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RADIOISOTOPE PROGRAM (8000) PROGRESS REPORT FOR MAY 1972

A. F. Rupp

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ISOTOPES DEVELOPMENT CENTER

RADIOISOTOPE PROGRAM (8000) PROGRESS REPORT FOR MAY 1972

A. F. Rupp

Work Sponsored by
AEC Division of Applied Technology

JUNE 1972

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Oak Ridge, Tennessee 37830
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RADIOISOTOPE PROGRAM (8000) PROGRESS REPORT FOR MAY 1972

A. F. Rupp

RADIOISOTOPE PRODUCTION AND MATERIALS DEVELOPMENT

REACTOR-PRODUCED ISOTOPES - 08-01-01

A. Biomedical Radioisotopes

1. 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.

Potassium-43 was shipped to recipients including University of Mississippi Medical Center, Johns Hopkins Medical Institutes, University Hospitals of Cleveland, V. A. Center, Wood, Wisconsin, and National Institutes of Health.

Shipments were made in multidose bottles with diaphragm tops which are preferred by the users because they can better recover the radioisotope by inverting the bottle and penetrating the top with a hypodermic needle. Although we do not claim sterility, the bottles are evacuated, autoclaved, and then filled by a hypodermic needle attached to a micropore filter. This method should further reduce chance of contamination, although not one shipment has failed sterility or pyrogen tests.

Isotopic analysis of the 49% enriched calcium target showed essentially the same isotopic abundances as determined a year ago, confirming that reprocessing the target does not introduce normal calcium.

2. 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 (12%) and 129 keV (2.8%) that make it suitable for whole-body scanning.

The half-life of platinum-195m was determined over a period exceeding four half-lives, and a value of 4.020 ± 0.009 days was obtained as the average of the decay of the 99- and 129-keV photon peaks. The quoted error is one standard deviation of the mean. The observed photons and their relative intensities are compared below with those of Schöneberg *et al.*[1] The relative intensities are believed to be within 5% accuracy.

E_γ (keV)	Relative Gamma Intensity	
	Present Work	Schöneberg *et al.*[1]
30.89 ± 0.02	17.1	—
98.90 ± 0.02	100	100
129.79 ± 0.02	25.6	25.5
140.6	0.263	0.21
211.6	0.341	0.32
239.6	0.477	0.56

Platinum-195m was shipped under cooperative programs to Michigan State University and University of Southern California. The target was not sealed in quartz but was placed in a small aluminum can with a press cap. This eliminated neptunium-239 which in previous batches was produced in the quartz ampul from natural uranium and leached out by the aqua regia used to dissolve the platinum.

The chloroplatinic acid preparation showed radiocontaminants similar to those reported in the February report,[2] except for elimination of neptunium-239 and the presence of some sodium-24. The yield of platinum-195m in a 6-day irradiation suggests a capture cross-section of ∿10 mb, rather than 90 mb.

B. Gadolinium-153

Gadolinium-153 (242 days) offers potential as a useful and important radionuclide due to its favorable physical decay characteristics. It decays solely by electron capture to stable europium-153 with 97- and 103-keV

---

[1] R. Schöneberg, D. Gföller, and A. Flammersfeld, *Z. Physik* 203, 453 (1967).
[2] A. F. Rupp, *Radioisotope Program (8000) Progress Report for February 1972*, ORNL-TM-3750, Oak Ridge National Laboratory.

gamma 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 provide a source of highly enriched by-product gadolinium-152 target that would be prohibitively expensive to match by calutron separations.

Previously we separated gadolinium and europium by continuous electrolysis of europium into a mercury—lithium mercury amalgam mixture.[3] To increase the process efficiency, we tried a chemical separation based on the chemical differences between the Gd(III)-Eu(II) oxidation states to see if an initial Gd-Eu separation could be made to reduce the amount of material that must be handled in a final separation. Electrolysis or high-pressure, large-scale ion-exchange chromatography and solvent extraction were tried. The work of Peppard *et al.*[4] which utilizes acidic phosphoric esters in aromatic diluents to extract the trivalent rare earths, but not the bivalent ones, has been extended successfully to process scale concentrations. Tracer experiments indicate that the system offers great potential for a rapid, efficient separation of Gd-Eu. The initial Eu:Gd ratio of approximately 100:1 can probably be reduced to approximately 1:1 with $\leq$10% loss of gadolinium. Tests are being made to determine the effect, if any, of radiolysis products that will be formed from the high-level radiation in full-scale separations.

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

Processed Units		Service Irradiations	
Radioisotope	Amount (mCi)	Type	Number
Copper-67	38	Platinum-196	2
Copper-64	41		
Calcium-47	13		

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

[3] A. F. Rupp, *Radioisotope Program (8000) Progress Report for August 1970*, ORNL-TM-3133, Oak Ridge National Laboratory.

[4] D. F. Peppard, E. P. Howitz, and G. W. Mason, "Comparative Liquid-Liquid Extraction Behavior of Eu(II) and Eu(III)," *J. Inorg. Nucl. Chem.* 24, 429 (1962).

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.

Four shipments of 100-mCi batches of pharmaceutical-grade gallium-67 citrate were supplied to ORAU during this period 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.

Development work on a high yield electrodeposited zinc-on-copper target has been directed to improving target life and integrity and reducing the rate of increase of zinc-65 (243.7 days) contamination of the enriched zinc-68 target material. As mentioned last month, all electrodeposited zinc-on-copper targets that have been irradiated with 22-MeV protons in the ORNL 86-Inch Cyclotron to high beam exposures, e.g., ≥ 2 mA-hr, have shown discoloration in the region of maximum beam intensity and have given some contamination problems because of loose material upon removal from the machine. In addition, appreciable amounts of the enriched target material have been lost, with the amount lost apparently increasing with increases in the integrated beam current. It is known that the intermetallic alloys formed upon Cu-Zn interdiffusion are brittle, have much lower thermal conductivities, and undergo volume changes upon formation. All these factors would be detrimental to target life and integrity. Alloy layer growth can be minimized by irradiating at lower temperatures (i.e., lower beam currents) for longer times to reach a required integrated beam current rather than at higher currents for shorter periods of time.

Two approaches other than optimization of the irradiation conditions that might improve target integrity (zinc-65 would still diffuse into the enriched zinc-68 target material) are also being investigated. One is to use a few tenths of a mil thick nickel plate between the copper and zinc to act as a diffusion barrier, while the other is to use silver tubes, rather than copper, for the plating substrate. Targets have been prepared that will be test irradiated during the next month. The problems of target integrity and zinc-65 contamination would seem to be of greater concern than target life, for if these two problems cannot be surmounted, production via this target design will be severely limited.

B. Medically Useful Radioisotopes of Bismuth

The purpose of this program is to prepare high radiochemical purity radioisotopes of bismuth in high yield, establish a definitive decay scheme, measure the production yield, prepare a suitable radiopharmaceutical product, and test the product.

Several medical research groups have demonstrated that this isotope is better for brain tumor localization than other nuclides such as ^{43}K, ^{84}Rb, ^{52}Mn, ^{74}As, ^{110}Ag, ^{68}Ga, ^{65}Zn, ^{95}Zr, ^{131}I, and various compounds labelled with ^{131}I, ^{110}Ag, and ^{51}Cr. Bismuth-206 has a very high tumor/brain concentration ratio and has, as well, a low concentration in muscle and blood. Other bismuth radioisotopes are also potentially very useful because of the availability of half-lives varying from 28 years for bismuth-207 to 1.8 hr for bismuth-201.

1. Bismuth-205

Work on the decay scheme of 15.3-day bismuth-205 has been completed. In addition to the gamma-ray singles work with Ge(Li) detectors, the Compton suppression work, and the Ge(Li)-Ge(Li) coincident experiments, conversion electron data were taken with a Si(Li) detector. From these data, 146 transitions were identified in the decay of bismuth-205, and 130 were placed in a level scheme which includes 26 definite and 1 tentatively assigned level. Several previously suggested levels were confirmed and one was rejected. Two new levels were established. The work, submitted to *Physical Reviews* for publication, included K-conversion coefficients for 43 transitions.

2. Bismuth-206

Work on the study of the decay scheme of 6.24-day bismuth-206 has been completed. The work included gamma-ray singles, Compton suppression spectra, Ge(Li)-Ge(Li) coincidence measurements, and electron conversion spectroscopy. From this, 66 gamma rays were identified in the decay of bismuth-206 and 61 of these were placed in a decay scheme with 19 excited states (including 4 new levels).

With the completion of the bismuth-205 and bismuth-206 work we now have, in each case, a good knowledge of the total number of gamma rays emitted per 100 disintegrations, the K-conversion coefficients of the major transitions, and a picture of the decay scheme. These parameters are all necessary for the determination of yield, calibration of product, and estimation of radiation dose.

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

May 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 May 1972

Product	No. of Runs	Time (hr:min)	Total Charges
		ORNL Programs	
Cobalt-57	1	51:15	$ 4,876
Gallium-67	4	21:45	2,094
Indium-111	4	18:45	1,909
		91:45	$ 8,879
		Non-ORNL Programs	
Cobalt-57	2	100:30	$16,923
Germanium-68	1	9:15	1,533
		109:45	$18,456

FISSION PRODUCTS - 08-01-03

A. Krypton-85 Enrichment

The product from units A and B was transferred into CD, and the ends from units A and B were transferred into C and D. These manipulations increased the curie loading of units C, CD, and D and will permit them to reach higher product concentrations. Separation units A and B are now shut down due to lack of feed material.

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

1. Processing and Process Status

Oxalate crystallizations are in progress with an expected yield of about 100,000 Ci; chloride conversions should begin early in June.

The current cesium-137 process status is shown below. These values have been revised in accord with the annual physical inventory (as of May 3, 1972). A detailed list is given to show the actual status of the various materials.

Item	Cesium-137 (Ci)
In-process material	
Stored feed	494,000
In process	295,000
Residues from cesium-137 glass process	87,000
Total in-process material	876,000

Item	Cesium-137 (Ci)
Product-grade material	
^{137}CsCl powder unencapsulated	13,300
Miscellaneous special form cans	6,300
Old cesium-137 sources (assorted)	32,200
Holding for customers	182,000[a]
Total product-grade material	233,800

[a]Includes 113,600 Ci for AECL; 30,200 Ci for Lockheed; and 38,200 Ci for Radiation Resources.

2. Operational Summary

Item	May 1972		CY 1972		FY 1972	
	No.	Ci	No.	Ci	No.	Ci
HAPO shipments received	0	0	0	0	1	414,200
Product 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	113,900	23	114,500
Shipped	0	0	3	800	16	2,300

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	500	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 about 113,000 Ci of cesium-137 as CsCl powder. The powder has been canned and stored awaiting release by the customer.

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

1. Processing and Process Status

Strontium-90 processing was confined to the calcination of a 7000-Ci batch of strontium carbonate. All strontium-90 fuel material was inventoried and temporarily stored while cell decontamination operations were in progress.

The current strontium-90 process status is shown below, with the values revised in accord with the annual physical inventory (May 3, 1972).

Item	Strontium-90 (Ci)
In-process material	
Stored feed	730,500
"Recovered material"	75,400
Total in-process material	805,900
Product-grade material	
$^{90}Sr_2TiO_4$ powder	374,000
$^{90}SrTiO_3$ (from Quehanna operation)	118,200
Returned SNAP sources	446,600
Liners to be defueled	148,700
Miscellaneous stock cans	6,700
Total product-grade material	1,094,200

2. Operational Summary

Item	May 1972		CY 1972		FY 1972	
	No.	Ci	No.	Ci	No.	Ci
HAPO shipments received	0	0	0	0	1	660,500
Product prepared	0	0	13	510,000	16	626,000
Sources						
Fabricated	0	0	5	360,000	6	409,600
Shipped	0	0	5	360,000	6	409,600
Special form cans						
Fabricated	39	5600	39	5,600	39	5,600
Shipped	0	0	1	500	2	600

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	June 1972
Lovelace Foundation	100	June 1972
University of California, Berkeley	10	June 1972
Radiochemical Centre, England	250	June 1972

4. Source Fabrication

A strontium-90 "stock" powder campaign is now underway and should be completed during the first week of June.

D. Melting Point of Promethium

A highly accurate value for the melting point of promethium metal has not been reported in the literature. We have made for a research project a very pure sample of promethium metal and a portion of it was used in an existing apparatus to determine the melting point of promethium.

A basically visual observation method was used. Thermocouples were calibrated by melting silver and gold. An optical pyrometer was used to observe the melting process. The promethium sample was allowed to come to thermal equilibrium at 950°C and the temperature was then increased at 0.5°C/min until melting was observed. The melting point of the promethium was found to be 1105 ± 5°C, which compares with an experimental value[5] of 1168°C and estimated value[6] of 1035°C. Further determinations on different samples will be made.

E. Encapsulation of Source Compounds During Hot-Pressing

The simultaneous densification and encapsulation of source compounds are desirable for several reasons: (1) the number of operations are reduced; (2) the source compound is protected from chemical attack by atmospheric gases (desirable for both Cm_2O_3 and SrO); (3) better heat transfer from the source to the jacket is obtained, and (4) the encapsulated source is denser because there are no voids between it and the capsule walls. Preliminary studies of methods to encapsulate a strontium source compound during hot-pressing indicated that a method which uses nickel felt was best. This method is being investigated further.

In four encapsulations, 350 g of strontium orthotitanate was cold-pressed at 4000 psi in a 2.3-in.-diam steel die. The ejected compact was then placed inside a nickel felt container, which was formed inside a 1/8-in. wall ATJ graphite die liner. All seams in the nickel felt were overlapped to prevent escape of the strontium material. Details of the nickel felt container are shown in Fig. 1. ATJ graphite disks, 1/8 in. thick, were placed between die punches and the source-jacket assembly. The source and jacket were prepressed at 2000 psi and hot-pressed at 1250°C for 30 min. Upon removal from the die, the compacted sources adhered to the ATJ liner and end disks. The graphite was then burned off at 650-700°C for 10 to 12 hr, leaving a lightly oxidized surface on the nickel container. This oxide layer was easily removed by immersion in warm HNO_3 solution without apparent damage to the sources.

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[5]J. H. Jarrett and H. H. Van Tuyl, *Promethium Isotopic Power Data Sheets*, BNWL-1309, Battelle Northwest Laboratory, May 1970.
[6]K. A. Gschneidner, Jr., *Rare Earth Alloys*, Van Nostrand, New York, 1961.

Two attempts were made to encapsulate pellets which were previously hot-pressed. One involved a pellet containing 350 g of Sr_2TiO_4 hot-pressed at 1100°C to a density of 53%. This pellet was subsequently encapsulated, as previously described, by hot-pressing at 1250°C. Although the container was apparently leak free, the density of the Sr_2TiO_4 in the pellet was only 70%. The second attempt involved encapsulation of a 350-g Sr_2TiO_4 pellet hot-pressed to 98% density. This pellet was encapsulated on both ends, but no densification of the nickel on the side wall was evident.

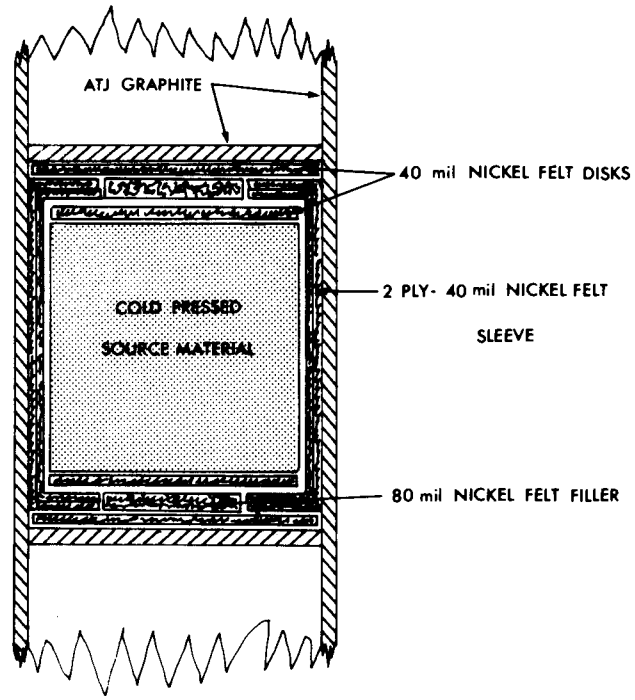

Fig. 1. Source and Jacket Assembly Before Hot-Pressing.

In all cases, except where source material was previously hot-pressed, well formed cylindrical pellets were obtained with good densities. Examination of sectioned pellets with a microscope showed the metal jacket to be continuous metal phase and 1/16-in. average thickness. Several encased pellets were cleaned in HNO_3. Densities and dimensions are shown below:

Pellet Number	Source Material	Diameter (in.)	Height (in.)	Weight (g)	Volume (cm^3)	Source Density (% theoretical)
1	350 g Sr_2TiO_4 (calcined powder)	2.510	1.079	479.5	87.45	4.80 = 94
2	350 g Sr_2TiO_4 (calcined powder)	2.508	1.109	487.5	89.74	4.71 = 92
3	350 g Sr_2TiO_4 (calcined powder)	2.483	1.141	484.4	90.54	4.64 = 91
4	450 g Sr_2TiO_4 (hot-pressed at 1100°C)	2.485	1.786	588.5	141.88	3.56 = 70
5	350 g Sr_2TiO_4 (calcined powder)	not measured — one end pitted by overheating				
6	350 g Sr_2TiO_4 (hot-pressed at 1480°C)	not measured — walls not densified				

Pellet 3 was thermally shocked three times by placing it in a furnace at 650°C for 1 hr, removing it, and dropping it into cold water. During the first quenching, fracturing of the source was audible; however, very little dimensional change was evident. After the second quenching, cracks

in the ends of the nickel jacket were noted. These cracks were opened completely during the third cycle due to reaction of water with the source material. Dimensional changes after each quenching are as follows:

Quenching	Diameter (in.)	Height (in.)
1st	2.496 + 0.013	1.145 + 0.0045
2nd	2.507 + 0.024	1.194 + 0.0525
3rd	2.516 + 0.033	1.251 + 0.1095

Simultaneous densification and encapsulation of strontium sources appear to be feasible using this technique. The additional equipment and operation required to cold-press source material would be partially offset by the removal of the requirement of fabricating and welding the currently used inner container. Sources prepared by this method could be easily decontaminated for loading into outer containers. This method may also allow a convenient operation for the fabrication of SrO sources by the use of $SrCO_3$ as a starting material since the SrO would not be exposed to atmospheric moisture.

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

Isotope	Number of Batches	Amount (Ci)
Xenon-133	2	700
Iodine-131	1	7
Niobium-95	1	4

G. 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. Fifty thousand curies will be shipped in July, and the remaining 51,000 will be shipped in November 1972.

SOURCE DEVELOPMENT - 08-01-04

A. Strontium-90 Beta Sources for Brookhaven National Laboratory

Two large flat plate strontium-90 beta sources (20 by 50 cm active area) were shipped to BNL for use in biological experiments. These sources, prepared by uniformly impregnating porous graphite with $^{90}Sr(NO_3)_2$ followed by decomposition of the nitrate to oxide, were encapsulated in aluminum, shelf-tested for leakage, and found to smear <0.005 μCi. Output measurements were made using thermoluminescent dosimeters which were calibrated against an extrapolation chamber. The beta dose rate at 10 cm from the source window was determined to be 1500 rads/hr.

B. Aluminum Foil Krypton-85 Sources

Two aluminum foil disks 3/4 in. in diameter were bombarded at 4.0 kV with krypton containing 1% krypton-85. During the first week following implantation the quantities of activity in the two sources fell to 93.5% of the initial count rates which were 43,000 and 52,000 counts/min and have held constant at these values for 18 days. The activity of a foil made earlier with 500-V ions fell to 60% of the initial count rate in 18 days. These observations would indicate that bombardment with 4-kV ions implants the krypton deeply enough in the aluminum foil to produce a fairly stable source.

Another foil was bombarded with krypton ions at 3.8-kV energy and reached a radiation level of 25 mR/hr at ∿2 in. The same foil was again placed in the apparatus and bombarded for 5 min with air ions at 3-kV energy and the radiation level dropped to 22 mR/hr.

C. Krypton-85 Source for NaK Flow Measurement

A neutrally buoyant krypton-85 source for measuring the flow rate of NaK in a magnetically pumped liquid metal loop has been designed. The source is 0.514 in. in OD by 0.700 in. long with 0.010-in.-thick walls and contains 50 mCi of krypton-85. The source will be used in an experimental MHD power generation loop by the Reactor Division at ORNL.

D. Radioisotope Safety

1. Tritium Loss from Self-Luminous Paints

a. Activity Concentration of the Paint Pigments

The study to determine the loss of tritium from commercial tritium-activated phosphor paints under various conditions was continued.

Since compliance with USAEC regulations (CFR 10, Part 32, ¶32.14) presumes knowledge of the initial amount of activity contained in the paint, the activity concentration per gram of unmixed pigment (phosphor coated with tritiated polymer) was determined for each of the eleven paints we have purchased and compared with the values supplied by the manufacturers (Table 2). The reason for the large differences is not clear. Repeated checks of the ORNL data gave results within ±10% of the average values shown in Table 2. It may be that some of the manufacturers are basing their stated activity concentration on the activity per gram of tritiated polymer rather than on the activity per gram of pigment (tritiated polymer plus ZnS phosphor).

Table 2. Tritium Concentration in the Paint Pigments

Sample No.	Manufacturers' Stated Concentration (mCi/g)	ORNL Analysis[a] (mCi/g)	Ratio of Manufacturers' Stated Concentration/ ORNL Analysis
W-H-285	100[b]	63	1.59
W-H-280	150	125	1.20
W-H-276	200	186	1.08
W-H-286	200[b]	139	1.44
W-H-283	222	105	2.11
W-H-277	240	198	1.21
W-H-281	300	135	2.22
W-H-284	370	107	3.46
W-H-278	450	317	1.42
W-H-282	500	305	1.64
W-H-279	550	390	1.41

[a]Average of either two or three determinations.
[b]Not stated by the manufacturer, but assumed from the manufacturers' sales information, which lists only these two activity concentrations available.

Additional tests are planned to determine the weight of tritiated polymer in the pigments so that the activity concentrations of the polymers can be calculated.

b. Water Leach Tests of the Paint Pigments

The stability of two of the pigments in water was determined by subjecting ∿60 mg of each pigment to a 24-hr water leach test. The results of these tests are given in Table 3.

Table 3. Tritium Losses in Water Leach Test

Sample No.	Manufacturers' Stated Tritium Concentration (mCi/g)	Tritium Loss (μCi)	Tritium Loss (% of initial)[a]
W-H-276	200	74.8	0.645
W-H-281	300	87.8	0.496

[a]Based on manufacturers' stated tritium concentration.

c. Tritium Loss from Paint Pigments in Air

Tritium losses from 2.00-g samples of three of the paint pigments were measured in air at room temperature. The results of these tests follow. The measured losses varied from 94 to 99% tritium oxide.

Pigment W-H-279 (550 mCi/g)

After purging with room air for 20 hr, the average loss rate during a 4-hr sample was 21.9 μCi/hr. A second sample, taken 4.5 days after the start of the test, gave an average loss rate of 19.8 μCi/hr.

Pigment W-H-281 (300 mCi/g)

Before starting the room air purge on this pigment, a sample was taken to determine the initial loss rate from the pigment immediately after it was removed from the storage bottle. The average measured loss rate over a 4-hr period was 18.0 μCi/hr. A second sample taken 4 days after the start of the test gave an average loss rate of 13.4 μCi/hr during a 2-hr sample period. A third sample taken 13 days after the start of the test gave an average loss rate of 7.8 μCi/hr during a 2-hr sampling period.

Pigment W-H-276 (200 mCi/g)

This pigment was also sampled for tritium loss before starting the room air purge. The average loss rate over a 4-hr sample period was 157 μCi/hr. The second sample, taken 5 days after the start of the test, gave an average loss rate of 2.7 μCi/hr during a 2-hr sampling period. The third sample, taken 13 days after the start of the test, gave an average loss rate of 2.4 μCi/hr over a sample period of 2 hr.

d. Tritium Loss from Freshly Mixed Paint During Drying

The tritium loss from freshly mixed paint at about 25°C was determined by mixing equal weights of pigment and binder, according to the manufacturers' instructions. The results are given in Table 4 for the two paints which were tested. Ninety-nine to one hundred percent of the tritium lost was as the oxide.

Table 4. Tritium Loss from Freshly Mixed Paint During Drying

Sample No.	Tritium Loss (% of initial)[a]			Manufacturers' Stated Activity (mCi/g)
	1st 2 hr	2nd 2 hr	3rd 2 hr	
W-H-276	0.12	0.021	0.013	200
W-H-281	0.082	0.043	0.017	300

[a]Based on manufacturers' stated activity concentration.

e. Tritium Loss from Paint Which Had Been Cured Three Months

The loss of tritium fron one paint sample which had been aged (cured) for 3 months after mixing according to the manufacturers' instructions was measured. The initial 9-hr sample gave an average loss rate of

0.125 μCi/hr (0.011% of the initial activity).* The second 9-hr sample, taken 24 hr after the start of the test gave an average loss rate of 0.0569 μCi/hr (0.0051% of the initial activity).* The third 9-hr sample, taken 120 hr after the start of the test gave an average loss rate of 0.0299 μCi/hr (0.00296% of the initial activity).*

E. Radioisotope Characterization, Quality Control, and Standards

1. Radioisotope Characterization

In response to an inquiry, the thermal cross section and resonance integral of iron-58 were estimated as 1.13 ± 0.02 and 1.2 barns, respectively. The cross section of cobalt-59 was given as 37.3 barns, and its resonance integral as 75 barns, while the effective cross section of cobalt-60 was stated to be 2 barns, all in accord with ASTM adoptions. Another inquiry was concerned with the specific activity of berkelium-249.

2. Radioisotope Special Analysis and Quality Control

The usefulness of cadmium-109 as a standard has been enhanced by an experimental finding[7] of 3.79 ± 0.07% as the intensity of the 88-keV gamma radiation, in excellent agreement with our value of 3.8 ± 0.1%. The following have been adopted for gamma intensities for platinum radionuclides: platinum-195m, 99 keV, 11.1% and platinum-197, 191 keV, 6.5%. These quantities are important in analysis of platinum-195m preparations.

At the request of a representative of the Atomic Industrial Forum, a list of commercial suppliers of radioactivity standards was furnished to them. S. A. Reynolds participated in the Executive Conference on Nuclear Standards at Monterey, California, May 1-2, as well as the meeting of ANS-ICONS advisors held at the same location. Included were discussions with the Chairman of ANS-16, the standards committee of ANS-IRD, and the Director of the NBS Center for Radiation Research.

F. Work for Others

Approximately 150 gallons of chicken processing plant waste was processed through the continuous flow charcoal column irradiator. Chemical pretreatment was used to reduce the COD to the lowest possible level prior to irradiation. This chemical treatment removed 95% of the COD, and flowing the solution through the column provided an overall removal of 98% of the COD at a residence time of 20 min in the charcoal column. Since the gamma dose to the solution flowing through the column is only approximately 1.5×10^5 rads, little reduction in COD can be expected due to radiation-induced oxidation. Some benefit was observed in elimination of a yellow color and removal of residual odor in the effluent that was not removed by charcoal alone. An increase in residence time of the effluent on charcoal from 20 to 40 min removed greater than 99% of the COD.

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*Based on manufacturers' stated activity concentration.

[7] Y. LeGallic, CEA, Saclay, France, personal communication.

RADIOISOTOPE SALES

ITEMS OF INTEREST

An order was received from CEA, France, for 60 mCi of argon-37. An order was received from Instrumentation and Controls Division of ORNL for a 200-Ci promethium-147 source. An order was received from Radiochemical Centre, England, for 250 Ci of strontium-90 as carbonate prepared in ten cans of 25 Ci each.

Shipments of interest during the month are listed below:

Customer	Isotope	Amount
Large Quantities		
New England Nuclear Corporation	Tritium	4,000 Ci
Schwarz Bio-Research, Inc.	Tritium	1,000 Ci
U.S. Radium Corporation	Tritium	10,000 Ci
Canrad Precision, Inc.	Tritium	1,000 Ci
Radium-Chemie A. Zeller & Company	Tritium	20,000 Ci
Western Electric Company	Krypton-85	160 Ci
Items Shipped for Use in Cooperative Programs		
Oak Ridge Associated Universities	Gallium-67	
University of Southern California	Platinum-195m	
Michigan State University	Platinum-195m	
Specially Approved Withdrawn Items		
University of North Carolina	Irradiation of platinum-196	2 shipments
University of California	Iodine-131	30 mCi
University of Rochester	Iodine-131	425 mCi
Argonne National Laboratory	Copper-67	1 lot
Argonne National Laboratory	Copper-64	1 lot
Mayo Clinic	Copper-67	2 lots

Two graphite plates 20 cm × 40 cm × 1.5 mm thick, each coated with ∿4 Ci of strontium-90, were prepared for Brookhaven National Laboratory.

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

Table 5. Summary of Radioisotopes Shipped During April 1972

Radioisotope	Number of Shipments	Quantity (mCi)
Argon-37	1	10
Calcium-47	5	3
Cerium-144	15	20,000
Cesium-137	2	113,652,250
Iodine-129	1	1,000 mg

Table 5. Continued

Radioisotope	Number of Shipments	Quantity (mCi)
Iodine-131	3	225
Krypton-85, normal	5	195,000
Krypton-85, enriched	1	5,000
Neodymium-147	1	2
Niobium-95	2	2
Phosphorus-33	3	106
Praseodymium-143	3	7
Promethium-147	0	50,010,000
Strontium-89	1	5
Strontium-90	2	328,400,005
Tritium	15	32,054,500
Xenon-133	59	60,020
Yttrium-91	1	10
Zirconium-95	1	6
Experimental products	35	
Service irradiations		
Reactor	4	
	160	524,397,151
		1,000 mg

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

Table 6. Radioisotope Sales and Shipments

Item	7-1-70 thru 4-30-71	7-1-71 thru 4-30-72
Inventory items	$ 464,381	$446,426
Major products	102,757	52,238
Radioisotope services	228,819	257,302
Cyclotron irradiations	96,585	103,250
Miscellaneous processed materials	55,287	50,052
Packing and shipping	69,758	59,255
Total	$1,017,587	$968,523
Number of radioisotope shipments	2,608	2,004

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
Boston	To attend Institute of Nuclear Materials Management meeting

Table 8. IDC Visitors

Visitors (affiliation)	Subject Discussed
NASA-Marshall Space Flight Center	Fabrication of high-purity zinc samples
Max Planck Institut für Kernphysik Heidelberg, Germany	Preparation of beta-ray spectrometer sources and ion-exchange separations of rare-earth radionuclides
Bhabha Atomic Research Center Bombay, India	General discussions
Thayer School of Engineering Dartmouth University	Graduate research program: nuclear batteries
Physics Department Vanderbilt University	Writing of holmium-166m paper and gamma-gamma angular correlations paper
Belmont College Nashville, Tennessee	Tour and lecture on isotope production

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