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RADIOISOTOPE PROGRAM (8000) PROGRESS REPORT FOR JUNE 1973

A. F. Rupp

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ISOTOPES DEVELOPMENT CENTER

RADIOISOTOPE PROGRAM (8000) PROGRESS REPORT FOR JUNE 1973

A. F. Rupp

Work Sponsored by
AEC Division of Applied Technology

JULY 1973

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OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee 37830
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RADIOISOTOPE PROGRAM (8000) PROGRESS REPORT FOR JUNE 1973

A. F. Rupp

RADIOISOTOPE PRODUCTION AND MATERIALS DEVELOPMENT

REACTOR-PRODUCED ISOTOPES — 08-01-01

Potassium-43 *(J. K. Poggenburg)*

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 and to establish cooperative programs with medical institutions to evaluate it. Potassium-43 is potentially useful for metabolic and clinical studies of blood flow, rejection of transplanted organs, and kidney function. The half-life of potassium-43 is 22.5 hr; gamma-rays are 0.373 and 0.617 MeV.

Two batches of potassium-43 were processed this month and three shipments totaling 10.5 mCi were made.

The continuing interest in potassium-43 as a myocardial scan agent is indicated by the fact that five papers at the recent Annual Meeting of the Society of Nuclear Medicine in Miami related to potassium-43. The University of Mississippi group reported on their technique for evaluating coronary bypass grafts by serial scan images. Using dogs as experimental animals, they have recently tested a computer processing program which permits background subtraction of a first dose with capability of observing the course of a second potassium-43 administration the same day. One paper compared cesium-129 and potassium-43, with the conclusion that, although adequate imaging of infarcts can be obtained with either nuclide, a larger amount of cesium-129 activity must normally be given for comparable images, so that the patient dose is frequently greater with cesium-129. Other attempts at finding suitable myocardial scan agents (e.g., thallium-201, technetium-99m—labeled tetracycline) were reported, but none appear to have any distinct advantage over potassium-43. A report on potassium-43 production using a liquid argon target on a compact cyclotron led to the conclusion that yields are limited by the high concomitant production of 12.4-hr potassium-42 which requires extended decay allowances. The chief obstacles to full clinical use of potassium-43 are its cost and availability, and progress at removing these obstacles is reported under Accelerator Products.

Platinum-195m *(J. K. Poggenburg)*

> *Recent medical research has shown that some platinum compounds act as chemical therapeutic agents towards certain kinds of tumors. Platinum radioisotopes are needed for investigation of the therapeutic mechanism involved. Platinum-195m (4.0 d) has gamma emissions of 99 keV (11%) and 129 keV (2.8%) that make it suitable for whole-body scanning.*

We are working on the synthesis of inactive cis-dichlorodiammine-Pt(II) compound, using the procedures supplied by Wolfe of the University of Southern California, and we have not yet met with success using this lengthy multistep procedure.

Gadolinium-153 *(L. C. Brown)*

> *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 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.*

An earlier study[1] of methods for reactor production of gadolinium-153 concluded that irradiation of europium-151 targets (either natural or enriched) would be the method of choice. Subsequently, a method of recovering gadolinium target material isotopically enriched in gadolinium-152 from spent Eu_2O_3-Al HFIR control plates was developed[2,3] and utilized in producing curie quantities of high-purity gadolinium-153 with no detectable radionuclidic impurities. In an effort to determine whether the product character can be appreciably improved (i.e., increase the present 20-30 Ci/g specific activity to the maximum of ∿80 Ci/g) and whether the production technology can be extended to users that might not have access to enriched gadolinium-152 targets, a study of the original assumptions and rationale for using spent HFIR control plates has been performed within the context of the operating experience gained to date.

The nuclear reactions of importance for the production of gadolinium-153 from europium-151 are shown in Fig. 1. The capture cross section and burn-up cross section for gadolinium-152 and gadolinium-153, respectively, are effective cross-section values (values marked with asterisks) taken from earlier work by Case *et al.*[1] The branching ratios for europium-152

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[1]F. N. Case, E. H. Acree, and N. H. Cutshall, *Production of Gadolinium-153*, ORNL-TM-2632, Oak Ridge National Laboratory (August 1969), p. 6.
[2]A. F. Rupp, *Radioisotope Program (8000) Progress Report for July 1972*, ORNL-TM-3942, Oak Ridge National Laboratory, p. 2.
[3]A. F. Rupp, *Radioisotope Program (8000) Progress Report for November 1972*, ORNL-TM-4067, Oak Ridge National Laboratory, p. 2.

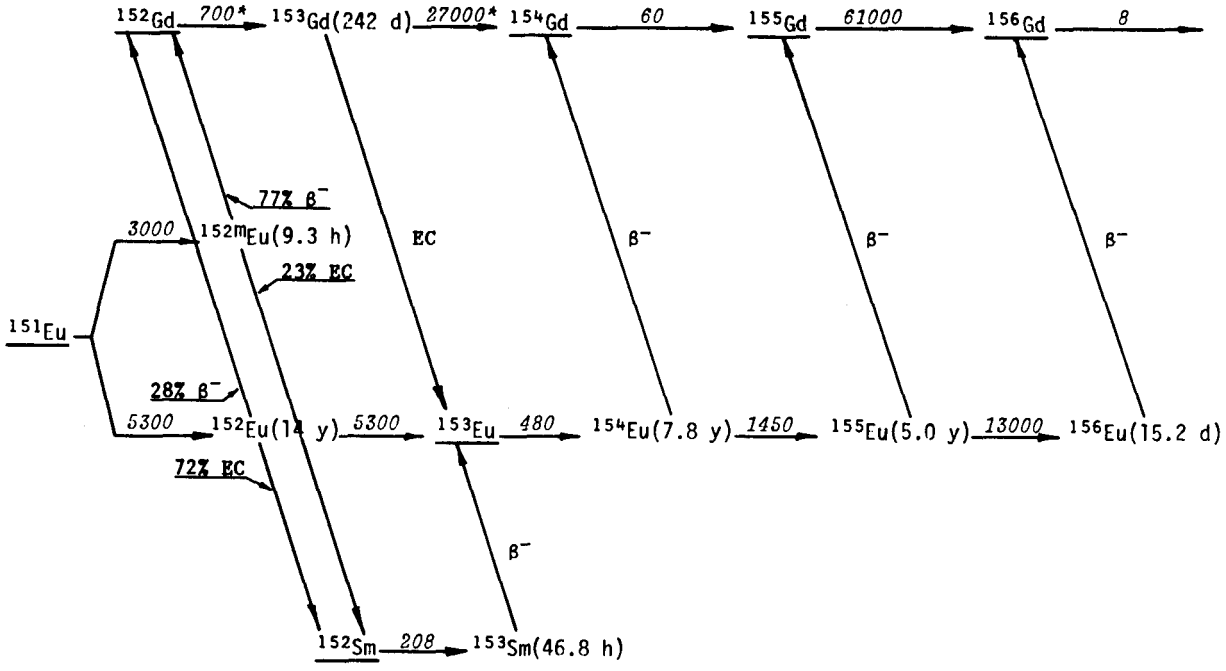

Fig. 1. Nuclear Reaction Paths
of Importance in the Production of Gadolinium-153.

and europium-152m were taken from the *Table of Isotopes*, 6th Ed.,[4] and all the other data were taken from the *Chart of the Nuclides*, 10th Ed.[5] The thermal neutron capture cross sections, except as noted above, are shown in italics on the horizontal arrows. Figure 1 has not been extended past A = 156 for two reasons: (1) a lack of data on the europium-156 capture cross section and (2) the small capture cross-section value for gadolinium-156. In the case of europium-151 irradiations designed to optimize gadolinium-153 production and specific activity, the contributions from higher mass gadolinium isotopes can be neglected.

The differential equations for the reaction paths shown in Fig. 1 were developed and programmed for a Wang 720C/711 programmable calculator/output system, and the yields and specific activities of gadolinium-153 (as Ci/g of target material and Ci/g of gadolinium, respectively) were determined at various neutron fluxes and irradiation times by an iterative-type calculation. The results of these calculations are shown in Fig. 2. The solid curves represent the specific activities for europium targets of natural isotopic abundance, whereas the dashed curves represent similar data for a 96.83 atom % isotopically enriched europium-151 target. The lower family of curves represents the yield of gadolinium-153 in Ci/g of europium target

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[4]C. M. Lederer, J. M. Hollander, and I. Perlman, *Table of Isotopes*, 6th Ed., J. Wiley and Sons, Inc., New York, 1967, p. 88.
[5]N. E. Holden and F. Walker, *Chart of the Nuclides*, 10th Ed., General Electric Company, Schenectady, New York, 1968.

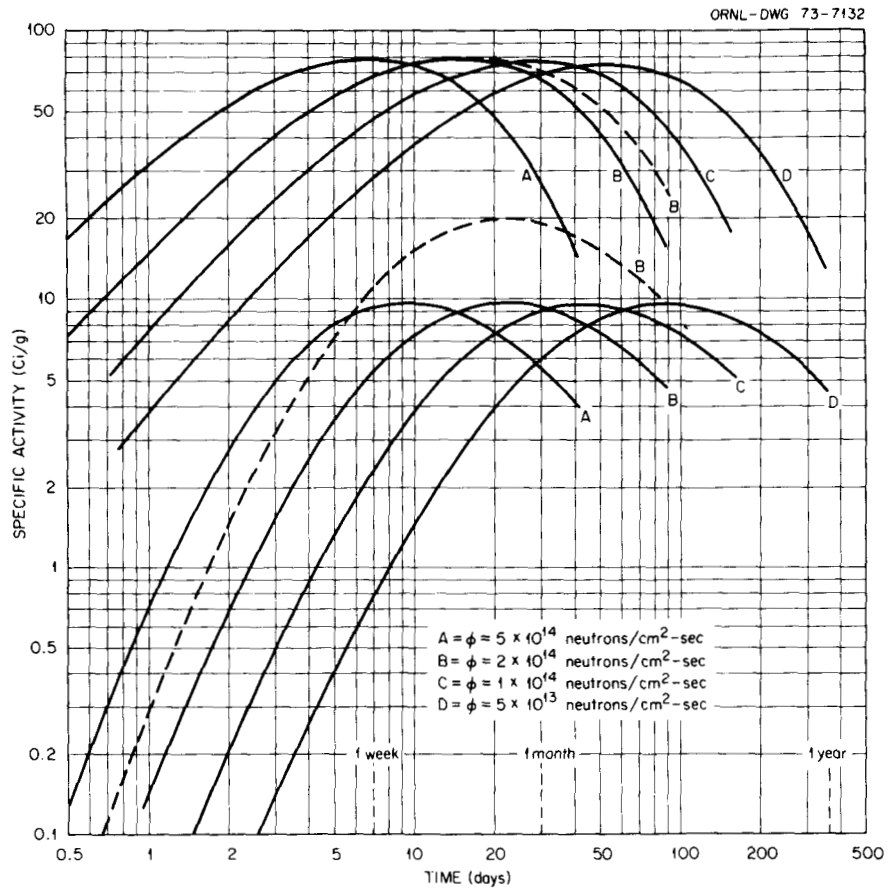

Fig. 2. Calculated Production of Gadolinium-153 From Europium Targets. [The upper and lower families of curves represent the specific activities (in Ci/g of gadolinium) and yields (in Ci/g of europium), respectively, for europium targets of natural isotopic abundance at different thermal neutron fluxes, whereas the dashed curves show similar data for a 96.83% isotopically enriched europium-151 target at a flux of 2×10^{14} n/cm^2·sec.]

material. The upper family represents the specific activity of gadolinium-153 in Ci/g of gadolinium which might be expected from the purified gadolinium product for source fabrication. It should be pointed out that these calculations do not include self-shielding or flux depression effects (which will be discussed later), both of which will be appreciable for europium-151. The main value of these idealized calculations is to show that if one wants to maximize the gadolinium-153 specific activity (i.e., 75-80 Ci/g based on the effective cross sections determined in ref. 1), then a short irradiation of a natural europium target should be employed rather than re-irradiation of gadolinium separated from highly irradiated HFIR control plates, which will contain appreciable amounts of higher mass gadolinium isotopes formed by multiple neutron capture reactions on europium-153. The use of europium targets of natural isotopic abundance may offer a number of advantages over the HFIR control plate material as outlined below:

1. The reduced radiation hazard of fabricating targets from inactive materials rather than moderate-to-high radiation dose gadolinium fraction — i.e., a Gd-Eu separation factor of $>10^4$ must be achieved with HFIR control plate material for safe handling.

2. The use of inactive europium targets would open up production of this radionuclide to reactors with low-to-moderate neutron fluxes, e.g., 1×10^{13} to 2×10^{14} n/cm^2·sec, without requiring access to enriched gadolinium-152 from HFIR control plate material. (It should be pointed out that high-level radiation handling facilities, i.e., hot cells, would still be required due to the high levels of europium-152, europium-154, and europium-155 that would be produced during the irradiation.)

3. The HDEHP solvent extraction separation[6] requirements for separating Gd-Eu can be reduced from a decontamination factor $>10^4$ to <100, since only the major portion of the europium must be removed to prevent overloading the high-pressure ion-exchange column used for the final Gd-Eu separation.[7]

4. The levels of europium radioactivities encountered would be reduced since the irradiations would be optimized for gadolinium-153 production rather than for europium burn-up as in the case of the spent HFIR control plate material.

5. Reliable gadolinium-153 yields and specific activities could be predetermined, whereas with the HFIR control plate material, these parameters vary greatly depending upon the region and integrated exposure of the plate sampled.

Other advantages such as the flexibility of inactive europium target inventory and management, the possibility of lower target fabrication costs relative to handling highly radioactive HFIR control plate material, increased product specific activity, etc., will need to be proven by actual operating experience.

The two main disadvantages of this production scheme are the self-shielding and flux depression due to the large europium-151 cross section (σ = 8300 b). The self-shielding problem can be minimized by diluting the Eu_2O_3 target material with an appreciable fraction, e.g., >90% by weight, of a low cross-section material such as aluminum or carbon. This approach has been used before with enriched europium-151 targets and found to work well. Resolution of the flux depression problem will depend upon the reactor used for the irradiation and the limits imposed on inserting "black" absorbers. Once a design has been approved by the reactor operating personnel, a few empirical measurements of the effective depressed flux should be made, and then the irradiation time adjusted to yield the maximum specific activity as shown in Fig. 2.

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[6]A. F. Rupp, ORNL-TM-3942, p. 2.
[7]A. F. Rupp, ORNL-TM-4067, p. 2.

There is little or no advantage in the use of enriched europium-151 targets with regard to gadolinium-153 specific activity unless one allows the irradiation to progress past the maximum value (see Fig. 2). In that case, the specific activity falls off less rapidly (as shown by the upper dashed curve in Fig. 2) since the higher mass gadolinium isotopes are formed more slowly. The apparent higher total production capacity of gadolinium-153 is not attainable since the europium-151 cross section limits the amount of target material that can be encapsulated into a target. The europium-153 acts mainly as a cross-section diluent, so long as the irradiation parameters are optimized for maximum gadolinium-153 specific activity, and the savings in target costs of natural versus enriched obviate the use of the latter. The use of an enriched europium-151 target might be rationalized if high-purity europium-152 were desired as a by-product from the irradiation.

Reactor Products Pilot Production *(R. W. Schaich)*
(Production and Inventory Accounts)

Processed Units	
Radioisotope	Amount (mCi)
Calcium-47	22.37
Copper-67	22.96
Zinc-69m	266

ACCELERATOR-PRODUCED ISOTOPES — 08-01-02

Potassium-43 *(L. C. Brown and J. K. Poggenburg)*

Potassium-43 has been shown to be an excellent tool for studying blood perfusion in the heart and for diagnosing myocardial infarctions. Potassium-43 is currently produced in millicurie quantities in the HFIR by the $^{43}Ca(n,p)^{43}K$ reaction. The objective of this project is to investigate the large-scale photonuclear production of this radionuclide using the Oak Ridge Electron Linear Accelerator (ORELA). The ultimate goal is to develop the technology required to produce curie quantities of potassium-43 at costs that will allow its widespread clinical application as a primary agent for the early detection and diagnosis of heart disease.

A second target for assessing the practical potassium-43 production rates at ORELA has been designed and fabricated. It consists of twenty-two 14 × 14 × 1 in. slabs of white marble held together in a stainless steel frame, and enclosed in plastic to prevent the possible spread of loose contamination. The target face consists of a thin aluminum sheet to which a 0.105-in.-thick stack of tantalum foils has been affixed to act as a bremsstrahlung converter of the incident electron beam. This target will be irradiated in

early July in the same low power mode as the experiment described earlier.[8] It will be desirable to extend the length of irradiation relative to the earlier experiment in order to produce sufficient potassium-43 to image the localized distribution within the target volume. This will be done by disassembling the target and scanning the marble slabs using the rectilinear NaI scanning system of the ORNL Medical Instrumentation Group. The geometry factors required for determining the amount of potassium-43 produced will be obtained by attaching a source with a known amount of potassium-43 activity and repeating the scan. A preliminary test has been made with the scanner to confirm that adequate spatial resolution of a finite source can be obtained through the marble slab. This will allow quantification of the relative isoproduction curves.

Bismuth-206 *(L. C. Brown)*

> *The objectives of this program are to evaluate the pertinent production parameters and to provide bismuth-206 (6.24 d) for evaluation as a diagnostic radionuclide in medicine. Bismuth-206 has been used to delineate brain tumors in patients with known brain tumors which are poorly visualized by more common radionuclides. Even though very favorable tumor-to-background activity ratio has been obtained by earlier workers, the lack of appropriate scanning systems suitable for the high-energy emissions of this nuclide has retarded the evaluation of the isotope. Cooperative programs have been established for the evaluation of bismuth-206 as a diagnostic radionuclide using high-energy photon scanning systems.*

Test irradiation of a high-yield, electrodeposited lead-207-on-copper target design has been continued in order to optimize irradiation conditions and to provide high-purity bismuth-206 to medical cooperative participants for evaluation. A target of 7.6-mil average lead-207 plate thickness (92.4 atom % lead-207) was irradiated with a 22-MeV proton beam at an average current of 250 μA without failure. The loss of bismuth-206 during chemical processing and product finishing mentioned last period[9] was reduced to less than 2%. Optimization of the enriched lead-207 target irradiation conditions, chemical process testing under actual operating conditions, extensive product characterization, and process quality control development will be continued during the next report period.

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[8] L. C. Brown and J. K. Poggenburg, "Potassium-43," in *Radioisotope Program (8000) Progress Report for May 1973*, ORNL-TM-4281, Oak Ridge National Laboratory, pp. 3-4.

[9] L. C. Brown, "Bismuth-206," in *Radioisotope Program (8000) Progress Report for May 1973*, ORNL-TM-4281, Oak Ridge National Laboratory, p. 5.

Rare-Earth Radionuclides *(L. C. Brown)*

Ytterbium-169 has been found to exhibit high tumor-to-nontumor uptake ratios, comparable to the uptake ratios of gallium-67. A cooperative project has begun with the Medical Division of ORAU to determine whether all the rare earths show this tumor localization, or whether it is limited to either the light (La-Gd) or heavy (Tb-Lu) members of this series, and to evaluate the most promising radionuclides for tumor localization in humans.

As part of a cooperative arrangement with Battelle Pacific Northwest Laboratories for the preparation of high-purity promethium-143 for human excretion experiments,[10,11] two additional $^{144}Sm_2O_3$ irradiations were performed at the ORNL 86-Inch Cyclotron. The cyclotron yields for an irradiation performed 3 years ago[12] and for this recent set of three irradiations are shown in Table 1. The proton energy on the target is estimated to be ~20.5 MeV at a 31-5/8-in. radius versus ~22 MeV at the 32-in. maximum radius. The yields from the recent set of irradiations seem to be consistent and reproducible, even though much lower than the earlier value. The rationale for the discrepancy between these two sets of data is unknown at present, and will be investigated further if additional promethium-143 is required by BPNL in the future.

Table 1. ORNL 86-Inch Cyclotron Yields for the $^{144}Sm(p,2n)^{143}Eu \xrightarrow{\beta^+,\ EC} {}^{143}Sm \xrightarrow{\beta^+,\ EC} {}^{143}Pm$ Reaction Path

Sample	Target Position	Machine Yield	Reference
Pm-143-1	32 in.	0.44 μCi/μA-hr	12
Pm-143-2	31-5/8 in.	0.17 μCi/μA-hr	11
Pm-143-3	31-5/8 in.	0.16 μCi/μA-hr	New
Pm-143-4	32 in.	0.20 μCi/μA-hr	New

[10] L. C. Brown, "Rare-Earth Radionuclides," in *Radioisotope Program (8000) Progress Report for March 1973*, ORNL-TM-4214, Oak Ridge National Laboratory, p. 4.

[11] L. C. Brown, "Rare-Earth Radionuclides," in *Radioisotope Program (8000) Progress Report for April 1973*, ORNL-TM-4229, Oak Ridge National Laboratory, p. 4.

[12] A. F. Rupp, *Radioisotope Program (8000) Progress Report for March 1970*, ORNL-TM-2965, Oak Ridge National Laboratory, p. 9.

Cyclotron Products Pilot Production *(M. R. Skidmore)*
(Production and Inventory Accounts)

June 1963 ORNL 86-Inch Cyclotron runs for ORNL and non-ORNL programs are given in Table 2.

Table 2. Cyclotron Irradiations and Runs for June 1973

Date	Customer	Product	Target	Total Time (hr:min)	Total Charges
		ORNL Programs			
5-21-73	Isotopes Division	Promethium-143	Samarium-144 Oxide	17:15	$1652
5-24-73	ORAU and Others	Gallium-67	Zinc-68	9:25	902
5-31-73	ORAU and Others	Gallium-67	Zinc-68	9:30	909
6-14-73	ORAU and Others	Gallium-67	Zinc-68	9:45	933
6-26-73	Isotopes Division	Bismuth-206	Lead-207	2:15	221
				48:10	$4617
		Non-ORNL Programs			
		None for reporting period			
		Sales Department Inventory			
6-12-73	Isotopes Division	Cobalt-57	Nickel	54:15	$7822

FISSION PRODUCTS — 08-01-03

Cesium-137 Pilot Production *(R. W. Schaich)*
(Production and Inventory Accounts)

1. Process Status

Cesium processing has been discontinued and the cesium manipulator cells are being used as a high-level waste storage and loadout facility. The current cesium-137 inventory status is shown below.

Item	Cesium-137 (Ci)[a]
In-process material	0
Cesium-137 chloride products	574,200
Returned sources[b]	134,700
Completed sources awaiting shipment	125,300
	834,200

[a]All values based on physical inventory, May 1, 1973.
[b]Including cancelled orders.

2. Operational Summary

Item	June 1973 No.	June 1973 Ci	CY 1973 No.	CY 1973 Ci	FY 1973 No.	FY 1973 Ci
HAPO shipments received	0	0	0	0	0	0
Product batches prepared	0	0	17	225,000	47	532,000
Sources						
Fabricated	0	0	10	22,800	36	48,500
Shipped	1	8,000	10	22,800	36	48,500
Special Form Cans						
Fabricated	2	20,046	43	25,200	53	25,700
Shipped	34	22,106	40	23,106	44	24,506

3. Current Orders

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

Customer	Amount (Ci)	Estimated Shipping Date
J. L. Shepherd Associates[a]	8,000	July 1973
White Sands Missile Base	14,221	July 1973

[a]Source is completed awaiting receipt of customer's container.

Strontium-90 Pilot Production *(R. W. Schaich)*
(Production and Inventory Accounts)

1. Process Status

Preliminary decontamination of the strontium titanate processing cell has been completed and preparations have been completed to pull the calcination furnace from the cell. Further decontamination will be required to refurbish this cell for future strontium-90 processing.

Preparations are complete for hot pressing 46,000 Ci of strontium titanate to fabricate a heat source for Messerschmitt-Borkow-Blohm.

The current strontium-90 process status has been revised to agree with the annual physical inventory of May 1, 1973, and is shown below:

Item	Strontium-90 (Ci)
In-process material	766,800
Strontium-80 products	430,400[a]
Sources in fabrication	0
Returned sources	435,200
Completed sources awaiting shipment	214,400
	1,846,800

[a]Includes 30,500 Ci of $^{90}SrSiO_3$.

2. Operational Summary

Item	June 1973 No.	June 1973 Ci	CY 1973 No.	CY 1973 Ci	FY 1973 No.	FY 1973 Ci
HAPO shipments received	0	0	0	0	0	0
Product batches prepared	0	0	0	0	0	0
Sources						
Fabricated	0	0	0	0	0	0
Shipped	0	0	0	0	0	0
Special Form Cans						
Fabricated	17	400	71	1700	71	1700
Shipped	67	1717	72	1817	72	1817

3. Current Orders

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

Customer	Amount (Ci)	Estimated Shipping Date
U. S. Navy	221,000	a

[a]Sources are complete; customer has been billed; and sources are awaiting receipt of DOT Permit to ship.

Short-Lived Fission Products Pilot Production
(Production and Inventory Accounts) *(R. W. Schaich)*

Isotope	Number of Batches	Amount (Ci)
Xenon-133	2	500
Iodine-131	1	22
Niobium-95	1	14

Promethium-147 Shipments and Current Orders

Customer	Amount (Ci)	Estimated Shipping Date
W. Maier K.G., Germany	100	July 1973
Radium Chemie, Switzerland	300	July 1973
3M Company	5400	July 1973

RADIOISOTOPE SALES

J. E. Ratledge

An order was received from Radiochemical Centre, England, for 20 kCi cesium-137 as CsCl powder. Orders were received from Nuclear Sources and Services for 210 Ci krypton-85 and from Texas Instruments, Inc., for 200 Ci krypton-85. An order was received from Minnesota Mining and Manufacturing Company for 5400 Ci promethium-147 as Pm_2O_3. A request for quotation was received from Teledyne Isotopes, Inc., for one strontium-90 source containing approximately 50 kCi.

Shipments made during the month that may be of interest are listed below:

Customer	Isotope	Amount
Large Quantities		
New England Nuclear Corporation	Tritium	10,000 Ci
U. S. Radium Corporation	Tritium	10,000 Ci
Texas Instruments, Inc.	Krypton-85	200 Ci
Nuclear Sources and Services	Krypton-85	210 Ci
Canrad Precision Industries	Tritium	2,000 Ci
General Radioisotope Products	Strontium-90	237 Ci
Radiochemical Centre	Strontium-90	1,506 Ci
Radiochemical Centre	Cesium-137	21,621 Ci
Minnesota Mining & Manufacturing Company	Cesium-137	500 Ci
Withdrawn Items		
Mayo Clinic	Copper-67	10 mCi
George Washington University	Copper-67	3 mCi
Cleveland Metropolitan General Hospital	Iodine-131	50 mCi
University of Pittsburgh	Iodine-131	50 mCi
University of Rochester	Iodine-131	100 mCi
University of California, Livermore	Cobalt-60	1,100 Ci
Items Used in Cooperative Programs		
Oak Ridge Associated Universities	Gallium-67	520 mCi
University of Southern California	Platinum-195m	~15 mCi
Unusual Items		
Technischen Universitat, Munchen, Germany	Tungsten[a]	5 mg

[a]Enriched in tungsten-180; irradiated in HFIR.

The radioisotope sales proceeds and shipments for the first 11 months of FY 1972 and FY 1973 are given in Table 3.

Table 3. Radioisotope Sales and Shipments

Item	7-1-71 thru 5-31-72	7-1-72 thru 5-31-73
Inventory Items	$ 477,504	$ 449,350
Major Products	57,862	50,966
Radioisotope Services	278,402	208,007
Cyclotron Irradiations	117,782	104,252
Miscellaneous Processed Materials	56,336	55,933
Packing and Shipping	65,738	62,177
Total	$1,053,624	$ 930,685
Number of Shipments	2,219	1,943

A summary of radioisotopes shipped during May 1973 is given in Table 4.

Table 4. Summary of Radioisotopes Shipped During May 1973

Isotope	Number of Shipments	Quantity (mCi)
Calcium-47	7	6
Cesium-137	3	6,200
Cobalt-60	1	25
Copper-67	5	33
Gadolinium-153	4	450
Gallium-67	14	3,183
Hafnium-181	1	2
Iodine-129	2	551 mg
Iodine-131	3	400
Krypton-85, normal	12	173,000
Krypton-85, enriched	1	39,000
Lead-203	1	8
Niobium-95	1	10
Phosphorus-33	4	579
Potassium-43	8	25
Promethium-147	10	50,100,000
Ruthenium-106	1	1
Samarium-151	1	8 mg
Strontium-89	1	1 μCi
Strontium-90	5	100,105
Technetium-99	1	2,000 mg
Tritium	25	14,900,002
Xenon-133	55	59,110
Miscellaneous preparations and targets	8	658
Cyclotron irradiations:		
New England Nuclear Corp.	1[a]	2,000
Philips-Duphar	1[a]	400
Totals	176	65,385,197 2,559 mg

[a]Cobalt-57.

ADMINISTRATIVE

Visitors to the IDC and travel by IDC personnel are given in Tables 5 and 6, respectively.

Table 5. Visitors to IDC

Visitor	Affiliation	Subject Discussed
Milt Lewis	Donald W. Douglas Laboratories Richland, Washington	Promethium-147 burn-up

Table 6. Travel by IDC Personnel

Traveler	Site Visited	Purpose of Visit
J. C. Posey	Detroit, Michigan	Present paper at AIChE Meeting
A. F. Rupp	Detroit, Michigan	Attend AIChE Meeting
J. K. Poggenburg	Miami, Florida	Attend Annual Meeting of Society of Nuclear Medicine
H. R. Brashear, F. N. Case, and J. H. Gillette	Norfolk, Virginia	Confer with ship personnel concerning NOAA Sand Tracing Study (Work for Others)
H. R. Brashear and F. N. Case	Miami, Florida	Discuss New York Bight sand tracing experiment with NOAA personnel (Work for Others)

PUBLICATIONS

REPORTS

A. F. Rupp, *Report of Foreign Travel — Symposium on New Developments in Radiopharmaceuticals and Labelled Compounds, Copenhagen, Denmark, March 26-30, 1973*, ORNL-CF-73-5-52, Oak Ridge National Laboratory (May 15, 1973).

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