99	4:00	1:15	5:15	820.10
	Cobalt-57	Nickel-58	50:00	1:15	51:15	8,388.64
	Yttrium-87	Strontium-87	2:00	1:15	3:15	540.10
	Rubidium-84	Krypton	2:30	:25	2:55	481.67
	Yttrium-87	Strontium-87	2:30	1:15	3:45	610.10
Abbott Laboratories	Cobalt-57	Nickel	11:55	:50	12:45	2,009.14[a]
Bell Telephone Laboratories	Europium-149	Samarium-150	16:00	1:15	17:15	2,433.45
Huntington Hospital	Rubidium-84	Krypton	8:00	:50	8:50	1,383.32
Stock	Cobalt-57	Nickel	18:05	:25	18:30	
Total			117:00	10:00	127:00	$17,206.62

[a]Purchased but not shipped.

Table 6. Radioisotope Sales and Shipments

Item	7-1-68 thru 1-31-69	7-1-69 thru 1-31-70
Inventory items	$295,292	$348,321
Major products	58,506	65,063
Radioisotope services	102,960	133,985
Cyclotron irradiations	99,817	80,519
Miscellaneous processed materials	39,621	43,169
Packing and shipping	48,004	42,945
Total Radioisotope Sales	$644,200	$714,002
Total Radioisotope Shipments	1,617	1,440

RADIOISOTOPE PRODUCTION

CESIUM-137 PROCESSING

No cesium-137 processing was done this month.

STRONTIUM-90 PROCESSING

One $^{90}SrTiO_3$ product was made using the NH_4OH-$(NH_4)_2CO_3$ precipitation procedure with a yield of approximately 50,000 Ci. Analysis and material balance calculations for the last three runs are currently in progress. Approximately 150,000 Ci of $^{90}SrTiO_3$ made by this procedure will be used to complete the bulk powder order for France (CEA).

MAINTENANCE

Decontamination of the manipulator cells used for cesium-137 processing continued on an intermittent basis. It is expected that personnel entry for decontamination purposes will be effected in March.

The installation of the off-gas dehumidifier system is approximately 90% complete. This equipment, after installation in the plant off-gas system, will reduce the plugging problems associated with the off-gas system HEPA filters.

Operations at the FPDL during February were reduced 50% to supply manpower for the curium-244 project.

SOURCE FABRICATION

A. Cesium-137

A shipment of 116,200 Ci of $^{137}CsCl$, contained in 30 rectangular doubly encapsulated sources, was made to Brookhaven National Laboratory. Fifteen of these sources were fully loaded; the other 15 were patterned sources of five different arrangements of fuel pellets and spacers. A miscellaneous $^{137}CsCl$ source run is in progress.

During the baking at 400°C of some cylindrical $^{137}CsCl$ pellets for the miscellaneous run, the pellets all increased in diameter by 2 to 3%. Pellets made from three different batches of $^{137}CsCl$, all of which were within acceptable purity specifications, were pressed and baked. All pellets exhibited approximately the same degree of swelling. Similar pellets were pressed from chemically pure, inert cesium chloride, and they also showed the same effect. The baking of pellets after pressing is a new procedure and is an outgrowth of the recent work with rectangular pellets. This is the first time that the procedure has been used for cylindrical pellets. Operationally, the only difficulty presented by this effect would occur during fabrication of sources requiring both high density pellets and very close clearances between pellet and capsule.

B. Strontium-90

Nine pellets of strontium-90 fuel materials were made for compatibility tests and thermal conductivity measurements. All of these were made by hot pressing. The pellets to be used for thermal conductivity measurements had thin platinum shims on either end; the others were pressed without additives. The pellet characteristics are given in Table 7.

Table 7. Physical Characteristics of Strontium-90 Pellets

Material	Pellet Diameter, in.	Activity Concentration, Ci/g	Power Density, W/cm^3	Power Output, W
$^{90}Sr_2TiO_4$[a]	1.00	40.4	1.10	6.81
	1.00	40.4	1.12	7.42
$^{90}SrSiO_3$[a]	1.00	31.8	0.66	4.50
$^{90}SrTiO_3$	0.50	28.7	0.81	1.31
	0.50	28.7	0.77	1.31
	0.50	28.5	0.76	1.45
	0.50	28.5	0.82	1.47
	0.50	28.7	0.85	1.37
	0.50	28.7	0.85	1.37

[a]Thermal conductivity pellets.

C. Miscellaneous Sources

Miscellaneous sources fabricated and shipped during January and February 1970 include: five carbon-14 sources (30 mCi), two americium-241 sources (<1 Ci), one cobalt-60 source (<1 Ci), one gadolinium-153 source (<1 Ci), two cobalt-56 sources (<1 Ci), one promethium-147 source (25 Ci), and 65 strontium-90 sources (10 Ci).

SPECIAL FORM MATERIALS

One shipment of 180,000 Ci of $^{90}SrTiO_3$ powder in 13 welded cans was made to France (CEA) and another shipment of 120,000 Ci of $^{90}SrTiO_3$ in eight cans was prepared for shipment to the same customer. Twenty-five $^{137}CsCl$ powder shipping cans, ranging in content from 10 to 500 Ci, were welded and put into storage for delivery to customers.

In a continuation of the investigation of the effect of carbonate in $^{90}SrTiO_3$ bulk powder a sample of 12,500 Ci of $^{90}SrTiO_3$ powder (433 g - comparable to the largest bulk packages) was placed in a sealed chamber and the pressure was observed. The $^{90}SrTiO_3$ used contained the highest carbonate contamination detected in any batch - 1.3 wt %. The chamber pressure was stabilized at approximately 5.5 psig after being allowed to come to thermal equilibrium; then, the pressure was recorded for three weeks. Except for minor fluctuations due to changing thermal conditions (approximately ±0.5 psig), no pressurization was noted.

OPERATIONAL SUMMARY

	February 1970		FY 1970	
	Number	Amount, Ci	Number	Amount, Ci
Cesium-137				
HAPO shipments received	0	0	0	0
Product batches prepared	0	0	0	0
Sources fabricated	0	0	185	305,400
Special form containers	25	7,700	25	7,700
Strontium-90				
HAPO shipments received	0	0	0	0
Product batches prepared	1	50,000	14	560,000
Sources fabricated	0	0	0	0
Special form containers	8	120,000	40	332,900

PROCESS STATUS

Physical Inventory

Item	Cesium-137, Ci	Strontium-90, Ci
In-process materials	243,500	1,042,000
Product powder	425,300	856,000
Sources in fabrication	10,400	0
Completed sources[a]	345,800	648,000

[a]Includes bulk shipping cans.

MISCELLANEOUS RADIOISOTOPE PROCESSING

Short-Lived Fission Products			Processed Units			Service Irradiations	
Radioisotope	No. of Batches	Amount, Ci	Radioisotopes	No. of Batches	Amount, Ci	Type	No. of Irradiations
Xenon-133	3	1,800	Copper-67	2	0.041	Zinc-68	1
Iodine-131	1	50	Silver-111	1	0.150	Y_2O_3	1
Yttrium-91	1	6	Rubidium-86	1	6.400	Vanadium	1
Strontium-89	1	47	Rhenium-186	1	1.770	Tellurium-130	1
Barium-140	1	2	Iodine-131[a]	6	0.450	Uranium-236	1
^{95}Zr-^{95}Nb	1	b	Barium-131	1	0.018	Platinum-196	1
Neodymium-147	1	b	Calcium-47	1	0.012	Antimony	1
Praseodymium-143	1	b	Tritium	1	20,981	Gold	1
			He/^{3}H	3	359	Chromium	1
	10	1,905	Xenon-133	2	178		
			Argon-37	1	<1		9
			Rubidium-85	c	<1		
				22	21,528.841		

[a]Elemental.
[b]Awaiting analytical results.
[c]Two cyclotron targets.

Work is continuing on the cerium-144 separation.

ADMINISTRATIVE SECTION

Travel by IDC personnel and visitors to the IDC are given in Tables 8 and 9.

Table 8. Travel by IDC Personnel

Site Visited	Purpose of Visit
Vienna, Austria[a]	Attend International Nuclear Information Systems meeting
Washington, D. C.	Discuss cesium encapsulation with AEC-DID
	Discuss Isotopes Kilowatt Program plans with AEC-RDT
	Discuss RIST program with AEC-DID

[a]Personally paid for.

Table 9. Visitors to IDC

Visitors (affiliation)	Purpose of Visit
U. S. Navy	Strontium-90 sources
General Dynamics	Strontium-90 sources
Sandia Corporation, Monsanto Research Corporation, LASL, AEC-SNS, AEC-Dayton, and Donald W. Douglas Laboratories	Isotope Fuels and Materials Committee meeting
University of Miami	Cyclotron production

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EXTERNAL DISTRIBUTION

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62. T. D. Chikalla, PNL, Richland, Washington
63. D. F. Cope, AEC Site Representative, ORNL
64. D. C. Davis, AEC, Oak Ridge, Tennessee
65. J. C. Dempsey, AEC, Washington, D. C.
66. W. K. Eister, AEC, Washington, D. C.
67. E. E. Fowler, AEC, Washington, D. C.
68. J. D. Goldstein, AEC, Washington, D. C.
69. A. Gottschalk, Argonne Cancer Research Hospital, Chicago, Illinois
70. F. D. Haines, AEC, Washington, D. C.
71. J. W. Irvine, MIT (Consultant)
72. J. Lawrence, Lawrence Radiation Laboratory, Berkeley, California
73. J. E. Machurek, AEC, Washington, D. C.
74. J. N. Maddox, AEC, Washington, D. C.
75. J. C. Malaro, AEC, Washington, D. C.
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