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RADIOISOTOPE PROGRAM (8000)

PROGRESS REPORT FOR JUNE 1972

A. F. Rupp

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ORNL-TM-3905

Contract No. W-7405-eng-26

ISOTOPES DEVELOPMENT CENTER

RADIOISOTOPE PROGRAM (8000) PROGRESS REPORT FOR JUNE 1972

A. F. Rupp

Work Sponsored by
AEC Division of Applied Technology

JULY 1972

[redacted]

OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee 37830
operated by
UNION CARBIDE CORPORATION
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CONTENTS

Page

RADIOISOTOPE PRODUCTION AND MATERIALS DEVELOPMENT 1

REACTOR-PRODUCED ISOTOPES 1

A. Biomedical Radioisotopes 1

1. Phosphorus-33 1
2. Potassium-43 1
3. Platinum-195m 2

B. Gadolinium-153 2

C. Reactor Products Pilot Production 3

ACCELERATOR-PRODUCED ISOTOPES 3

A. Biomedical Radioisotopes 3

1. Gallium-67 3

B. Cyclotron Products Pilot Production 4

FISSION PRODUCTS 5

A. Krypton-85 Enrichment 5

B. Cesium-137 Pilot Production 5

1. Processing and Process Status 5
2. Operational Summary 6
3. Current Orders 6

C. Strontium-90 Pilot Production 6

1. Processing and Process Status 6
2. Operational Summary 7
3. Current Orders 8
4. Source Fabrication 8
5. URIPS-8 Generators 8

D. Short-Lived Fission Products Pilot Production 8

E. Promethium-147 Source Fabrication,
Powder Shipments, and Current Orders 8

F. Fission Xenon Purification 9

SOURCE DEVELOPMENT 9

A. Cobalt-57 9

B. Americium-241/Gold Alloy 9

C. Aluminum Foil Krypton-85 Sources 11

D. Radioisotope Safety 11

1. Tritium Loss from Luminous Wrist Watches 11

E. Radioisotope Characterization, Quality Control, and Standards 11

1. Radioisotope Characterization 11
2. Radioisotope Special Analysis and Quality Control . . 12

F. Work for Others 13

RADIOISOTOPE SYSTEMS DEVELOPMENT 13

OCEANOGRAPHIC AND NATURAL RESOURCES DEVELOPMENT 13

A. Sand Transport Study 13

RADIOISOTOPE SALES 13

ITEMS OF INTEREST 13

ADMINISTRATIVE 15

PUBLICATIONS 17

JOURNALS 17

REPORTS 17

TRANSLATION 17

RADIOISOTOPE PROGRAM (8000) PROGRESS REPORT FOR JUNE 1972

A. F. Rupp

RADIOISOTOPE PRODUCTION AND MATERIALS DEVELOPMENT

REACTOR-PRODUCED ISOTOPES - 08-01-01

A. Biomedical Radioisotopes

1. Phosphorus-33

The purpose of this project is to develop methods of preparing hundred-millicurie quantities of carrier-free phosphorus-33 containing <5% phosphorus-32. The two methods for producing phosphorus-33 being evaluated are based on the irradiation of highly enriched targets of sulfur-33 (>92 at. %) or chlorine-36 (approximately 63 at. %) in a fast neutron flux. Phosphorus-33 (25.2 days; 0.248-MeV β^-_{max}) has both a longer half-life and a lower energy beta than phosphorus-32 (14.3 days; 1.709-MeV β^-_{max}), which makes it advantageous for autoradiography, longer ecological and agricultural experiments than with phosphorus-32, synthesis of tagged complex organophosphorus compounds, and double labeling experiments.

A 1.075-g 66% enriched $K^{36}Cl$ target was processed after irradiation for four cycles in the HFIR. The yield of phosphorus-33 was 1.2 Ci, but the phosphorus-32 content was 9%; therefore, a decay period of 25 days will be required. The only gamma impurity detected was <0.01% chromium-51.

The $K^{36}Cl$ was recovered for reirradiation; radiocontaminants cesium-134 and cesium-136 were removed. Two ampuls containing 52% enriched $K^{36}Cl$ were also included when the 66% material was canned for irradiation; these will be used as a backup source of phosphorus-33 while they are undergoing further enrichment of chlorine-36 and burnup of chlorine-35.

2. Potassium-43

The objectives of this project are: to prepare potassium-43 by the $^{43}Ca(n,p)^{43}K$ reaction, using isotopically enriched ^{43}CaO targets, in quantities sufficient for medical and biological experiments; to define a method for separating potassium-43 from the target in a purity suitable for medical use; and to establish cooperative programs with medical institutions interested in evaluating its usefulness. Potassium-43, with a half-life of 22.5 hr and gamma-ray

emissions of 0.373 and 0.617 MeV, is potentially useful for metabolic and clinical studies of blood flow, rejection of transplanted organs, and kidney function because the dose rate would permit multiple doses.

Three batches of potassium-43 were processed this month, and 60 mCi of potassium-43 was shipped to users who are primarily interested in clinical testing of potassium-43 as an alternative noninvasive technique for heart study. Over the past fiscal year we have supplied over 250 mCi of potassium-43 on a cost-recovery basis.

3. Platinum-195m

The objectives of this work are to develop methods for preparing platinum isotopes suitable for whole-body scanning in biological and medical experiments, to characterize the products, and to establish cooperative programs with medical research groups interested in evaluating their usefulness.

Recent medical research has shown that certain platinum compounds act as chemical therapeutic agents towards certain tumors. There is need for platinum radioisotopes as tools for investigating the therapeutic mechanism involved. Platinum-195m (4.1 d) has gamma emissions of 99 keV (11%) and 129 keV (2.8%) that make it suitable for whole-body scanning.

Thirty millicuries of platinum-195m was sent to the University of Southern California under our medical cooperative program. Dr. Wolfe reports that they have obtained good yields of the cis-dichlorodiammine Pt(II) compound and that it localizes strongly in the liver and spleen but is rapidly eliminated by the mice under study. National Institutes of Health has supplied mice with specific tumors for use in the USC study.

B. Gadolinium-153

Gadolinium-153 (242 days) offers potential as a useful and important radionuclide due to its favorable physical decay characteristics, e.g., it decays solely by electron capture to stable europium-153 with 97- and 103-keV γ-rays and europium x-rays as the predominant accompanying photon emissions. This energy range is useful for various types of gaging and transmission bone-scanning applications. The neutron burned-out europium control plates from HFIR offer a by-product source of highly enriched gadolinium-152 that would be prohibitively expensive to match by calutron separations.

To meet continuing requests for gadolinium-153, preparations are being made to process and recover several grams of gadolinium-152 target material from a HFIR control rod. Our Li-Hg amalgam process previously described will be used for the separation. The in-cell equipment consists of separate anode and cathode electrolysis cells separated by a sintered glass membrane. By separating the two compartments we have improved the efficiency of the separation due to the elimination of lithium carbonate in the cathode area where the gadolinium/europium separation occurs.

C. Reactor Products Pilot Production
(Production and Inventory Accounts)

Processed Units		Service Irradiations	
Radioisotope	Amount (mCi)	Type	Number
Copper-67	12	Platinum-196	2
Calcium-47	5		

ACCELERATOR-PRODUCED ISOTOPES - 08-01-02

A. Biomedical Radioisotopes

1. Gallium-67

The objectives of this program are to determine the optimal target configuration for gallium-67 (78.2 hr) production by the $^{68}Zn(p,2n)^{67}Ga$ reaction 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.

Gallium-67 decays by electron capture with the emission of four main gamma rays of 93, 185, 300, and 394 keV with intensities of 42, 24, 17, and 5%, respectively.

Five 100-mCi batches of pharmaceutical-grade gallium-67 citrate were supplied to ORAU as part of the project of the Cooperative Group to Study Localization of Radiopharmaceuticals. Additional fractions were packaged as either the citrate or the chloride for sale to other medical researchers.

Two alternative electrodeposited zinc-68 target designs which have been developed (1) to increase target life and integrity and (2) to reduce the "buildup" of zinc-65 (243.7 d) contamination in the enriched zinc-68 target material have been test irradiated and evaluated for their applicability to large-scale gallium-67 production. The previous design of zinc electrodeposition directly onto copper was found unacceptable due

to extensive diffusion of these two elements to form intermetallics during irradiation which shortened target life and resulted in a loss of target integrity with resulting problems of loose contamination. In one alternative target design tested, high-purity silver was used to replace the copper as the electrodeposition substrate material. This design met the requirements of increased target life and integrity, but its development was discontinued when it was found that the formation of small amounts of AgCl during dissolution in the HCl system used for the gallium-67—zinc separation carried a significant fraction of the carrier-free gallium-67 present. This resulted in very poor (<50%) chemical yields for the combined target dissolution and processing. Another important disadvantage of this target design is that the use of silver for the substrate material would probably make the use of this technology prohibitively expensive for small compact cyclotron producers who must use an elaborately shaped, internal copper target for large-scale production.

A second alternative target, of a different design that would be of wider interest, has also been developed and is currently under extensive irradiation testing. This design involves the use of a few tenths of a mil nickel plate as a "diffusion barrier" between the zinc target material and copper substrate material. Preliminary irradiation tests have shown no signs of copper diffusion through the nickel barrier into the zinc, as exemplified by an absence of zinc-62 [which would be formed in the copper substrate via the $^{63}Cu(p,2n)^{62}Zn$ reaction] in short-cooled dissolver solutions, and no signs of a loss of target integrity upon extended irradiation; e.g., a total integrated beam >2 mA. Additional development work and testing of this target design will be pursued during the next quarter. The successful development of this target design should open the door for the use of high-current, internal beams available in present compact cyclotrons for the large-scale production of gallium-67.

B. Cyclotron Products Pilot Production (Production and Inventory Accounts)

June 1972 ORNL 86-Inch Cyclotron runs for ORNL and non-ORNL programs are given in Table 1.

Table 1. ORNL 86-Inch Cyclotron Runs for June 1972

Product	No. of Runs	Time (hr:min)	Total Charges
		ORNL Programs	
Cobalt-61	2	6:05	$ 591
Gallium-67	4	20:50	2,187
Indium-111	4	16:30	1,695
Rubidium-83	1	3:50	384

Table 1. Continued

Product	No. of Runs	Time (hr:min)	Total Charges
Thulium-168	1	8:15	$ 797
Zinc-62	1	2:00	197
		57:30	$ 5,852
	Non-ORNL Programs		
Cobalt-57	2	100:30	$16,817

FISSION PRODUCTS - 08-01-03

A. Krypton-85 Enrichment

No gas transfers were made this month. The three thermal diffusion units loaded with krypton are in normal operation, and the count rate in the center section of each unit is continuing to build up.

A routine inspection and maintenance of the cooling tower was done without shutting down the separation units.

B. Cesium-137 Pilot Production (Production and Inventory Accounts)

1. Processing and Process Status

A cesium oxalate crystallization campaign was made, and calcination of the oxalate to cesium carbonate intermediate product was started on a part-time basis. All of the cesium oxalate will be converted to carbonate before cesium chloride final product is made. The oxalate filtrates were scavenged in the mainstream crystallizers; then the crystallizer system was shut down. The current cesium-137 process status is shown below.

Item	Cesium-137 (Ci)
In-process material	875,000
^{137}CsCl products	51,700[a]
Sources in fabrication	0
Completed sources awaiting shipment	182,000[b]

[a] Includes special form stock cans and old sources.

[b] Includes 113,600 Ci in special form cans.

2. Operational Summary

Item	June 1972 No.	June 1972 Ci	CY 1972 No.	CY 1972 Ci	FY 1972 No.	FY 1972 Ci
HAPO shipments received	0	0	0	0	1	414,200
Product batches prepared	0	0	1	7,000	19	195,000
Sources						
Fabricated	0	0	23	55,300	55	105,300
Shipped	0	0	23	55,300	39	73,900
Special form cans						
Fabricated	0	0	10	113,000	23	114,500
Shipped	2	100	5	900	17	2,400

3. Current Orders

Current orders for cesium-137 as sources or bulk powder are shown below:

Customer	Amount (Ci)	Estimated Shipping Date
Brookhaven National Laboratory	203,000	FY 1973
Lockheed Georgia Company	∿35,000	a
Radiochemical Centre, England	400	July 1972
Technical Operations, Inc.	8	July 1972

[a]Sources are in storage awaiting receipt of customer's container.

An order is on hand from Atomic Energy of Canada Limited for ∿113,000 Ci of cesium-137 as CsCl powder. The powder has been canned and stored awaiting release by customer.

C. Strontium-90 Pilot Production (Production and Inventory Accounts)

1. Processing and Process Status

An improved model vacuum hot-press that was installed in May was used to press the pellets for use in four 368-W (nominal) strontium-90 sources for the U.S. Navy. This hot-press is similar in design to the vacuum hot-presses used previously for both strontium and curium pelletizations, but it incorporates some improvements to facilitate remote handling. It is also large enough to accept the same die assembly that is used in the induction-heated press. The die assembly consists of a solid graphite die with punches, plus an outer sleeve made from spun graphite material; the loaded assembly is approximately 6-1/4 in. OD by 12 in. tall. Pellets with diameters as large as 10 cm can be pressed. Four 6.7-cm-diam pellets were pressed while the hot-press was being checked out and pressing conditions were being established. Then eight pellets were pressed and used to load

the four sources. In each case, two pellets were loaded into a stainless steel liner. The loaded liners were inserted into the capsules and made ready for welding.

All of the fuel used was nominally in the distrontium titanate form; the batches used ranged in power concentration from 0.275 to 0.290 W/g. The average pellet density was 4.55 g/cm^3 (range 4.30 to 4.65), and the average power density was 1.23 W/cm^3 (range 1.18 to 1.31). Each pellet contained two nickel reinforcement screens. The loaded capsules were assayed calorimetrically before welding as a check against the wattages calculated from calorimetric assay of samples of fuel. The loading data for the four inner containers are shown below.

Inner Container	Fuel Wt (g)	Nickel Wt (g)	Watts Calculated	Watts Actual
25	1,350	20	370	384
26	1,338	32	373	374
27	1,340	32	368	377
28	1,279	19	364	368
				1,503 (221,000 Ci)

No additional strontium-90 work is planned for the near future. The current strontium-90 process status is shown below.

Item	Strontium-90 (Ci)
In-process material	805,900
Strontium-90 products	426,200
Sources in fabrication	0
Returned SNAP sources	446,000
Completed sources awaiting shipment	221,000

2. Operational Summary

Item	June 1972 No.	June 1972 Ci	CY 1972 No.	CY 1972 Ci	FY 1972 No.	FY 1972 Ci
HAPO shipments received	0	0	0	0	1	660,500
Product batches prepared	0	0	13	510,000	16	626,000
Sources						
Fabricated	4	220,000	9	580,000	10	629,000
Shipped	0	0	5	360,000	6,	409,000
Special form cans						
Fabricated	0	0	39	5,600	39	5,600
Shipped	12	360	13	900	14	1,000

3. Current Orders

Current orders for strontium-90 as sources or bulk powder are shown below:

Customer	Amount (Ci)	Estimated Shipping Date
U.S. Navy (ORNL)	208,000	FY 1973
U.S. Navy	278,675	a

[a]To be released by customer.

4. Source Fabrication

Our latest strontium-90 "stock" powder campaign was completed. Thirty-nine capsules of various curie content were fabricated. Twelve of these capsules have been shipped to customers.

Four strontium-90 sources, each containing a nominal 55,000 Ci (368 W), were sealed by plasma-arc welding, and calorimetry of these sources was completed. These sources are being leak tested; this project will be completed in July with the loading of the thermoelectric generators.

5. URIPS-8 Generators

The URIPS-8 sources have been loaded and installed in the shields. Pumping down and purging the insulation systems has begun. The generators should be installed during the first two weeks of July.

D. Short-Lived Fission Products Pilot Production (Production and Inventory Accounts)

Isotope	Number of Batches	Amount (Ci)
Xenon-133	2	∿700
Iodine-131	1	1
Ruthenium-103	1	38

E. Promethium-147 Source Fabrication, Powder Shipments, and Current Orders

Donald W. Douglas Laboratories' order for 101,000 Ci of promethium-147 will be made in two shipments from Battelle Northwest Laboratory. Fifty thousand curies will be shipped in July, and the remaining 51,000 Ci will be shipped in November 1972.

F. Fission Xenon Purification

Approximately 900 liters of fission xenon was processed for removal of krypton-85. The krypton-85 present in the product was below the limits of routine analysis (<0.8 μCi/liter). Improvements in methods of analysis will be sought in order to establish the limit of krypton-85 decontamination — possibly an analytical chromatographic separation method.

SOURCE DEVELOPMENT - 08-01-01

A. Cobalt-57

In our studies of fabrication methods and radiation output measurements from low-energy gamma sources, cyclotron-produced cobalt-57 was prepared by ion exchange to provide a solution that contained less than 5 mg of nickel per 100 mCi of cobalt-57. This solution was analyzed by measuring the ∿123-keV energy peak on a multichannel analyzer and aliquots containing 4 to 150 mCi were measured in a gross ionization chamber as a method for standardizing the chamber for use in measuring fabricated sources for gamma output. Three sources of different cobalt-57 loadings were prepared with 40-mil-thick stainless steel windows, and the input versus the output was measured. The results are shown in Table 2.

Table 2. Input Versus Output from Cobalt-57 Encapsulated in 40-mil Stainless Steel

Activity in Source (mCi)	Output Measured (mCi)	Output (%)
0.9696	0.752	77.5
9.696	7.16	73.8
157.1	123.1	76.0

The percentage output when calculated using established absorption coefficients for iron and 125-keV photons indicates the output of the sources should be 79-80% of the activity contained in the capsules. The experimental results are in good agreement with this theoretical output; therefore, the measurement can be used in the evaluation of various source designs.

B. Americium-241/Gold Alloy

Americium metal prepared by reduction of AmO_2 with thorium metal was alloyed with gold by levitation melting in vacuum of the two components. Mixing of the components was assured by eddy currents induced by the radiofrequencies used to heat the mass.

The alloy contained 7.5% americium by weight and was cold-rolled into foil form having a weight of ∿100 mg/cm^2. After ultrasonic cleaning of the foils, they were tested for resistance to smearing of the americium-241 activity by positioning them under a 10-g weight with a 1- by 10-cm strip of filter paper placed between the weight and the foil. The weight was constrained with a ring from moving as the paper was pulled over the americium/gold alloy foil. This assembly provided a uniform pressure to the smear tests of ∿10 g/cm^2.

Activity transferred to the filter paper averaged 0.006 μCi for 10 smears taken as described. The alloy was then tested for loss of activity by leaching in 20 ml each of water, normal saline solution, and 0.01 *N* HCl. The results of this leach test are shown in Table 3.

Table 3. Total Activity Leached from 2-cm^2 Surface Area of 7.5%/92.5% Americium/Gold Alloy (μCi)

Test Medium	Hours of Exposure			
	120	168	240	960
Water	0.025	0.051	0.033	0.118
Saline Solution	0.005	0.111	0.0025	0.0015
0.01 *N* HCl	79.5	110	181	500

Since the level of activity in the water and normal saline leach solutions did not show a linear increase with time, it was suspected that activity was being removed from the solution by precipitation or by adsorption on the walls of the test containers. The foils were removed from the solutions, and the leach solvents were acidified with HCl. After this treatment the water contained 4.5 μCi of americium-241 and the saline solution contained 0.50 μCi of americium-241 indicating that both water and normal saline does cause some leaching of activity from the alloy. Tests for extended time periods are in progress.

The conclusions at this time are:

1. Although activity apparently can be rubbed from the alloy tested, millicurie quantities of americium-241 can be handled without the hazard associated with dry powder.

2. A nonradioactive barrier such as cold-bonded gold foil is necessary to achieve a practicable smear-free source.

3. The alloy is reasonably stable and should provide considerable second line protection in the case of source capsule failure.

C. Aluminum Foil Krypton-85 Sources

Of three ion-bombarded foils on long-term test to determine the retention of krypton-85, two made at 4000 V show 92% of the initial activity remaining after 45 days, and one made at 500 V was down to 45% of the original activity after the same time.

A series of foils was made at 3000 V to study the effect of various bombardment times upon the activity implanted in the metal. Depleted krypton containing 1% krypton-85 was used in this study. A 30-sec bombardment yielded an average activity measured with a cutie pie at contact of 7.3 $mR/hr/cm^2$, but a bombardment for 6 min gave less than twice this value, or 12.1 $mR/hr/cm^2$. The foils made under apparently identical conditions have been rather inconsistent, with activities varying by a factor of 2.

D. Radioisotope Safety

1. Tritium Loss from Luminous Wrist Watches

Determination of the amounts of tritium released from commercial wrist watches at room temperature (75°F) and at body temperature (98.6°F) was completed. The tritium is incorporated as the activator in self-luminous paint applied to the dials and hands. The loss rates are given in Table 4 for the two brands of watches tested.

Table 4. Tritium Loss from Wrist Watches (μCi/day/watch)

Sample No.	Room Temperature	Body Temperature
W-H-274	0.105	0.238
W-H-275	0.020	0.058

E. Radioisotope Characterization, Quality Control, and Standards

1. Radioisotope Characterization

For gallium-67, unpublished relative gamma intensities were obtained from National Bureau of Standards personnel. The intensities agree adequately with the local values.[1] The half-life, 78.2 hr, was also supported. With respect to gamma-photon percentages, used in radioassay, our selections

[1]A. F. Rupp, *Radioisotope Program (8000) Progress Report for March 1971*, ORNL-TM-3380, Oak Ridge National Laboratory.

were confirmed within about 8% by approximate data "buried" in a paper,[2] for example, our estimate was 17% for the 300-keV gamma, against their 18.4%.

A report containing a large table of gamma energies and intensities was reviewed for the *Health Physics* journal. Though the data were reasonably good, issuance of the report demonstrates again the proliferation of such projects and the need for unification of effort.

An inquiry was answered regarding the chemical states of gold radioisotope products and the likelihood of gold-195 in fallout (probably undetectable).

2. Radioisotope Special Analysis and Quality Control

Two gamma ionization chambers used for assay of some products were checked electrically, a potentiometer was added to one to improve the precision of its readings, and they were calibrated with a local set of cobalt-60 standards. When an NBS standard was measured, a discrepancy of 3% was found, significantly above the estimated errors in the respective calibrations. A sample of cobalt-60 was sent to NBS for their calibration.

In the assay of carbon-14, anomalous results were obtained by the "standard-addition" method, in which liquid scintillation counting of a sample mixture is first done, followed by addition of a portion of an organic standard from NBS, and a recount to determine efficiency. Results were high, compared to the accepted method, which uses a separate standard mixture, and by comparison with mass spectrometry. Although no explanation has been found, the phenomenon seems to be related to that reported by Moghissi[3] and, privately, by representatives of two reputable radioactivity laboratories. The standard-addition method was therefore abandoned. Tests of various aspects of carbon-14 analysis were made in the course of the investigation. Accuracy of weighing and dilution were confirmed, stability of the counting mixture was tested, effects of components were studied, linearity of counter response was verified, and crude spectra of various samples were compared. Additional tests will be done when NBS issues a new carbon-14 standard later this year.

A meeting of radioactivity-standards producers was held June 5, sponsored by NBS and the Atomic Industrial Forum. By invitation, we presented a report on users' attitudes and needs, based on our survey in 1970. It was decided to form a task group of manufacturers to attempt to provide methodology and establish NBS traceability. On June 6, a meeting of the NRC standards subcommittee was held, covering decay-scheme findings and needs, including compilations, recent NBS activities, and other items. At both meetings, we suggested that weightless deposits of carbon-14 on metal disks should be provided as counter test sources and that ORNL might undertake their preparation if a need is found to exist, and if no other

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[2] W. H. Zoller, G. E. Gordon, and W. B. Walters, *Nuclear Phys.* A137, 606 (1969).

[3] A. A. Moghissi and M. W. Carter, *Anal. Chem.* 40, 812 (1968).

organization is willing or able to provide them. One participant questioned the need, which will be established by correspondence.

F. Work for Others

Unbleached kraft softwood wash effluent, caustic extraction stage effluent, chlorination stage effluent from mixed hardwood and softwood bleaching, and untreated effluent from sodium-base neutral sulfite process were studied for decolorization and COD reduction by radiation alone and with oxygen at 500 psi.

In all of the effluent samples studied, the addition of oxygen during irradiation caused significant improvement in both color and COD.

The test results obtained with unbleached softwood waste are typical of all of the effluents tested. In this case, the COD was reduced from 1115 to 1045 and the color units were reduced from 162 to 157 when tested without oxygen to a radiation dose of 10^7 R. With the same radiation dose to the effluent under 500 psi oxygen, the COD was reduced form 1115 to 480 and the color units were reduced from 162 to 10.

RADIOISOTOPE SYSTEMS DEVELOPMENT

OCEANOGRAPHIC AND NATURAL RESOURCES DEVELOPMENT - 08-03-02

A. Sand Transport Study

Radionuclides appear to be useful tools to study sand transport phenomena and, although many experiments have been conducted to demonstrate their effectiveness as tracers, little is known concerning whether or not the dynamic systems in which they are used can be tagged well enough to achieve quantitative data leading to an understanding of basic mechanisms. The Radioisotope Sand Tracing (RIST) study has progressed through equipment development to the point where important system tagging parameters can be studied.

The simplified RIST Survey System to be transferred to the Coastal Engineering Research Center is being interfaced with a computer provided by them. The unit will be placed in service by the Corps of Engineers for routine survey of sediment transport.

RADIOISOTOPE SALES

ITEMS OF INTEREST

An order was received from the U.S. Navy for 278,675 Ci of strontium-90 as strontium titanate. A request for quotation was received from Isotope

Products Laboratories for aluminum foils containing krypton-85 prepared by ion implantation. A request for quotation was received from Atomic Energy of Canada Limited for twelve cesium-137 sources, each containing 1800 Ci. A request for quotation was received from Metronex, Warsaw, Poland, for 75,000 Ci of strontium-90 as disk pellets.

Shipments of interest during the month are listed below:

Customer	Isotope	Amount
Large Quantities		
New England Nuclear Corporation	Tritium	4,000 Ci
Radiochemical Centre, England	Promethium-147	1,340 Ci
Radiochemical Centre, England	Strontium-90	260 Ci
Radiochemical Centre, England	Cesium-137	123 Ci
Argonne National Laboratory	Technetium-99	40 g
Items Shipped for Use in Cooperative Programs		
Oak Ridge Associated Universities	Gallium-67	
Vanderbilt University	Xenon-133	
University of Southern California	Platinum-195m	
Specially Approved Withdrawn Items		
University of North Carolina	Irradiation of platinum-196	2 shipments
Mayo Clinic	Copper-67	1 lot
University of Chicago	Copper-67	1 lot

A summary of radioisotopes shipped during May is given in Table 5.

Table 5. Summary of Radioisotopes Shipped During May 1972

Radioisotope	Number of Shipments	Quantity (mCi)
Argon-37	6	11
Calcium-47	11	8
Cesium-137	5	6,560
Cobalt-60	1	25
Iodine-129	2	800 mg
Iodine-131	4	535
Krypton-85	9	323,500
Niobium-95	4	18
Phosphorus-33	3	300
Promethium-147	7	52,050
Samarium-151 oxide	1	15 mg
Strontium-90	20	8,642
Tritium	17	27,970,000

Table 5. Continued

Radioisotope	Number of Shipments	Quantity (mCi)
Xenon-133	62	64,410
Zirconium-95	4	23
Carbon-14	1	
Experimental products	49	
Service irradiations		
Cyclotron	4	
Reactor	2	
Targets	3	
	215	28,426,082
		815 mg

The radioisotope sales proceeds and shipments for the first eleven months of FY 1971 and FY 1972 are given in Table 6.

Table 6. Radioisotope Sales and Shipments

Item	7-1-70 thru 5-31-71	7-1-71 thru 5-31-72
Inventory items	$ 504,354	$ 477,504
Major products	109,973	57,862
Radioisotope services	241,800	278,402
Cyclotron irradiations	112,062	117,782
Miscellaneous processed materials	59,700	56,336
Packing and shipping	78,750	65,738
Total	$1,106,639	$1,053,624
Number of radioisotope shipments	2,854	2,219

ADMINISTRATIVE

Travel by IDC personnel and visitors to the IDC are given in Tables 7 and 8.

Table 7. Travel of IDC Personnel

Site Visited	Purpose of Visit
Philadelphia, Pa.	Attend seminar on fine-powder measurement technology
AEC-DAT Washington, D.C.	Program discussions

Table 7. Continued

Site Visited	Purpose of Visit
Union Camp Corporation Savannah, Georgia	Discuss tall oil bleaching by radiation
Brookhaven National Laboratory New York	Discuss coordination of programs and ORNL use of BLIP for target irradiation
Washington, D.C.	Participate in meeting of NRC Standards Subcommitte and NBS-AIF meeting
Las Vegas	Attend ANS meeting and ASTM E-10 Subcommittee 5

Table 8. IDC Visitors

Visitors (affiliation)	Subject Discussed
Atlantic Richfield Hanford Company Richland, Washington	Cesium-137 and strontium-90 source fabrication
Institut für Strahlen und Kernphysik Der Universitat Bonn West Germany	Writing paper on strontium-84 (ORIC), planning of experiment on strontium-88 (ORIC), and production of tin-119m
Eastern Mennonite College Harrisonburg, Virginia	Member of group planning experiment on germanium-67
Louisiana State University Baton Rouge, Louisiana	Molecular plating of rare earths and production of dysprosium-166
Vanderbilt University Nashville, Tennessee	Preparation of europium-150 by p- and α-particle bombardment and cooperative programs
UNISOR Project Oak Ridge, Tennessee	Ion sources for mass separator cooperative programs
Vanderbilt University Nashville, Tennessee	Writing holmium-166m paper
Division of Nuclear Medicine Vanderbilt University Nashville, Tennessee	Lecture and tour of IDC facilities

PUBLICATIONS

JOURNALS

J. H. Hamilton, M. Fujioka, J. J. Pinajian, and D. J. McMillan, "Spin and Parity Assignments and Levels in ^{156}Gd Populated by the Decay of ^{156}Tb," *Phys. Rev.* C5, 1800-1806 (1972).

REPORTS

E. Lamb, *IPF Monthly Status Report for April 1972*, ORNL-TM-3835, Oak Ridge National Laboratory.

E. Lamb, *IPF Monthly Status Report for May 1972*, ORNL-TM-3861, Oak Ridge National Laboratory.

A. F. Rupp, *Radioisotope Program (8000) Progress Report for May 1972*, ORNL-TM-3873, Oak Ridge National Laboratory.

TRANSLATION

J. Bastian and K. H. Lieser, "Rapid Separation of ^{137m}Ba from a ^{137}Cs-loaded Column," *Radiochimica Acta*, 6, 216 (1966), translated by J. J. Pinajian.

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